Final Submittal

BROWNS FERRY RETAKE
EXAM 50-25912003-301
ADMIN PART A

MAY 8,2003

Final Admin JPMs & Outline ES-301-1

FINAL

acility: BFN Date of Examination: 05/08/2003
xamination Level (circle one): SRO Operating Test Number: Remedial

Administrative Topic/Subject Description		Describe method of evaluation: 1. ONE Administrative JPM, OR 2. TWO Administrative Questions
.1	MODECHANGE	JPM A1.1R TECH SPEC 3.0.4 DETERMINATION
	PERFORM SPECIFIC AND INTEGRATED PLANT PROCEDURE	JPM A1.2R DETERMINE REGULATORY REPORTING REQUIREMENT
4.2	SURVEILLANCE TESTING	JPM A2.1R CORE SPRAY SYSTEM VALVE TIMING
4.3	RADIATION EXPOSURE LIMITS AND RADIATION CONTROL	JPM A3.1R REVIEW AND INTERPRET A RADIOLGOICAL SURVEY MAP
A.4	EMERGENCY ACTION LEVEL /COMMUNICATIONS	JPM A4.2 R CLASSIFY EVENT PER REP PROCEDURE

FINAL

Facility: Date of Examination: Operating Test Number:

	1. GENERAL CRITERIA	Initials a	b*	c#
a.	The operating lest conforms with the previously approved outline; changes are consistent with sampling requirements (e.g., 10 CFR 55.45, operational importance. safety function distribution).	Rm	RK	EL
b.	There is no day-to-day repetition between this and other operating tests to be administered during this examination.	Rm	RK	EL
c.	The operating test shall not duplicate items from the applicants' audit test(s)(see Section D.1.a).	Rm	RK	EL
d.	Overlap with the written examination and between operating test categories is within acceptable limits.	Rm	RK	EL
e.	It appears that the operating test will differentiate between competent and less-than-competent applicants at the designated license level.	Rm	RK	EL
	2. WALK-THROUGH (CATEGORY A B B) CRITERIA	--	--	--
a.	Each JPM includes the following, as applicable: . initial conditions , initiating cues . references and tools, including associated procedures . reasonable and validated time limits (average time allowed for completion) and specific designation if deemed to be time critical by the facility licensee . specific performance criteria that include: - detailed expected actions with exact criteria and nomenclature - system response and other examiner cues - statements describing important observations to be made by the applicant - criteria for successful completion of the task - identification of critical steps and their associated performance standards - restrictions on the sequence of steps. if applicable	Rm	RK	EL
b.	The prescripted questions in Category A are predominantly open reference and meet the criteria in Attachment 1 of ES-301.	Rm	RK	EL
c.	Repetition from operating tests used during the previous licensing examination is within acceptable limits (30% for the walk-through) and do not compromise test integrity.	Rm	RK	EL
d.	At least 20 percent of the JPMs on each test are new or significantly modified.			
	3. SIMULATOR (CATEGORY C) CRITERIA	--	--	--
a.	The associated simulator operating tests (scenario sets) have been reviewed in accordance with Form ES-301-4 and a copy is attached.	N/A	NA	N/A

	Printed Name / Signature	Date
a. Author	ROBERT H MCDOWELL / R H McDowell	4-28-03
b. Facility Reviewer(*)	RANDY E. KNIGHT / RK	4-28-2003
c. NRC Chief Examiner (#)	Edwin Lea / Edwin Lea Jr.	4/30/2003
d. NRC Supervisor	MICHAEL E. ERNSTES / Michael E. Ernstes	5/1/2003

NOTE: * The facility signature is not applicable for NRC-developed tests.
Independent NRC reviewer initial items in Column "c;" chief examiner concurrence required.

BROWNS FERRY NUCLEAR PLANT
JOB PERFORMANCE MEASURE

JPM NUMBER: A1.1R

TITLE. TECH SPEC DETERMINATION – 3 0.4

TASK NUMBER: S-000-AD-27

SUBMITTED BY:____ DATE:____

VALIDATED BY:____ DATE:____

APPROVED: ____ DATE:____
TRAINING

PLANT CONCURRENCE: ________________________________ DATE:____
OPERATIONS

* Examination JPMs Require Operations Training Manager or Designee Approval and Plant Concurrence

BROWNS FERRY NUCLEAR PLANT
JOB PERFORMANCE MEASURE

OPERATOR:________________________________

RO ____ SRO _X__ DATE:_______

JPM NUMBER: A1.1R

TASK NUMBER: S-000-AD-27

TASK TITLE: TECH SPEC DETERMINATION – 3.0.4

WA NUMBER: 2.1.12 *WA* RATING: RO_2.9_ SRO: _4.0_

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TASK STANDARD: Determine that a mode change from Mode 2 to Mode 1 with inoperable. required equipment is not allowed in accordance with Technical Specification 3.0.4.

LOCATION OF PERFORMANCE: SIMULATOR _x_ PLANT _x_ CONTROL ROOM x_

REFERENCES/PROCEDURES NEEDED: Unit 2 technical Specifications, 3.0.4, 3.5.1

VALIDATION TIME: CONTROL ROOM: _N/A_ LOCAL: _N/A_

MAX. TIME ALLOWED: _NIA_. (Completed for Time Critical JPMs onlyj

PERFORMANCE TIME: _NIA_ CONTROL ROOM _N/A_ LOCAL _N/A_

COMMENTS:____

____

____

Additional comment sheets attached? YES __ NO __

RESULTS: SATISFACTORY ___ UNSATISFACTORY ___

EXAMINER SIGNATURE:____________________ DATE: ___
EXAMINER

BROWNS FERRY NUCLEAR PLANT
JOE PERFORMANCE MEASURE

EXAMINER'S KEY
REFERENCES ALLOWED

INITIAL CONDITION

The current time *is* 1900 hrs, Unit 2 **is** starting up from cold shutdown and is now holding at 8% rated thermal power with the Mode Switch in Startup. An oil leak was discovered on **2A** RHR pump motor at 1500 hours today. A 7 day **LCO** was entered. Mainterance was performed and the oil leak repaired at 1800 hours today The Hold Order has been released. **Cleanup** and post maintenance testing is expected be completed by 2100 hours today.

INITIATING CUE:

Determine If reactor stamp to 30% rated thermal power may continue while cleanup and post maintenance testing to prove **2A** WHR pump operable is in progress.

Indicate the documentation which **defends your answer**

EXPECTED RESPONSE (exact wording not required)

No, this would require a mode change from Mode 2 to Mode I with inoperable, required ECCS equipment.

REFERENCES:

Knowledge item

The operator will have to reach the conclusion that for the startup to continue to ?he desired power level that **entry** into Mode 1 would be required. This will required application of LCO 3.0.4 and LCQ 3.5.1

Tech Spec LCO 3.0.4(Page 3.0-2)

When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when ?he associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are a part of a shutdown of the unit.

Tech Spec LCO 3.0.4 Basis (Page B3.0-6)

The basis fer LCO 3.0.4 states, "LCO 3.0.4 establishes limitation on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It precludes placing the unit in a different Mode or other specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist:

a. Unit conditions are such that requirements of the LCO would not be met in the Applicability desired to be entered; and
b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in the unit being required to exit the Applicability desired to !)e entered to comply with the Required Actions."

Tech **Spec LCO 3.5.1 ECCS**

The applicability of LCO 3.5.1 is Mode 1 and Mode 2 & 3 (with an exception for HPCl and ADS at <150psig). This LCQ requires ALL ECCS systems to be operable. Current Mode is Mode 2 and for startup to continue would require entry into Mode 1. LCO 3.5.1 Condition A requires the inoperable system to be restored within 7 days, condition B would be entered if the subsystem were not restored and would require the Unit be placed in Mode 3 then Mode 4.

Conclusion:
Although it is expected that RHR would be operable and LCO 3.5.1 met prior to exceeding the 7 day completion time of 3.5.1.A, LCO 3.0.4 forces the operator to consider that the inoperability would be "continued noncompliance" and ultimately action statement 3.5.1.B would be entered which would require exit from Mode 1. For the current plant conditions and application of TS 3.0.4 and TS 3.5.1, the startup can not continue because entry into Mode 1 would be required.

CANDIDATE'S HANDOUT

REFERENCES ALLOWED

INITIAL CONDITION

The current time is 1900 hours, Unit 2 is starting up from cold shutdown and is now holding at 8% rated thermal power with the Mode Switch in Startup. An oil leak was discovered on 2A WHR pump motor at 1500 hours today. A 7 day LCO was entered. Maintenance was performed and the oil leak repaired at 1800 hours today. The Hold Order has been released. Cleanup and post maintenance testing is expected to be completed by 2100 hours today.

INITIATING CUE:

Determine if reactor startup to 30% rated thermal power may continue while cleanup and post maintenance testing to prove 2A WHR pump operable is in progress.

Indicate the documentation which defend5 your answer.

3.0 LCO APPLICABILITY (continued)

LCO 3.0.4

When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MQDE or other specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are a part of a shutdown of the unit.

Exceptions to this Specification are stated in the individual Specifications.

LCO 3.Q.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, and 3

LCO 3.0.5

Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

LCO *3.0.6*

When a supported system LCO is not met solely due to *a* support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to **be** entered. Phis is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.11, "Safety Function Determination Program (SFDP)." If a loss of safety function is

(continued)

3 5 EMERGENCY COKE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3 5 1 ECCS - Operating

LCO 3.5 1 Each ECCS injectionlspray subsystem and the Automatic Depressurization System (ADS) function of six safetykelief valves shall be OPERABLE.

APPLICABILITY MODE 1,
MODES 2 and 3 except high pressure coolant injection (HPCI) and ADS valves are not required to be OPERABLE with reactor steam dome pressure < 150 psig

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One low pressure ECCS injectionlspray subsystem inoperable. OR One low pressure coolant injection (LPCI) pump in both LPCI subsystems inoperable.	\.1 Restore low pressure ECCS injectionlspray subsystem(s) to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met.	3.1 Be in MODE 3. AND 3.2 Be in MODE 4	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. HPCI System inoperable.	C 1	Verify by administrative means RCIC System is OPERABLE	Immediately
	AND		
	C 2	Restore HPCI System to OPERABLE status	14 days
D. HPCI System inoperable. AND Condition A entered	D 1	Restore HPCI System to OPERABLE status	72 hours
	OR		
	D 2	Restore low pressure ECCS injection/spray subsystem to OPERABLE status	72 hours
E One ABS valve inoperable	E.1	Restore ABS valve to OPERABLE status.	14 days
F. One ADS valve inoperable. AND Condition A entered	F 1	Restore ADS valve to OPERABLE status	72 hours
	OR		
	F 2	Restore low pressure ECCS injection/spray subsystem to OPERABLE status	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. Two or more ADS valves inoperable. OR Required Action and associated Completion Time of Condition C, D, E, or F not met.	G.1 Be in MODE 3. AND G.2 Reduce reactor steam dome pressure to ≤ 150 psig.	12 hours 36 hours
H. Two or more low pressure ECCS injection/spray subsystems inoperable for reasons other than Condition A. OR HPCI System and one or more ADS valves inoperable	H.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify, for each ECCS injection/spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve.	31 days
SR 3.5.1.2	-------------------------NOTE------------------------ Low pressure coolant injection (LPCI) subsystems may be considered OPERABLE during alignment and operation for decay heat removal with reactor steam dome pressure less than the Residual Heat Removal (RHR) tow pressure permissive pressure in MODE 3. if capable of being manually realigned and not otherwise inoperable. --- Verify each ECCS injection/spray subsystem manual, power operated, and automatic valve in the flow path, that is riot locked, sealed, or otherwise secured in position, is in the correct position.	 31 days
SR 3.5.1.3	Verify ADS air supply header pressure is $\geq$ 81 psig.	31 days
SR 3.5.1.4	Verify the LPCI cross tie valve is closed and power is removed from the valve operator.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.5.1.5	-------------------------NOTES------------------------- 1. Only required to be performed when in MODE 4 > 48 hours. -- Verify each recirculation pump discharge valve cycles through one complete cycle of full trave!.	Once prior to entering MODE 2 from MODE 3 or 4
	specified flow rate against a system head corresponding to the specified pressure.	with the Inservice Testing Program

SYSTEM	FLOW RATE,	NO OF PUMPS	SYSTEM HEAD CORRESPONDING TO A VESSEL TO TORUS DIFFERENTIAL PSESSURE OF
Core Spray	≥ 6250 gpm	2	≥ 105 psid

SYSTEM	FLOW RATE	NO OF PUMPS	INDICATED SYSTEM PRESSURE
LPCI	≥ 12,000 gpm	2	≥ 250 psig
LPCI	≥ 9,000 gpm	1	≥ 125 psig

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.5.1.7	-------------------------NOTE------------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify, with reactor pressure ≤ 1040 and ≥ 950 psig, the HPCI pump can develop a flow rate > 5000 gpm against a system head corresponding to reactor pressure.	92 days
SR 3.5.1.8	-------------------------NOTE------------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify, with reactor pressure ≤ 165 psig, the HPCI pump can develop a flow rate ≥ 5000 gpm against a system head corresponding to reactor pressure.	24 months
SR 3.5.1.9	-------------------------NOTE------------------------- Vessel injection/spray may be excluded. -- Verify each ECCS injectionlspray subsystem actuates on an actual or simulated automatic initiation signal.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.10	-------------------------NOTE------------------------- Valve actuation may be excluded. -- Verify the ABS actuates on an actual or simulated automatic initiation signal.	24 months
SR 3.5.1.11	-------------------------NOTE------------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify each ADS valve opens when manually actuated.	24 months
SR 3.5.1.12	Verify automatic transfer of the power supply from the normal source to the alternate source for each LPCI subsystem inboard injection valve and each recirculation pump discharge valve.	24 months

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5 2 ECCS - Shutdown

LCO 3.5.2 Two low pressure ECCS injectionlspray subsystems shall be OPERABLE.

APPLICABILITY: MODE **4,**
MODE 5, except with the spent fuel storage pool gates removed and water level ≥ 22 ft over the top of the reactor pressure vessel flange

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required ECCS injectionlspray subsystem inoperable.	A.1	Restore required ECCS injectionlspray subsystem to OPERABLE status.	*4* hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Initiate action to suspend OPDRVs.	Immediately
C. Two required ECCS injectionispray subsystems inoperable.	C.1	Initiate action to suspend OPDRVs.	Immediately
	AND		
	C 2	Restore one ECCS injection/spray subsystem to OPERABLE status	4 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action C.2 and associated Completion Time not met.	8.1	Initiate action to restore secondary containment to OPERABLE status.	immediately
	AND		
	D.2	Initiate action to restore two standby gas treatment subsystems to OPERABLE status.	Immediately
	D.3	Initiate action to restore isolation capability in each required secondary containment penetration flow path net isolated.	immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify, for each required ECCS injection/spray subsystem, the suppression pool water level is ≥ -6.25 inches with or -7.25 inches without differential pressure, control.	12 hours
SR 3.5.2.2	Verify, for each required ECCS injectionispray subsystem; the piping is filled with water from the pump discharge valve to the injection valve.	31 days
SR 3.5.2.3	---------------------------NOTE--------------------------- One LPCI subsystem may be considered OPERABLE during alignment and operation for decay heat removal if capable of being manually realigned and not otherwise inoperable. -- Verify each required ECCS injection/spray subsystem manual: power operated: and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.4	Verify each required ECCS pump develops the specified flow rate against a system head corresponding to the specified pressure. SYSTEM / FLOW RATE / NO OF PUMPS / SYSTEM HEAD CORRESPONDING TO A VESSEL TO TORUS DIFFERENTIAL PRESSURE OF CS / ≥ 6250 gpm / 2 / ≥ 105 psid SYSTEM / FLOW RATE / NO OF PUMPS / INDICATED SYSTEM PRESSURE LPCI / ≥ 9,000 gpm / 1 / ≥ 125 psig	In accordance with the Inservice Testing Program
SR 3.5.2.5	-------------------------NOTE------------------------- Vessel injection/spray may be excluded. --- Verify each required ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal.	24 months

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3 5 3 RCIC System

LCO 3.5.3 The RCIC System shall be OPERABLE

APPLICABILITY: MODE. 1,
MODES 2 and 3 with reactor steam dome pressure > 150 psig

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. RCIC System inoperable.	A.I	Verify by administrative means High Pressure Coolant Injection System is OPERABLE.	Immediateiy
	AND		
	A.2	Restore RCIC System to OPERABLE status.	14 days
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	12 hours
	AND		
	8.2	Reduce reactor steam dome pressure to $\leq$ 150 psig.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	Verify the RCIC System piping is filled with water from the pump discharge valve to the injection valve.	31 days
SR 3.5.3.2	Verify each RCIC System manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.3.3	------------------------NOTE------------------------ Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with reactor pressure ≤ 1040 psig and ≥ 950 psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure	32 days
SR 3.5.3.4	------------------------NOTE------------------------ Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with reactor pressure ≤ 165 psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.5.3.5	-------------------NOTE------------------- Vessel injection may be excluded. -- Verify the RCIC System actuates on an actual or simulated automatic initiation signal.	24 months

BASES

LCO 3.0.3 (continued)

Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.6 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.6 of "Suspend movement of irradiated fuel assemblies in the spent fuel storage pool" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It precludes placing the unit in a different MODE or other specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist:

a. Unit conditions are such that requirements of the LCO would not be met in the Applicability desired to be entered; and

b. Continued noncompliance with the LCO requirements, if *the* Applicability were entered, would result in the unit being required to exit the Applicability desired to be entered to comply with the Required Actions.

Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of !he unit before or after the MODE change. Therefore, in such cases. entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before unit startup.

(continued)

BASES

LCO 3.0.4 (continued)

The provisions of LCO 3.0.4 shall not prevent changes in MORES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. The exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

LCO 3.0.4 Is only applicable when entering MODE 3 from MODE 4, MOBE 2 from MODE 3 or 4, or MODE 1 from MODE 2. Furthermore, LCO 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MORE 1, 2, or 3. The requirements of LCO 3.0.4 do not apply in MODES 4 and 5, or in other specified conditions of the Applicability (unless in MOBE 1, 2, or 3) because the ACTIONS of individual specifications sufficiently define the remedial measures to be taken. [In some cases (e.g., ..) these ACTIONS provide a Note that states "While this LCO is not met. entry into a MODE or other specified condition in the Applicability is not permitted, unless required to comply with ACTIONS." This Note is a requirement explicitly precluding entry into a MODE or other specified condition of the Applicability.]

(continued)

BASES

LCO 3.0.4 (continued)

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, changing MODES or other specified conditions while in an ACTIONS Condition, either in compliance with LCO 3.0.4 or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or SR 3.0.4 for those Surveillances that do not have to be performed due to the associated inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of required testing to demonstrate:

a. The OPERABILITY of the equipment being returned to service: or

b. The OPERABILITY *of* other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the required testing to demonstrate OPERABILITY. This Specification does not provide time to perform any other preventive or corrective maintenance.

(continued)

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

B 3.5.1 ECCS - Operating

BASES

BACKGROUND

The ECCS are designed, in conjunction with the primary and secondary containment, to limit the release of radioactive materials to the environment following a loss of coolant accident (LOCA). The ECCS uses two independent methods (flooding and spraying) to cool the core during a LOCA. The ECCS network consists of the High Pressure Coolant Injection (HPCI) System, the Core Spray (CS) System, the low pressure coolant injection (LPCI) mode of the Residual Heat Removal (RHR) System, and the Automatic Depressurization System **(ADS).** The suppression pool provides the required source of water for the ECCS. Although no credit is taken in the safety analyses for the condensate storage tank (CST), it is capable of providing a source of water for the HPCI, RHR and CS systems. The ECCS design requirements ensure that the criteria of Reference 12 are satisfied.

On receipt of an initiation signal, ECCS pumps automatically start; simultaneously, the system aligns and the pumps inject water, taken either from the CST or suppression pool, into the Reactor Coolant System (RCS) as RCS pressure is overcome by the discharge pressure of the ECCS pumps. Although the system is initiated, **ADS** action is delayed, allowing the operator to interrupt the timed sequence if the system is not needed. The HPCI pump discharge pressure almost immediately exceeds that of the RCS, and the pump injects coolant into the vessel to cool the core. If the break is small: the HPCI System will maintain coolant inventory as well as vessel level while the

(continued)

BASES

BACKGROUND (continued)

RCS is still pressurized. If HPCI fails, it is backed up by ADS in combination with LPCI and CS. In this event, either the vessel would be manually depressurized or the ABS timed sequence would be allowed *to* time out and open the selected safety/relief valves (S/RVs) depressurizing the RCS, thus allowing the LPCI and CS to overcome RCS pressure and inject coolant into the vessel. If the break is large, RCS pressure initially drops rapidly and the LPCI and CS cool the core.

Water from the break returns to the suppression pool where it is used again and again. Water in the suppression pool may be circulated through a heat exchanger cooled by the RHR Service Water System. Depending on the location and size of the break, portions of the ECCS may be ineffective; however, the overall design is effective in cooling the core regardless of the size or location of the piping break.

All ECCS subsystems are designed to ensure that no single active component failure will prevent automatic initiation and successful operation of the minimum required ECCS equipment.

The CS System (Ref. 1) is composed of two independent subsystems. Each subsystem consists of two 50% capacity motor driven pumps, a spray sparger above the core, and piping and valves *to* transfer water from the suppression pool to the sparger. The LOCA analysis (Ref. 13) requires both pumps in a subsystem (loop) to be OPERABLE for the subsystem to be OPERABLE. Failure of one CS pump results in the loss of the associated CS loop for LOCA mitigation. The CS System is designed to provide cooling to the reactor core when reactor pressure is low. Upon receipt of an initiation signal. the CS pumps in both subsystems are automatically started **(A** pump

(continued)

BASES

BACKGROUND (continued)

immediately when offsite power is available and B, C, and D pumps approximately 7, 14, and 21 seconds afterwards and if offsite power is not available all pumps 7 seconds after AC power is available). When the RPV pressure drops sufficiently, CS System flow to the RPV begins. A full flow test line Is provided to route water from and *to* the suppression pool to allow testing of the CS System without spraying water in the RPV.

LPCI is an independent operating mode of the RHR System. There are two LPCI subsystems (Ref. 2), each consisting of two motor driven pumps and piping and valves to transfer water from the suppression pool to the RPV via the corresponding recirculation Coop.

The two LPCI pumps and associated motor operated valve? in each LPCI subsystem are powered from separate 4 kV shutdown boards. Both pumps in a LPCI subsystem inject water into the reactor vessel through *a* common inboard injection valve and depend on the closure of the recirculation pump discharge valve following a LPCI injection signal. Therefore, each LPCI subsystem's common inboard injection valve and recirculation pump discharge valve are powered from one of the two 4 kV shutdown boards associated with that subsystem. The ability to provide power to the inboard injection valve and the recirculation pump discharge valve from two independent 4 kV shutdown boards ensures that a single failure of a diesel generator (DG) will not result in the failure of both LPCI pumps in one subsystem.

(continued)

BASES

BACKGROUND (continued)

The two LPCI subsystems can be interconnected via the LPCI cross tie valve; however, the cross tie valve is maintained closed with its power removed to prevent loss of both LPCI subsystems during a LOCA. The LPCI subsystems are designed to provide core cooling at low RPV pressure. Upon receipt of an initiation signal, all four LPCI pumps are automatically started (A pump immediately when offsite power is available, and B, C, and D pumps approximately 7, 14, and 21 seconds afterwards; if offsite power is not available, all pumps immediately when AC power is available). RHR System valves in the LPCI flow path are automatically positioned to ensure the proper flow path for water from the suppression pool to inject into the recirculation loops. When the RPV pressure drops sufficiently, the LPCI flow to the RPV, via the corresponding recirculation loop, begins. The water then enters the reactor through the jet pumps. Full flow test lines are provided for the four LPCI pumps to route water from the suppression pool. to allow testing of the LPCI pumps without injecting water into the RPV. These test lines also provide suppression pool cooling capability, as described in LCO 3.6.2.3, "RHR Suppression Pool Cooling."

The HPCI System (Ref. 3) consists of a steam driven turbine pump unit, piping, and valves to provide steam to the turbine. as well as piping and valves to transfer water from the suction source to the core via the feedwater system line, where the coolant is distributed within the RPV through the feedwater sparger. Suction piping for the system is provided from the CST and the suppression pool. Pump suction for HPCI is normally aligned to the CST source to minimize injection of suppression pool water into the RPV. However, if the CST

(continued)

BASES

BACKGROUND (continued)

water supply is low, or if the suppression pool level is high, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the HPCI System. With HPCI taking suction from the condensate storage tank and injecting to the reactor vessel, there is sufficient inventory in the tank such that the high suppression pool level suction transfer will occur before a low condensate header level would be created. The steam supply to the HPCI turbine is piped from a main steam line upstream of the associated inboard main steam isolation valve.

The HPCI System is designed to provide core cooling for a wide range of reactor pressures (150 psig to 1174 psig). Upon receipt of an initiation signal, the HPCI turbine stop valve and turbine control valve open and the turbine accelerates to a specified speed. As the HPCI flow increases, the turbine governor valve is automatically adjusted to maintain design flow. Exhaust steam from the HPCI turbine is discharged to the suppression pool. A full flow test line is provided to route water from and to the CST to allow testing of the HPCI System during normal operation without injecting water into the RPV.

The ECCS pumps are provided with minimum flow bypass lines, which discharge to the suppression pool. The valves in these lines automatically open (for CS and RHR they are already open) to prevent pump damage due to overheating when other discharge line valves are closed. To ensure rapid delivery of water to the RPV and to minimize water hammer effects, all ECCS pump discharge lines are filled with water. The LPCI and CS System discharge lines are kept full of water using the pressure suppression chamber head tank or condensate head tank. The HPCI System is normally aligned to the CST. The height of water in the CST is sufficient to maintain the piping full of water up to the first isolation valve. The relative height of the feedwater line connection for HPCI is such that the water in the feedwater lines keeps the remaining portion of the HPCI discharge line full of water.

(continued)

BASES

BACKGROUND (continued)

The **ADS** (Ref. **4)** consists of 6 of the 13 S/RVs. It is designed to provide depressurization of the RCS during a small break LQCA if HPCI fails or is unable to maintain required water level in the RPV. ADS operation reduces the RPV pressure to within the operating pressure range of the low pressure ECCS subsystems (CS and LPCI), so that these subsystems can provide coolant inventory makeup. Each of the S/RVs used for automatic depressurization is equipped with one air accumulator and associated inlet check valves. The accumulator provides the pneumatic power to actuate the valves.

APPLICABLE SAFETY ANALYSES

The ECCS performance is evaluated for the entire spectrum of break sizes for a postulated LOCA. The accidents fer which ECCS operation is required are presented in References *5* and 6. The required analyses and assumptions are defined in Reference 7. The results of these analyses are described in Reference 8.

This LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 9), will be met following a LOCA, assuming the worst case single active component failure in the ECCS:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

(continued)

BASES

AF .· .ICABLE SAFETY ANALYSES (continued)

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. The core is maintained in a coolable geometry; and

e. Adequate long term cooling capability is maintained.

The limiting single failures are discussed in Reference 13. For **a** large or small pipe break LOCA and events requiring ADS operation, selected battery failure is considered the *most* severe single failure. The remaining OPERABLE ECCS subsystems provide the capability to adequately cool the core and prevent excessive fuel damage.

The ECCS satisfy Criterion 3 of the NRC Policy Statement (Ref. 14).

LCO

Each ECCS injection/spray subsystem and six ADS valves are required *to* be OPERABLE. The **ECCS** injection/spray subsystems are defined as the two CS subsystems, the two LPCI subsystems, and one *HPCI* System. The low pressure ECCS injectionlspray subsystems are defined as the two CS subsystems and the two LPCI subsystems.

With less than the required number of ECCS subsystems OPERABLE, the potential exists that during a limiting design basis LOCA concurrent with the worst case single failure, the limits specified in Reference 9 could be exceeded. All ECCS subsystems and AD§ must therefore be OPERABLE *to* satisfy the single failure criterion required by Reference 9.

(continued)

BASES

LCO (continued)	LPCI subsystems may be considered OPERABLE during alignment and operation for decay heat removal when below the actual RHR low pressure permissive pressure in MOBE 3, if capable of being manually realigned (remote or local) to the LPCI mode and not otherwise inoperable. At these low pressures and decay heat levels, a reduced complement of ECCS subsystems should provide the required core coaling, thereby allowing operation of RHR shutdown cooling when necessary.
APPLICABILITY	All ECCS subsystems are required to be OPERABLE during MODES 1, 2, and 3, when there is considerable energy in the reactor core and core cooling would be required to prevent fuel damage in the event of *a* break in the primary system piping. In MODES 2 and 3, when reactor steam dome pressure is ≤ 150 psig, ABS and HPCI are not required to be OPERABLE because the low pressure ECCS subsystems can provide sufficient flow below this pressure. ECCS requirements for MODES **4** and *5* are specified in LCO 3.5.2, "ECCS - Shutdown."
ACTIONS	A.1 If any one low pressure ECCS injection/spray subsystem *is* inoperable, or if one LPCI pump in both LPCI subsystems is inoperable: the inoperable subsystem(s) must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE subsystems provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced, because a single failure in one of the remaining OPERABLE subsystems, concurrent with a LOCA, may result in the **ECCS** not being able to perform its intended safety function.

(continued)

BROWNS FERRY NUCLEAR PLANT
JOB PERFORMANCE MEASURE

JPM NUMBER: A1.2R

TITLE: DETERMINE REGULATORY REPORTING REQUIREMENTS

TASK NUMBER: S-000-AD-89

SUBMITTED BY:________________________________ DATE:_____

VALIDATED BY:________________________________ DATE:_____

APPROVED: ________________________________ DATE:_____
TRAINING

PLANT CONCURRENCE: ________________________________ DATE:_____
OPERATIONS

* Examination JPMs Require Operations Training Manager or Designee Approval and Plant Concurrence

BROWNS FERRY NUCLEAR PLANT
JOB PERFORMANCE MEASURE

OPERATOR:______________________________

RO ____ SRO _X__ DATE:_______
JPM NUMBER: A1.2R

TASK NUMBER: S-000-AD-89

TASK TITLE: DETERMINE REGULATORY REPORTING REQUIRMENTS

K/A NUMBER: 2.4.30 K/A RATING: RO_2.2_ SRO: 3.6__
****

TASK STANDARD: Using Tech Spec and available references, determine that condition require entry into TS 3.0.3 and determine per SPP-3.5, Regulatory Reporting Requirements, that a 4-hour report and follow-cp 60 days report are required for a Tech Spec required shutdown.

LOCATION OF PERFORMANCE: SIMULATOR x_ PLANT x CONTROL ROOM x_

REFERENCES/PROCEDURES NEEDED: *SPP-3.5* – Regulatory Reporting Requirements

VALIDATION TIME: CONTROLROQM: NA___ LOCAL: NA____
MAX. TIME ALLOWED: N/A___ (Completed for Time Critical JPMs only)

PERFORMANCE TIME: N/A____ CONTROL. ROOM N/A LOCAL N/A _
COMMENTS: ________

Additional comment sheets attached? YES __ NO

RESULTS: SATISFACTORY ___ UNSATISFACTORY

EXAMINER SIGNATURE:_______________ *DATE:*
EXAMINER

BROWNS FERRY NUCLEAR PLANT
JOB PERFORMANCE MEASURE

EXAMINER'S KEY
REFERENCES ALLOWED

INITIAL CONDITIONS

Unit 2 was operating at 100% rated thermal power when HPCI was declared inoperabie due to a failed area temperature surveillance requirement. Two hours later 2-PCV-1-19, an ADS valve, was declared inoperable due to a failed power supply.

You are the Shift Manager.

INITIATING CUE
Determine the required action by Technical Specifications as a result of the above conditions.

Examiners **CUE (this is not on the student** handout)
Once TS 3.0.3 ***entry* is *identified,*** then state "Per your direction, *the Unit Supervisoı has* commenced *a* shutdown per *TS* 3.0.3, Determine any Regulatory *Reporting Requirements, if any."*

EXPECTED RESPONSE
****

Critical Step **SAT_____** **UNSAT______**
Student should identify that the combination of HPCI and an ADS valve in Mode 1 is Tech Spec Condition 3.5.1.H which requires entry into TS LCO 3.0.3 Immediately.

Reference
Unit *2* Technical Specification

Provide CUE to determine reporting requirements (above)
****

Critical Step **SAT_____** **UNSAT______**
3 Hour Notification to NRC once the shutdown is initiated as required Cy TS

Reference SPP 3.5
****

(Continued on nex?page)

BROWNS FERRY NUCLEAR PLANT
JOB PERFORMANCE MEASURE

EXAMINER'S KEY
REFERENCES ALLOWED

****

NON Critical **Step** **SAT**_____ **UNSAT**______

60 Day LER is required upon completion of shutdown, which is defined *as* entry into the first shutdown mode as required by the TS condition and required action.

Examiners Note: The immediate notification requirement **(4** hour report) is a responsibility of Operations and the 60 day LER is Site Licensing's responsibility.

****

References

SPP-3.5, Regulatory Reporting Req. irements

Appensix A

3.1.C. The following criteria require 4-hour notification:

I. 50.72(b)(2)(i) -- The initiation of any nuclear plant shutdown required by the plant's Technical Specification

3.5.D Licensee Event Reports

A written report shall be prepared in accordance with 10 CFR 50.73(a)(i) for items in !he 60-day report criteria or Technical Specification. The report shall be ....

Report *Criteria*

1. 50.73(a)(2)(i)(A) – The **completion** of ary nuciear shutdown required by the plant's Technical Specifications.

CANDIDATE'S HANDOUT

REFERENCES ALLOWED

INITIAL CONDITIONS

Unit 2 was operating at 100% rated thermal power when HPCI was declared inoperable due to a failed area temperature surveillance requirement. Two hours later 2-PCV-1-19, an ADS valve, was declared inoperable due to a failed power supply.

You are the Shift Manager on duty.

INITIATING CUE

Determine the required action (if any) by Technical Specifications as a result of the above conditions.

Tennessee Valley Authority **N A N STANDARD PROGRAMS AND PROCESSES**	TITLE **REGU!' ATORY REPORTING REQUIREMENTS**	**SPP-3.5** **Rev. 11** **P[redacted]e Iof 53** Q[redacted]ty Related ☐ Yes ☒ No P[redacted]C Required ☐ Yes ☒ No 1[redacted]FR50 59 Review ☐ Yes ☒ No **Effective** Date **11/19/2002**

RESPONSIBLE PEER TEAM: Lic~~ens~~ing
Organization

CONCURRENCES

Mark J. Burzynski * *Primary Sponsor*	10/28/02 *Rate*
Ashok S. Bhatnagar *Peer Team Mentor*	11/13/02 *Date*

APPROVAL

For Nuclear Assurance Sponsored SPPs

N/A *General Manager, NA*	 *Cafe*
Karl W. Singer * *Senior Vice President, Nuclear Operations*	11/14/02 *Bate*

* Site-specific changes are approved by Site Sponsor and Site Vice President (see PCF)

REVISION LOG
Page 1 of 2

Revision Number	Effective Date	Pages Affected	Description of Revision
0	06-30-97	All	Initial issue. Replaces STD-4.5, SSP-4.5 **(SQN** and BFN) and SSP-**4.05** (WBN).
1	10-24-97	2, 17, 18	This revision adds: 1) a new condition to the regulatory reporting matrix contained in Appendix A and 2) the actual locations of the cooling lower, off-gas stack. and nieteorological tower obstruction lights to Appendix 8.
2	06-12-98	2, 4, 8, 9, 13-17, 20	Revised to modify definition of "Safe Shutdown" and make other minor editorial changes.
3	10-14-98	2, 11, 13, 17, 18, 28	Revised Appendix A, "Site Event Notification Matrix" to include additional notification requirements. Revised Appendix A, Section 3.1, "Immediate Notification - NRC" and Section 3.2, "Twenty-Four Hour Notification - NWC" to incorporate minor clarifying NRC change (replaced "eye" with "lens') per RIN 3150-AF46 [Ref: FR: 7/23/98 (Volume 63, Number 141)]. Revised Appendix B, "Other Regulatory Reportiny" to add telephone number for the FAA Also corrected typos on the "NRC Event Notification Worksheet."
4	72/21/98	2, 3, 28, 29, and 30	Added Appendix E. "Reporting of Decommissioning Funding' and corrected typos.
5	9/17/99	2, 12, 18, 28-30	Revised Appendix **A,** "Reporting of Event Related or Conditional Reports", to clarify reporting requirements for follow-up notifications. Revised Appendix 8, 'Other Regulatory Reporting", to revise telephone numbers for the FAA. Revised Appendix E, "Repotting of Decommissioning Funding", to address requirements for notifying NRC when shutting down the operation of a reactor, as required by 10 CFR 50.54(bb).
6	12/10/99	2. 10	Added note to clarify requirements for making an ENS notification for events or conditions that are discovered which met the emergency plan criteria **but** no emergency was declared and the basis for the emergency no longer exists.
7	1/23/01	2-7, 10-36	Procedure updated to reflect changes to 10 CFR 50.72 and 59.1'3 and a general update to the organization of the reporting guidance. Added Appendix B to provide guidance regarding reporting criteria for Events or Conditions Affecting Activities Involving By-product, Source or Special Nuclear Material Licenses. Added Appendix C which includes information previously included in Appendix **A** and further guidance regarding expectations for notification of Senior Management regarding plant events.

Defect - (a) A deviation in a basic component delivered for use in a facility, installed, used, or operated if, on the basis of an evaluation, the deviation could create a substantial safety hazard; or (b) the installation, use. or operation of a basic component containing a defect as defined above: (c) A deviation in a portion of a facility subject to the construction permit provided the deviation could, on the basis of an evaluation, create a substantial safety hazard and the portion of the facility containing the deviation has been offered to the purchaser for acceptance; or (d) A conditiori or circumstance involving a basic component that could contribute to the exceedirrg of a safety limit as defined by the plant operating technical specifications.

Department Level Manager - Any manager who functionally or administratively reports directly to the site vice president or plant manager.

Deviation - A departure from the technical or quality assurance requirements included in a procurement document. safety analysis report, construction permit or other documents provided for basic components.

Evaluation - The process of determining whether a particular deviation could create a substantial safety hazard or determining whether a failure to comply is associated with a substantial safety hazard.

Event - Any occurrence surrounding unit operation.

External Conditions - Events created by thicgs outside the desiyn features of the plant.

Government Agencies - An agency of the Federal government as defined in 10 CFR 50.2.

Incident Investigation - Process conducted by the NRC for the purpose of accident prevention. The process includes gathering and analyzing information, determining findings and conclusions, including the cause(s) of a significant operational event; and the disseminating of the investigation results for NRC, industry, and public review.

Initiation of Shutdown - Physical act of reducing power or temperature to change modes

Invalid Actuation (Signal) - Signals that do not meet the criteria for being valid. invalid actuations include instances where instrument drift, spurious signals, human error or other invalid signals that result in manual or automatic actuation of the systems listed in 10 CFR 50.73(a)(2)(iv)(B).

Major Loss of Communication - Constitutes the loss of communication capabilities

Major Deficiency - A condition or circumstance which under normal operating conditions, an anticipated transient. or postulated design basis accident could contribute to exceeding a safety limit or cause an accident. "Major deficiency" also means a condition or circumstance which in the event of an accident due to other causes could. considering an independent single failure, result in a loss of safety function necessary to mitigate the consequences of the accident.

Natural Phenomenon - Act of nature (e g., fire. flood, tornado)

Mews Release - Known items which may be distributed to the media (UPI, television, radio. newspaper, etc.) and those items identified to be going on TVA news tape distributed by the TVA Public Affairs Staff.

Noncompliance (Failure To Comply) - A noncompliance for the purposes of this procedure means any failure to comply with the Atomic Energy Act of 1954, as amended, or with any applicable rule or regulation of the NRC relating to substantial safety hazards. A noncompliance may be in operations, engineering, or construction of the facility or basic component thereof.

Organization Manager - This is the most senior manager available who is in the same organization as the individual who discovered the abnormal event. The senior manager is not normally interpreted to be the plant manager or site vice president.

Preplanned Sequence - Part of an approved procedure, including warkplans, work request, work orders, surveillance instructions, general operating instructions and system operating instructions.

Prevented The Fulfillment - Failure or possible failure of a safety system to properly complete a safety function.

Principal Safety Barrier - Fuel cladding, RCS pressure boundary. or the containment

Redundant Equipment - Equipment, systems, structures capable of performing the same intended function within the same Technical Specification allowable values. (In most cases, this means opposite train equipment.)

Safe Shutdown - Mode 3, as defined by the Technical Specifications

Safety Function - A component or structure designed to actuate upon receiving the proper signal (ESF or RPS).

Significant Operational Event - Any radiological, safeguards, or other safety-related operational event at an NRC licensed facility that poses an actual or potential hazard to the public health and safety, property, or the environment. These everits or those that typically result in a 10CFR50.72 immediate notification. (See Appendix A of this procedure) A significant operational event also may be referred to as "an incident". Examples of these events include:

operations that exceeded. or were not included in the design basis of the facility,
a major deficiency in design, construction, or operation having potential yeneric safety implications,
a significant loss of integrity of the fuel, the primary coolant boundary, or the primary containment boundary.
a loss of safety function or multiple failures in systems used to mitigate an actual event
significant unexpected system interactions,
repetitive failures or events involving safety related equipment or deficiencies in operation.
questions or concerns pertaining to licensee operational performance.

Substantial Safety Hazard - Loss of safety function to the extent that there is a major reduction in the degree of protection provided to public health and safety for any facility or activity licensed.

Threat - Physical hazard (e.g., fire. severe radioactive release)

Unanalyzed **Condition** - Plant Condition outside the bounds *of* the initial conditions as described in the FSAR accident analysis.

Valid Actuation (Signal) - Signals that are initiated in response to actual plant conditions or parameters satisfying the requirements for initiation,

APPENDIX a
Page 1 of 10
REPORTING OF EVENTS OR CONDITIONS AFFECTING LICENSED NUCLEAR POWEK PLANTS

1.0 PURPOSE

This Appendix identifies reporting requirements; and instructions for determining reportability, preparation, and transmittal of LERs; and notification to NRC for events occurring at TVA's licensed nuclear plants.

2.0 SCOPE

TVA is required by 10 CFR 50.72 and 50.74 to promptly report varicus types of conditions *or* events and provide written follow-up reports. as appropriate. This appendix provides reporting guidance applicable to licensed power reactors.

NOTE Appendix B provides additional reporting criteria found in 10 CFR Parts 20, 30, 40, and 70 that may be applicable to events involving byproduct, source or special nuclear material possessed by the licensed nuclear plant. Site Licensing and Site RadCon are responsible for making the repoctabilty determinations for 10 CFR Part 20, 30, 40. or 70 events associated with their site. Corporate Licensing and Corporate RadChem are responsible for making the reportability determinations for 10 CFR Part 20, 30, 40, or 70 events associated with ail other TVA licensed activities. Licensing is responsible for developing (with input from affected organizations) and submitting the immediate notification and wr..ten reports lo NRC in accordance with 10 CFR Part 20, 30, 40, or 70 requirements. Reporting requirements for personnel exposure required by 10 CFR Part 20 are contairied in RCDP-4, "Personnel Inprocessing and Dosimetry Administrative Processes."

NOTE Appendix C contains the criteria for reporting if events *or* conditicns affecting ISFSI. TVA. as the general licensee of ttie ISFSI. is required by 10 CFR 72.216 to make initial and written reports in accordance with 10 CFR 72.74 and 10 CFR 72.75. Operations is responsible for making the reportability determinations for 10 CFR 72.74 and 10 CFR 72.75 *reports.* Operations is responsible for making the immediate notification to NRC in accordance with 10 CFR 72.74. Cperations is responsible for makicg the immediate, 4-hour, and 24-hour notifications to NRC in accordance with 10 CFR 72.75. Licensing is responsible for developing (with input from affected organizations) and submitting the written reports required by 10 CFR 72.75.

NOTE Reporting requirements for events or conditions affecting the physical protection of the licensed nuclear plant specified in 10 CFR 73.71 are contained in SPP-1.3 "Plant Access and Security." Responsibilities for reportability determinations and immediate notification requirements are assigned to Site Nuclear Security and Corporate Nuclear Security. Licensing is responsible for developing (with input from affected organizations) and submitting the written reports required by 10 CFR Part 73.71

3.0 REQUIREMENTS

NOTE Internal management notification requirements for plant events are found in Appendix D Operations and the Plant Manager *(or* Duty Plant Manager) are responsible for making these internal rnanngement notifications.

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NOTE NRC NUREG-1022, Supplements arid subsequent revisions should be used as guidance for determining reportability of plant events pursuant to 10CFR50.72 and 10CFR50.73.

3.1 Immediate Notification - NRC

TVA is required by 10 CFR 50.72 to notify NRC immediately if certain types of everits occur. **This** appendix contains !he types of events and the allotted time in which NRC must be notified. (Refer to Form SPP-3.5-1). Operations is responsible for making the reportability determinations for 50.72 arid 50.73 reports. Operations is responsible for making the immediate notification to NRC in accordance with 10 CFR *50.72.*

Notification is via the Emergency Notification System. If the Emergency Notification System is not operative, use either a telephone, telegraph, mailgram, or facsimile.

NOTE The NRC Event Notification Worksheet may be used in preparing for notifying the NRC.

A. The immediate Notification Criteria of 10 CFR 50.72 is divided into I-hour, 4-hour, and 8- hour phone calls. Notify the NRC Operations Center within the applicable time limit for any item which is identified in the Immediate Notification Criteria.

B. The following criteria require 1-hour notification:

1. (Technical Specifications) - Safety Limits as defined by the Technical Specifications which have been violated.

2. 50.72 (a)(1)(i) - The declaration *of* any of the Emergency classes specified in the licensee's approved Emergency Plan.

MOTE If it is discovered that a condition existed which rnet the Emergency Plan criteria but no emergency was declared and the basis for the emergency class no longer exists at the time cf discovery. an ENS notification (and notification of the Operations Duty Specialist), within one hour of discovery of the undeclared (or misclassified) event. shall be made. However. actual declaration of the emergency class is not necessary in these circumstances.

3. 50.72(b).(1)) - Any deviation from !he plant's Technical Specifications authorized pursuant to 10 CFR 50.54(x).

C. The following criteria require 4-hour notification:

1. 50.72(b)(2)(i) - The initiation of any nuciear plant shutdown required by the plant's Technical Specifications.

2. 50.72(b)(2)(iv)(A) - Any event that results or should have resulted in Emergency Core Cooling System (ECCS) discharge into the reactor coolant system as a result of a valid signal except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

3. 50.72(b)(2)(iv)(B) - Any event or condition that results in actuation of the reactor protection system (RPS) when the reactor is critical except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

4. 50.72(b)(2)(xi) - Any event or situation, related to the health and safety of the public or onsite personnel, or protection of the environment, for which a news release is planned or notification to other government agencies has been or will be made. Such an event may include an onsite fatality or inadvertent release of radioactive contaminated materials.

D. The following criteria require 8-hour notification:

NOTE The non-emergency events specified below are only reportable if they occurred within three years of the date of discovery.

1. 50.72(b)(3)(ii)(A) - Any event or condition that results in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded.

2. 50.72(b)(3)(ii)(B) - Any event or condition that results in the nuclear power plant being in an unanalyzed condition that significantly degrades plant safety.

3. 50.72(b)(3)(iv)(A) - Any event or condition that results in valid actuation of any of the systems listed in paragraph (b)(3)(iv)(B) [see list below], except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

(1) Reactor protection system (RPS) including: Reactor scram and reactor trip.

NOTE Actuation of the RPS when the reactor is critical is also reportable under 50.72.b.2.iv.B above.

(2) General containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves (MSIVs).

(3) Emergency core cooling systems (ECCS) for pressurized water reactors (PWRs) including: High-head, intermediate-head, and low-head injection systems and the low pressure injection function of residual (decay) heat removal systems.

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(4) ECCS for boiling water reactors (BWRs) including: core spray systems; high-pressure coolant injection system; low pressure injection function of the residual heat removal system.

(5) BWR reactor core isolation cooling system

(6) PWR auxiliary or emergency feedwater system

(7) Containment heat removal and depressurization systems, including containment spray and fan cooler systems.

(8) Emergency ac electrical power systems, including: Emergency diesel generators (EDGs).

4. 50.72(b)(3)(v) - Any even! or condition that at the time of discovery could have prevented the fulfillment of the safety function of structures or systems that are needed to:

(A) Shut down the reactor and maintain it in a safe shutdown' condition;
(B) Remove residual heat:
(C) Control the release of radioactive material; or
(D) Mitigate the consequences of an accident.

NOTE According to 5G.72 (b)(3)(vi) events covered by 50.72(b)(3)(v) may include one or more procedural errors, equipment failures and/or discovery of design. analysis, fabrication, construction. and/or procedural inadequacies. However, individual component failures need not be reported pursuant this paragraph if redundant equipment in the same system was operable and available to perform the required safety function.

5. 50.72(b)(3)(xii) - Any event requiring the transport of a radioactively contaminated person to an offsite medical facility for treatment.

6. 50.72(b)(3)(xiii) - Any event that results in a major loss of emergency assessment capability, offsite response capability, or offsite communications capability (e.g., significant portion of control room indication, emergency notification system, or offsite notification system).

E. Follow-up Notification (50.72(c))

With respect to the telephone notifications made under paragraphs (a) and (b) [50.72 (a) and 50.72 **(b),** respectively] of this section [50.72], in addition to making the required initial notification, during the course of the event:

(1) Immediately report (i) any further degradation in the level of safety of the plant *or* other worsening plant conditions including those that require the declaration of the Emergency Classes, if such a declaration has not **been** previously made; or
(ii) any change from one Emergency Class to another, or
(iii) a termination of the Emergency Class.

(2) Immediately report (i) the results of ensuing evaluations or assessments of plant conditions,
(ii) the effectiveness of response or protective measures taken, and
(iii) information related to plant behavior that is not understood.

(3) Maintain an open, continuous communication channel with the NRC Qperetions Center upon request by the NRC.

3.2 Twenty-Four Hour Notification - NRC

Any violation of the requirement contained in specific operating license conditions, shall **be** reported to NRC in accordance with the license condition.

3.3 Two-Dav Notification - **NRC**

50.9(b) - The NRC shall be notified of *incomplete or* inaccurate infomation which contains significant implications fnr the public health and safety or common defense and security. Notification shill he provided to the administrator of the appropriate regional office within two working days of identifying the information. Licensing is responsible for determining reportability (with input from affected organizations) and notifying NRC in accordance with 10 CFR 50.9.

3.4 Sixty-Bay **Verbal Report**

50.73(a)(2)(iv)(A) requires that any event or condition that resulted in manual or automatic actuation of the specified systems be reported as a Licensee Event Report (LER (Refer to Appendix **A,** Section 3.51). This CFR section also allows that in the case of an invalid actuation, other than actuation of the reactor protection system when the reactor is critical, an optional teiephcne notification may **be** placed to the NRC Operations Center within 60 days after discovery of the event instead of submitting a written LER.

A. Verbal Report Required Content:

If the verbal notification option is selected (NUREG 1022, Revision 2, Section 3.2.6., "System Actuation"), instead of a LER, the verbal report:

1. Is not considered an LER

2. Should identify that the report is being made under 50.73(a)(2)(iv)(A)

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3. Should provide the following information:

   (a) The specific train(s) and system(s) that were actuated

   (b) Whether each train actuation was complete or partial

   (c) Whether or not the system started and functioned successfully

NOTE Licensing will ensure that the information that is provided to NRC during the Sixty-Day Verbal Report is verified in accordance with BP-213.

B. Verbal Report Development and Review

Licensing will:

1. Develop (with input from responsible organization) the response (i.e., report summary) to address the required input.

2. Ensure that the reporting Wails are reviewed by MRC

C. Telephone Report Timeliness

Licensing will make the 60-day telephone report promptly after the PER for the invalid actuation event is reviewed by MRC.

3.5 **Written Report - NRC**

A. A report on a Safety Limi?Violation shall be submitted to the NRC, the NSRB, and the Site Vice Fresident if required by Technical Specifications.

B. Any violation of the requirements contained in the Operating license conditions in lieu of other reporting requirements requires a written follow-up report if specified in the license.

C. Reporting Radiation Injuries

10 CFR 140.6(a) requires, as promptly as possible, submittal of a written notice [e.g., report] in the event of:

a. Bodily injury or property damage arising out of or in connection with the possession or **use** of the radioactive material at the licensee's faciiity [location]; or

b. In the course of transportation; or

c. In the event any radiation exposure claim is made. (Refer to RCDP-9 *Radiological and Chemistry Control Radiological Exposure Inquires")

The written notice shall contain particulars sufficient to identify the licensee and reasonably obtainable information with respect to time, place, and circamstances thereof, or the nature of the claim.

D. Licensee Event Reports

A written report shall be prepared in accordance with 10 GFR 50.73(a)(i) for items in the GO-day report criteria or Technical Specifications. The report shall be complete and accurate in accordance with the methods outlined in this procedure. The completed forms shall be submitted to the USNRC, Document Control Desk, Washington, DC **20555.** NUREG 1022, Revision *2,* contains the instructions for completion of the LER form. Licensing is responsible for *developing* (with input *from* affected organizations) and submitting the written reports (or optional telephone reports (refer to Appendix **A,** Section 3.41) required by 10 CFR 50.73.

NOTE Unless otherwise specified in ?hereporting criteria below, an event shall be reported if it occurred within three years of the date of discovery regardless of the plant mode or power level. and regardless ofthe significance of the structure, system, or component that initiated the event.

Report Griteria

1. 50.73(a)(2)(i)(A) - The completion of any nuclear plant shutdown required by the plant's Technical Specifications.

2. 50.73(a)(2)(i)(B) - Any operation or condition which was prohibited by the plant's Technical Specifications, except when:

   a. The Technical Specification is administrative in nature;

   b. The event consisted solely of a case of a late surveillance test where the oversight was corrected, the test was performed. and the equipment was found to be capable of performing its specified safety functions; or

   c. The Technical Specification was revised prior to discovery of the event such that the operation or condition was no longer prohibited at the time of discovery of the event.

3. 50.73(a)(2)(i)(C) -Any deviation from the plant's Technical Specifications authorized pursuant to 10 CFR 50.54(x).

4. 50.73(a)(2)(ii)(A) - Any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded.

5. 50.73(a)(2)(ii)(B) - Any event or condition that resulted in the nuclear power plant being in an unanalyzed condition that significantly degraded plant safety.

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6. 50.73(a)(2)(iii) - Any natural phenomenon or other external condition that posed an actual threat to the safety of the nuclear power plant or significantly hampered site personnel in the performance of duties necessary for the sate operation of the nuclear power plant.

7. 50.73(a)(2)(iv)(A) - Any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B) [see list in item no. *8* below], except when

(1) ***the*** actuation resulted from and was part *of* a pre-planned sequence during testing or reactor operation: or

(2) **The** actuation was invalid and (i) occurred while the system was properly removed froni service or (ii) occurred after the safety function had been already completed.

NOTE In the case of an invalid actuation, other than actuation of the reactor protection system (RPS) when the reactor is critical, a telephorie notification to the NRC Operations Center within 60 days after discovery *of* the event may be provided instead of submitting a written LER (13 CFR 50.73(a)). [Refer to Appendix A, Section 3.4]

8. 50.73(a)(2)(iv)(B) - The systems to which the requirements to paragraph (a)(2)(iv)(A) of this section apply are:

(1) Reactor protection system (RPS) inciuding: reactor scram or reactor trip.

(2) General containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves (MSIVs).

(3) Emergency core cooling systems **(ECCS)** for pressurized water reactors (PWRs) including: high-head intermediate-head, and low-head injection systems and the low pressure injection function of residual (decay) heat removal systems.

(4) ECCS for boiling water reactors (BWRs) including: core spray systems; high-pressure coolant injection system; low pressure injection function of the residua! heat removal system.

(5) BWR reactor core isolation cooling system.

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(6) PWR auxiliary or emergency feedwater system

(7) Containment heat removal and depressurization systems, including containment spray and fan cooler systems.

(8) Emergency ac electrical power systems, including: emergency diesel generators (EDGs).

(9) Emergency service water systems that do not normally run and that serve as ultimate heat sinks.

9. 50.73(a)(2)(v) - Any event or condition that could have prevented the fulfillment of the safely function of structures or systems that are needed to:

(A) Shut down the reactor and maintain it in a safe shutdown condition;

(B) Remove residual heat;

(C) Control the release of radioactive material; or

(D) Mitigate the consequences of an accident.

NOTE Events reported above may include one or more procedural errors, equipment failures, and/or discovery of design. analysis fabrication, construction, andior procedural inadequacies. However, individuai component failures need not he reported pursuant to this criterion if redundant equipment in the same system was operable and available to perform the required safety function [50.73(a)(2)(vi)]

10. 50.73(a)(2)(vii) - Any event where a single cause or condition caused at least one independent train or channel to become inoperable in multiple systems or two independent trains or channels to become inoperable in a single system designed to:

(A) Shut down the reactor and maintain it in a safe shutdown condition;

(B) Remove residual heat;

(C) Control the release of radioactive material; or

(D) Mitigate the consequences of an accident.

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11. 50.73(a)(2)(viii)(A) - Any airborne radioactivity release that, when averaged over a time period of 1 hour, resulted in airborne radionuclide concentrations in an unrestricted area that exceeded *20* times the applicable concentration limits specified in appendix B to part 20, table 2, column 1.

12. 50.73(a)(2)(viii)(B) - Any liquid effluent release that, when averaged over a time period of I hour, exceeds 20 times the applicable concentrations specified in appendix B to **part** 20, table 2, column 2, at the point of entry into the receiving waters (i.e., unrestricted area) for all radionuclides except tritium and dissolved noble gases.

13. 50.73(a)(2)(ix)(A) - Any event or condition that **as** a result of a single cause could have prevented the fulfillment of a safety function for two or more trains or channels in different systems that are needed to:

(1) Shut down the reactor and maintain it in a safe shutdowt? condition;

(2) Remove residual heat;

(3) Control the release of radioactive material; or

(4) Mitigate the consequences of **an** accident.

NOTE Events covered above may include cases of procedural error. equipment failure, and/or discovery of a design, analysis, fabrication, construction, andior prccedural inadequacy. However, licensees are not required to report an event pursuant to this criterion if the event results from a shared dependency among trains or channels that **is** a natural or expected consequence of the approved plant design or normal and expected wear or degradation [50.73(a)(2)(ix)(B)].

14 50.73(a)(2)(x) - Any event that posed **an** actual threat to the safety of the nuclear power plant or significantly hampered site personnel in the performance of duties necessary for the safe operation of the nuclear power plant including fires, toxic gas releases, or radioactive releases.

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'EPORTING OF EVENTS OR CONDITIONS AFFECTING ACTIVITIES INVOLVING BYPRODUCT, SOURCE OR SPECIAL NUCLEAR MATERIAL LICENSES

1.0 PURPOSE

This Appendix identifies reporting requirements; arid instructions for determining reportability, preparation, and transmittal of written reports. and notification to NRC for events affecting TVA activities governed by NRC byproduct. source or special nuclear materia! licenses.

2.0 SCOPE

TVA is required by its various NRC licenses to report certain events or conditions. 10 CFR Part 20 contains reporting requirements for events involving licensed byproduct, source, or special nuclear material. 10 CFF? 30.50 contains reporting requirements for events involving licensed byproduct material. 10 CFR 40.60 contains reporting requirements for events involving licensed source material. 10 CFR Part 70 contains reporting requirements for events and conditions involving licensed special nuclear material. This procedure contains the reporting requirements for these activities.

3.0 REQUIREMENTS

NOTE Internal manayement notification requirements for events reported to NRC are found in Appendix D. Operations and the Plant Manager (or Duty Plant Manager) are responsible for making these i..ternal management notifications.

3.1 Immediate Notification - NRC

TVA is required by the various byproduc:, source or special nuciear materia! licenses to notify NRC immediately if certain types of events or conditions occur. This appendix contains the types of events and the allotted time in which NRC must be notified.

Site Licensing and Site RadCon are responsible for making the reportability determinations for 10 CFR Part 20, 30, 40, or 70 events associated with their site. Corporate Licensing and Corporate RadChem are responsible for making the reportability determinations for 10 CFR Part 20, 30, 40, or 70 events associated with all other TVA licensed activities. Licensing is responsible for making the immediate notification and developing (with input from affected organizations) and submitting written reports *to* NRC in accordance with 10 CFR Part 20, 30, 40, or 70 requirements.

NOTE Reporting requirements for personnel exposure required by 10 CFR Part 20 are contained in RCDP-4, "Personnel Inprocessing and Dosimetry Administrative Processes."

Notification should be **made** to the NRC office identified in the specific reporting regulation.

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A. The following criteria require immediate notification:

1. 10CFR20.1906(d)(1) - Upon discovery, any removable radioactive surface contamination that exceeds the limits of 10CFR71.87(1).

2. 10CFR20.1906(d)(2) - Upon discovery, any external radiation levels that exceed the limits of 10CFR71.47.

3. 20.2201(a)(1)(i) - Upon discovery, any lost stolen, or missing licensed material has occurred in an aggregate quantity equal to or greater than 1,000 times the quantity specified in 10 CFR Part 20, Appendix C under circumstances that appears that exposure could result lo persons in unrestricted areas.

4. 20.2202(a)(1) - Any event involving byproduct, source, *or* special nuclear material possessed **by** the licensee that may have caused or threatens lo cause an individual to receive a total effective dose equivalent *of* 25 rems or more; a lens dose equivalent of 75 rems or more; or a shallow dose equivalent to the skin or extremities of 250 rads or more.

5. 20.2202(a)(2) - Any event involving byproduct, source, or special nuclear material possessed by the licensee that may have caused or threatens lo cause the release of radioactive material, inside or outside of a restricted area, so that. had an individual been present for 24 hours, the individual could have received an intake five times the occupational annual limit on intake.

B. The following criteria require I-hour notification:

1. 70.52(a) -Any case of accidental criticality and any loss, other than normal operating loss, of special nuclear material.

2. 70.52(b) -Any loss or theft or unlawful diversion of special nuclear material which the licensee is licensed to possess or any incident in which an attempt has been made or is believed to have been made to commit a theft or unlawful diversion *of* such material.

C. The following criteria require 4-hour notification:

1. 30.50(a) and 40.60(a) and 70.50(a) - Upon discovery, any event involving licensed byproduct, source or special nuclear material that prevents immediate protective actions necessary to avoid exposures to radiation or radioactive material that could exceed regulatory limits or releases of material that could exceed regulatory limits.

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D. The following criteria require 24-hour notification:

1. 20.2202(b)(1) - Upon discovery of the event, report any event involving loss of control of licensed material possessed by the licensee that may have caused, or threatens to cause, an individual to receive, in a period of 24 hours a total effective dose equivalent exceeding 5 rems, a lens dose equivalent exceeding 15 rems, or a shallow-dose equivalent to the skin or extremities exceeding 50 rems (0.5 Sv).

2. 20.2202(b)(2) - Upon discovery of the event, repott any event involving loss of control of licensed material possessed by the licensee that may have caused, or threatens to cause the release of radioactive material, inside or outside of a restricted area, so that. had an individual been present for 24 hours, the individual could have received an intake in excess of one occupational annual limit on intske (the provisions of this paragraph do not apply to locations where personnel are not normaily stationed during routine operations, such as hot-cells or process enclosures).

3. 30.50(b) and 40.60(b) and 70.50(b) - Upon discovery of any of the following events involving licensed material:

   a. An unplanned contamination event that requires access to the contaminated area. by workers or the public, *to* be restricted for more than 24 hours by imposing additional radiological controls or by prohibiting entry into the area <u>and</u> involves a quantity of material greater than five times the lowest annual limit on intake specified in Appendix B of §§20.1001 - 20.2401 of 10 CFR part 20 for the materiai <u>and</u> has access to the area restricted for a reason other than to allow isotopes with a half-life of less than 24 hours to decay prior to decontaminalion.

   b. An event in which equipment is disabled or fails to function as designed when the equipment is required by regulation or license condition to prevent releases exceeding regulatory limits, to prevent exposures to radiation and radioactive materials exceeding regulatory limits, or to mitigate the consequences of an accident <u>and</u> the equipment is required to be available and operable when it is disabled or fails to function <u>and</u> no redandant equipment is available and operable to perform the reqtiired safety function.

   c. An event that requires unplanned medical treatment at a medical facility of an individual with spreadable radioactive contamination or?the individual's clothing or body.

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d. An unplanned fire or explosion damaging any licensed material or any device. container, cr equipment containing licensed material when the quantity of material involved is greater than five times the lowest annual limit on intake specified in Appendix B of §§20.1001 - 20.2401 of 10 CFR Part 20 for the material and the damage affects the integrity of the licensed material or its container.

3.2 Verbal Report - NRC

A. The following criteria required 30-day notification:

1. 20.2201(a)(ii) - Within 30 days after the occurrence of any lost, Stolen. or missing licensed material becomes known, verbally report all licensed material in a quantity greater than 10 times the quantity specified in Appendix C to Part 20 that is still missing at this time.

3.3 Written Report - NRC

A. 2-Week Report

10CFR20, Appendix G(III)(E) - A written report is required within 2 weeks of cornpleticn of the investigation of any shipment or part of a shipment for which acknowledgment is not received within the times set forth in 10CFR20 Appendix G. This investigaticn must:

1. Be petformed by the shipper. if the shipper has not received notification of receipt within 23 days after transfer,

2. Be traced and reported. The investigation shail include this shipment and filing a report with the nearest Commission Regional Office listed in 10CFR20 Appendix D.

B. 30-Day Report

A written report is required within 30 days for the following items:

1. 20.2201(b) - Events reported in accordance with 10 CFR 20.2201(a).

2. 20.2203(a)(1) - Events reported in accordance with 10 CFR 20.2202.

3. 20.2203(a)(2) - Doses in excess of any of the following the occupational dose limits for adults in §20.1201, or the occupational dose limits for a minor in §20 1207. or the limits for an embryo/fetus of a deciared pregnant woman in §20.1208 or the limits for an individual member of the public in §20.1301. or any applicable limit in the license or the ALARA constraints for air emissions established under §20.1101(d)

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4. 20.2203(a)(3) - Levels of radiation or concentrations of radioactive material in *a* restricted area in excess of any applicable limit in the license or an unrestricted area in excess of 10 times any applicable limit set forth in this part or in the license (whether or not involving exposure of any individual in excess of the limits in §20.1301)

5. 20.2203(a)(4) - For licensees subject to the provisions of EPA's generally applicable environmental radiation standards in 40 CFR part 190, levels of radiation or releases of radioactive material in excess of those standards. or of license conditions related to those standards.

6. 20.2204 - Any planned special exposure conducted in accordance with §20.1206, informing the Commission that a planned special exposure was conducted and indicating the date the planned special exposure occurred and the information required by §20.2105.

7. 30.50(c)(2) - Events reported in accordance with 10 CFR 30.50(a) and 30.50(b).

8. 40.60(c)(2) - Events reported in accordance with 10 CFR 40.60.a and 40.60(b).

9. 70.50(c)(2) - Events reported in accordance with 10 CFR 70.50.a and 70.50)b).

The written report will contain the information specified by the appropriate regulation. The report shall be complete and accurate in accordance with the methods outlined in this procedure. The completed forms shall be submitted to the address identified in the specific regulation.

C. 90-Day Report

A written report is required within 90 days for the following items:

1. 30.34(b) and 40.46 and 70.36 - Notification of intent to transfer ownership or control of licensed activities shalt be made 90 days prior to the proposed action. (Reference: NRC information Notice 89-25, Revision 1).

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REPORTING OF EVENTS OR CONDITIONS AFFECTING INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI)

1.0 PURPOSE

This Appendix identifies NRC reportirig requirements; and provides instructions for determining reportability, preparation, and transmittal of written reports; and notification to NRC for events occurring at spent fuel storage installations at TVA's licensed nuclear plants.

2.0 SCOPE

TVA, as the general licensee *of* the ISFSI is required by 10 CFR 72.216 to make initial and written reports in accordance with 10 CFR 72.74 and 10 CFR 72.75. 10 CFR 72.74 and 10 CFR 72.75 requires the general licensee to promptly report various types of conditions or events and provide written follow-up reports, as appropriate. This appendix provides reporting guidance applicable to spent fuel storage installations at licensed power reactors.

3.0 REQUIREMENTS

This Section contains the types of events and the allotted time in which NRC must be notified. Operations is responsible for making the reportability deterrninations for 10 CFR 72.74 and 10 CFR 72.75 reports. Operations is responsible for rnaking the immediate notification to NRC in accordance with 10 CFR 72.74 and 10 CFK 72.75, and is responsible for making the four-hour and twenty-four hour 10 CF7 72.75 reports.

Notification is via the Emergency Notification System. If *the* Emergency Notification System is inoperative, make the required notification via commercial telephonic service or any other dedicated telephonic system or any other method that will ensure that a report is received by the NRC Operations Center within the required timeframe.

3.1 Immediate Notification - NRC

TVA is required **by** 10 CFR 72.74 and 72 75 to notify NRC imniediately if certain types of events occur.

A. The Immediate Notification Criteria of 70 CFR 72.74(a) require the licensee to notify the NRC Operations Center within 1-hour. The following criteria require 1-hour notification:

1. Discovery of accidental criticality. or

2. Any loss of special nuclear material

B. The Immediate Notificatior! Criteria of 10 CFR 72.75(a) require the licensee to notify the NRC Operations Center within 1-hour. The following criteria require 1-hour notification:

1. Emergency Notifications - The declaration of an emergency as specified in TVA's approved emergency plan addressed in 10 CFR 72.32.

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3.2 Four Hour Non-Emergency Notification - NRC

TVA is required by 10 CFR 72.75(b) to notify NRC as soon as possible, but no Cater than four hours after the discovery of any of the following events involving spent fuel, high level radioactive waste, or reactor-related Greater than Class C waste:

A. 10 CFR 72.75(b)(1) - An event that prevents immediate actions necessary to avoid exposures to radiation or radioactive materials that could exceed regulatory limits or releases of radioactive materials that could exceed regulatory limits (e.g., events such as fires, explosions, and toxic gas releases).

B. 10 CFR 72.75(b)(2) - A defect in any storage structure, system, or component which is important to safety.

C. 40 CFR 72.75(b)(3) - A significant reduction in the effectiveness of any storage confinenient system during use.

D. 10 CFR 72.75(b)(4) - An action taken in an emergency that departs from a condition or a technical specification contained in a license or certificate of compliance issued under this part when the action is immediately needed to protect the public health and safety and no action consistent with license or certificate of compliance conditions or technical specifications that can provide adequate or equivalent protection is immediately apparent.

E. 10 CFR 72.75(b)(5) - An event that requires unplanned medical treatment at an offsite medical facility of an individual with radioactive contamination on the individual's clothing or body which could cause further radioactive contamination.

F. 10 CFR 72.75(b)(6) - An unplanned fire or explosion damaging any spent fuel, high level radioactive waste, andior reactor-related Greater than Class C waste or any device, container, or equipment containing spent fuel, high level radioactive waste, andicr reactor-related Greater than Class C waste when the damage affects the integrity of the material or its container.

3.3 Twentv Four Hour Non-Emergency Notification - NRC

TVA is required by 10 CFR 72.75(c) to notify NRC within twenty four hours after the discovery of any of the following events involving spent fuel, high level radioactive waste, or reactor-related Greater than Class C waste:

A. 10 CFR 72.75(c)(1) - Any unplanned contamination event that requires access to the contaminated area by workers or the public to be restricted for more than 24 hours by imposing additional radiological controls or by prohibiting entry intc the area.

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B. 10 CFR 72.75(c)(2) - An event in which safety equipment is disabled or Fails to function as designed when:

1. 10 CFR 72.75(c)(2)(i) - The equipment is required by regulation, license condition, or certificate of compliance to be available and operable to prevent releases that could exceed regulatory limits, to prevent exposures to radiation or radioactive materials that could exceed regulatory limits, or to mitigate the consequences of an accident; and

2. 10 CFR 72.75(c)(2)(ii) - No redundant equipment was available and operable to perform the required safety function.

3.4 Information Required During Initial Notification - NRC

10 CFR 72.75(d)(1) requires that the nctifications made under 10 CFR 72.75(a), (b), or (c) [Section 3.1.B, 3.2, or 3.3 of this Appendix] by telephone to the NRC Operations Center provide. to the extent that the information is available at the time of notification, the following information:

a. The caller's name and call hack telephone number;

B. A description of the event, including date and time;

C. The exact location of the event;

D. The quantities, and chemical and physical forms of the spent fuel, high level radioactive waste, or reactor-related greater than Class C waste involved; and

E. Any personnel radiation exposure data.

3.5 **Thirty Dav** Written **Report - NRC**

10 CFR 72.75(d) requires submittal of a written report to NRC within thirty days following initial notification required by 10 CFR 72.75(a), **(b),** or (c) [Section 3.1.B, 3.2, or 3.3 of this Appendix]. These written reports must be sent to the NRC, in accordance with 10 CFR 72.4, "Communications."

NOTE Written reports prepared pursuant to other regulations may **be** submitted to fulfill this requirement if the reports contain all the necessary inforination and the appropriate distribution is made.

The written report must include:

A. A brief abstract describing the major occarrences during the event, including all component or system failures that contributed to the event and significant corrective action taken or planned to prevent recurrence;

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B. A clear, specific, narrative description of the event that occurred so that knowledgeable readers conversant with the design of ISFSI or Monitored Retrievable Storage Installations (MRS), but not familiar with the details of a particular facility, can understand the complete event. The narrative description must include the following specific information as appropriate for the particular event:

1. ISFSI and MRS operating conditions before the event;

2. Status of structures, components, or systems that were inoperable at the start of the event and that contributed to the event:

3. Dates arid approximate time of occurrences;

4. The cause of each component or system failure or personnel error, if known;

5. The failure mode, mechanism, and effect of each failed component, if known;

6. A list of systems or secondary functions that were also affected for failures of components with multiple functions;

7. For wet spent fuel storage systems only, after failure that rendered a train of a safety system inoperable, an estimate of the elapsed time from the discovery of the failure until the train was returned to service;

8. The method of discovery of each component of system failure or procedural error;

9. Operator actions thst affected the course of the event, including operator errors, procedural deficiencies, or both, that contributed to the event;

10. For each personnel error !he licensee shall discuss:

    a) Whether the error was a cognitive error (e.g., failure to recognize the actual facility condition, failure to realize which systems should be functioning, failure to recognize the true nature of the event) or a procedural error:

    b) Whether the error ivas contrary to an approved procedure, was a direct result of an error in an approved procedure, or was associated with an activity or task that was not covered by an approved procedure;

    c) Any unusual characteristics of the work location (e.g., heat, noise) that directly contributed to the error; ard

d) The type of personnel involved (e.g., contractor personnel, utility-licensed operator, utility nonlicensed operator, other utility personnel);

11. Automatically and manually initiated safety system responses (wet spent fuel storage system only);

12. The manufacturer and model number (or other identification) of each component that failed during the event:

13. The quantities, and chemical arid physical forms of the spent fuel, high level radioactive waste, or reactor-related greater than Class C waste;

C. An assessment of the safety consequences and implications of the event. This assessment must include the availability of other systems or components that could have performed the same function as the components and systems that failed during the event;

D. A description of any corrective actions planned as a result of the event, including those to reduce the probability of similar events occurring in the future;

E. Reference to any previous similar events at the same facility that are known to the licensee;

F. The name and telephone number of a person within the licensee's organization who is knowledgeable about the event and can provide additional information concerning the event and the facility's characteristics;

G. The extent of exposure of individuals to radiation or to radioactive materials without identification of individuals by name.

4.0 Records

4.1 Retention

10 CFR 72.80(c) - Records that are required by 10 CFR 72 or by the license conditions must be maintained for the period specified by the appropriate regulation or license condition. If a retention period is not otherwise specified the above records must be maintained until NRC terminates the license.

4.2 License Termination and Transfer

A. 10 CFR 72.80(e) - Prior to license termination, records required by 10 CFR 20.2103(b)(4) and 10 CFR 72.30(d) shall be forwarded to the NRC Region II office.

B. 10 CFR 72.80(f) - if licensed activities are transferred or assigned in accordance with 10 CFR 72.44(b)(1), records required by 10 CFR 20.2103(b)(4) and 10 CFR 72.30(d) shall be forwarded to the new licensee arid the new licensee will be responsible for maintaining these records until the license is terminated.

Event/Condition	Notification Requirements			
	Plant Manager	SVP, Nuclear Operations	Duty Plant Manager	Ops. Duty Spec. (ODS)
Reactor/Turbogenerator trip, unscheduled unit power reduction, or conscheduled unit shutdown; and when unit is restored to full service.	Yes*	Yes	Yes'	Yes
Unplanned entry into a Limiting Condition for Operation with time duration of *72* hours or **less.**	Yes*	Only for duration of 24 hours or less	Yes"	Only when TS safety limits exceeded
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	Yes*	[illegible]	[illegible]	[illegible]
[illegible] due to medi[illegible]	Yes*	Yes	Yes*	Yes
Release [illegible] discharge canal, permit.	No	No		Yes
[illegible] public. (1)	Yes*	Yes	Yes*	Yes
[illegible]	Yes*	Yes	Yes*	Yes
[illegible]	Yes*	No	Yes*	Yes
[illegible] release.	Yes*	No	Yes*	Yes
[illegible]	Yes	Yes	Yes	No
[illegible]	Yes	Yes	Yes	No
[illegible] change.	Yes	Yes	Yes	No

APPENDIX E
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OTHER REGULATORY REPORTING

1.0 PURPOSE

This Appendix identifies other reports required by various regulatory agencies other than !he NRC.

2.0 NOTIFICATION CRITERIA/REPORT INSTRUCTION

2.1 Immediate Notification - Federal Aviation Administration (FAA)

The Plant Operations Department is responsible for verifying the proper cperaticn of the cooling tower, off-gas stack and meteorological tower obstruction lights. This should be accomplished through visual observation at least once each 24 hours. Any observed or otherwise known extinguishment or improper functioning of any top obstruction light which will last more than 30 minutes should be immediately reported to the **FAA** at (800) 352-6751 (for SQN and WEN) and (800) 772-0547 (for BFN and BLN) **by** the shift manager (or designee). Information !he FAA will require ***is*** latitude and longitude for the tower, height of the tower (to mean sea level), facility, name of person making the notification. the condition of the light. or fights. the circumstance which caused the failure (if known), and the probable date normal operation will resume. This information is detailed in Notice to Airmen 7AA 79-30.2f. Further notification should be given upon resumption of normal operation of the obstruction lights.

Plant	Tower	Latitude	Longitude	Height (mean sea level)
BLN	Cooling Tower 1	34° 42' 27.46"N	85° 55' 51.39"W	459 ft. (1109)
	Cooling Tower 2	34° 42' 22.80"N	85° 55' 44.40"W	45 ft. (1109)
	MET Tower	34" 43' 08.33"N	85" 54' 56.99"W	300 ft. (1056)
BFN	Off-gas Stack	34° 42' 16.0"N	87" 07' 15.0"W	600 ft. (1165)
	MET Tower	34" 42' 03.19"N	87° 06' 29.28"W	300 ft. (865)
SQN	Cooling Tower 1	35° 13' 21.72"N	85° 05' 22.22"W	459 ft. (1159)
	Cooling Tower 2	35° 13' 15.07"N	85° 05' 19.85"W	459 ft. (1159)
	MET Tower	35" 13' 10.50"N	85" 06' 04.30"W	300 ft. (1056)
WBN	Cooling Tower 1	35° 36' 05.95"N	84° 47' 09.18"W	509 ft. (1240)
	Cooling Tower 2	35° 36' 11.88"N	84" 47' 10.37"W	509 ft. (1240)
	MET Power	35" 36' 10"N	84" 47' 24.24"W	300 ft. (1011)

NOTE NRC notification is not required when the FAA is notified of cooling tower. meteorological tower, or Browns Ferry off-gas stack lighting deficiencies.

2.2 Immediate Notification - Tennessee Ernerqencv Management Agency (TEMA)/Alabama Emerqency Manaqement Aqency (AEMA)

The following is a clarification cf what constitutes non-emergency events that require the Operatiorrs Duty Specialist lo immediately notify TEMA or AEMA.

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A. **A** confirmed fatality, whether or not it is related to nuclear operations.

Confirmation to he provided by TVA Medical Services when the death is the immediate result of an accident or iilness occurring on site.

B. Strikes or honoring of picket iines affecting plant operations.

C. Accidental activation of the Prompt Notification Systern (PNS) sirens

This will include confirmed inadvertent activation of the offsite PNS siren system or portions of the system. The shift manager rnay not he cognizant of ail inadvertent activations; however, if the shift manager becomes aware of such an incident, the Operations Duty Specialist (ODS) shall **be** called. If the ODS becomes aware of such an incident, from sources other than the shift manager, then the ODS shall notify the shift manager.

D. Undeclared emergency events reported to NRC

If an event or condition is discovered which meets the emergency class criteria but no emergency was previously deciared and the basis for the emergency class no longer exists; a report to NRC shall be made. The State personnel should be notified that the condition was reported to NRC and could result in potential media coverage.

E. Incidents which could attract attention from the immediate local residents.

Examples include: explosions, fires, release of steam or liquid onsite accountability siren soundings, and TVA or local emergency vehicles (fire, rescue, medical, law enforcement), with sirens sounding while entering or leaving the owner-controlled area (OCA) in response to a TVA situation. The above examples do not require notification if the State has been notified in advance. The testing or use of TVA vehicle sirens (fire, ambulance. security) within the OCA only does not require notification.

NOTE Notify the NRC Resident when notifications are to he made to TEMA or AEMA. Notification of the NRC Operations Center is not required due to notification of TEMA or AEMA, unless required by other applicabie reporting requirements.

APPENDIX F
***Page* 1 of 2**
EVALUATION AND REPORTING OF DEFECTS AND FAILURES TO COMPLY ASSOCIATED WITH SUBSTANTIAL SAFETY HAZARDS PER 10 CFR 50.55(e) REPORTING REQUIREMENTS

1.0 PURPOSE

The purpose of this Appendix is to identify the requirements for the evaluation and reporting of defects and failures to comply associated with substantial safety hazards to the NRC in accordance with 10 CFR 50.55(e) and defines the interface with 10 CFR 21 reporting procedures where appropriate.

2.5 SCOPE

Reportability for defects and failures to comply associated with substantial safety hazards under 10 CFR 50.55(e) is applicable to PVA nuclear plants with construction permits and requires TVA to notify the NRC of these deficiencies. These deficiencies are initially identified by the Administrative Control Programs defined in "Corrective Action Program", SPP-3.1.

Reportability of deficiencies which contain Safeguards information Pursuant to 10 CFR 73.71 will he processed in accordance with SPP-1.4, "Safeguards Information."

3.0 REQUIREMENTS

3.1 General

A. 10 CFR 50.55(e) requires that holders of construction permits evaluate deviations and failures to comply associated with substantial safety hazards as soon as practicable and in all cases within 60 days of discovery, except as provided in Subsection 3.1.D., in order to identify the reportable defect or fai'ure to comply that could create a substantial safety hazard were it to remain uncorrected

B. If the deficiency is related to any deficiency which has already been reported to NRC and the description of the deficiency and corrective actions for the new deficiency are within the scope of the previously reported deficiency. a memorandum should be prepared documenting that no further reporting is required.

C. If the deficiency is related to any deficiency which has already been reported to NRC and the description of the deficiency and corrective actions for the deficiency are not within the scope of the previously reported deficiency.

1. Initiate actions to develop a submittal to NRC expanding the Scope of the previously reported deficiency, OR

2. Initiate actions to deveiop an initial report to NRC

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D. If an evaluation of an identified deviation or failure *to* comply potentially . associated with a substantial safety hazard cannot be completed within 60 days of discovery, then an interim report must be submitted to NRC within 60 days of discovery of the deviation or failure to comply.

3.2 Determining Reportability of Deficiencies

A. Line managenient determines whether the deficiency is potentially reportable or not repoitable under 10 CFR 50.55(e) in accordance with the timeframe as specified in Section 3.2 above, in accordance with the corrective action program requirements specified in SPP-3.1.

B. If the deficiency is determined to be potentially reportable.

1. Notify the Site Licensing Manager arid the Site Vice President, within five working days of the completion of the evaluation, that a report is required. Site Licensing will make the final determination.

2. Initial NRC notification preferably by facsimile, to the NRC Operations Center. [telephone numbers specified in 10 CFR 50.55(e)(6)(i)] must then be made within two caiendar days following receipt of information by the Site Vice President or Site Licensing Manager. Verification that the facsimile has been received should be made by calling the NRC Operations Center.

a. Within 30 days following notification of the Site Vice President or Site Licensing Manager of a substantial safety hazard. a written report is required to be submitted to the NRC.

b. If deviations are evaiuated under 50.55(e) and result in either a negative reportability determination or reportable defect; then this satisfies the requirements of Part 21

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DETERMINATION OF REPORTABILITY UNDER 10 CFR PART 21

1.0 PURPOSE

The purpose of this Appendix is to specify instructions for reporting requirements and for the evaluation of potential defects and noncompliances pursuant to **10** CFR Part **21.**

2.0 SCOPE

Part **21** requires managers and responsible officers of certain firms and organizations which are building, operating. or owning NRC-licensed facilities or conducting NRC-licensed activities to (1) report any defects in basic components; **(2)** report any failures to comply with NRC requirements that could result in a substantial safety hazard; (3j post 10 CFR **21** regulations, Section **206** of the Energy Reorganization Act of 1974, and procedures adopted pursuant to Part **21** regulations; **(4)** specify in procurement documents Part **21** applicability: and **(5)** maintain evaluations of all deviations and failures to comply for a minimum of five years after the date of the evaluation. Posting will be in accordance with NADP-9.

Reportability of deficiencies which contain Safeguards Information pursuant lo 10 CFR 73.71 will **be** processed in accordance with SPP-1.4.

This Appendix establishes the methods for evaluating defects and failures to comply to determine if they are reportable in accordance with 10 CFR 21. This Appendix also implements the requirements in the regulation for timing and content of reports.

A flowchart illustrating the process for evaluating potential defects and failures to comply is shown in Figure 1 of this Appendix.

3.0 REQUIREMENTS

3.1 10 CFR 21 Evaluation and Reporting

If a deviation or failure to comply as evaluated could create a Substantial Safety Hazard. it must be reported to the NRC unless the Site Vice President has actual knowledge that it has been previously reported to the NRC in writing.

A. Reporting Criteria

Operating plants with potential defects in installed equipment must be evaluated under 10 CFR 50.72, 50.73, or 73.71 as appropriate rather than 10 CFR 21 Only those potential defects in basic components which have never been installed or used in the plant are required to be evaluated under 10 CFR 21 Plants with construction permits are required to report under 10 CFR 50.55(e) rather than under 10 CFR **21**

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B. 10 CFR 21 Evaluation Criteria

bine management will perform an evaluation of any defect or failure associated with substantial safety hazards in accordance with the corrective action program requirements specified in SPP-3.1. If the deficiency is determined to **be** potentially reportable under 10 CFR 21, notify the Site Licensing Manager within five working days of the completion of the evaluation.

3.2 Written **Report** Content

Each written report submitted to the NRC shall contain the following:

- The name and address ofthe individual or individuals informing the NRC.

  Identification of the facility, !he activity, or the basic component supplied for such facility or such activity within the United States which fails to comply or contains a defect.

- Identification of the firm constructing the facility or supplying the basic component which fails *to* comply or contains a defect.

- Nature of the defect or failure to comply and the safety hazard which is created or could be created by such defect or failure to comply.

- The date on which the information of such defect or failure to comply was obtained.

  In the case of a basic component which contains a defect or fails to comply. the number and location of all such components in use at, supplied for, or being supplied for one or more facilities or activities subject to the regulations in this part.

- The corrective action which has been, is being. or will be taken; the name of the individual or organization responsible for the action; and the length of time that has been or will be taken to complete the action.

- Any advice related to the defect or failure to comply about the facility, activity, or basic component that has been, is being, or will be given to purchasers or licensees.

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3.3 Identification of Deviations and Noncompliances

The principal means for identifying deviations and noncompliances in TVAN are: (a) the Problem Evaluation Report (PER) and the Administrative Control Programs (ACP), (b) the Receiving Inspection Report, (c) notices received from vendors, (d) Preoperational, Post-Modification, Surveillance and Post-Maintenance testing, (e) Licensee Event Reports of other sites (LERs), and (f) 10 CFR 50.55(e) reports at plants still under construction

3.4 Time Requirements for Reportability - Completed Evaluations

A. Site Licensing will review the completed Part 21 evaluation within 60 days of the discovery of the deviation or failure to comply.

B. Site Licensing will, within five working days, submit the completed evaluation to senior site rnanagerneot.

C. The NRC shall be notified by facsimile, which is the preferred method, by contacting ?heNRC Operations Center at the telephone numbers located in 10 CFR 21.21 (c)(3)(i) within two days of the information being provided to Senior site management. Verification ?hatthe facsimile has been received should be made by calling the NRC Operations Center.

D. A written report will be submitted to the NRC Document Control desk and a copy will be sen?to the appropriate Regional Administrator within 30 days following receipt of information by senior site management.

3.5 Interim Reporting

An interim report shall be made in writing within 60 days of discovery and shall contain, as a minimum, available information about the deviation or failure to comply that is being evaluated, and shall also state when the evaluation will be completed, if the evaluation can not be completed within 60 days of discovery.

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PART 21 EVALUATION **SHEET**

Type of Document ______________________ References ______________________

A. Does the deficiency involve a component, that is installed in the plan!? Yes ☐ No ☐

If **"yes",** then reportability should be evaluated under **10 CFR** 50.72, 50.33 (LER), or 73.71 (Safeguards Events). **A** separate Part **21** report is not required. If the event is determined to be not reportable under 50.72, 50.73, *or* **73.71**, *then* the obligations of Part 21 are stili met by !he evaluation. If "no", continue with B.

B. Does the component or service meet the Part 21 definition of a "Basic Component"? Yes ☐ No ☐

If "yes", go to C. If "no", go to E; the item is not reportable

C. Does !he deficiency involve:

1. A failure of the facility, activity. or basic component supplied to TVA, tu compiy with the Energy Reorganization Act of 1974, or any applicable NRC license requirenents arid regulations, or any rule or order issued by NRC to TVA? Yes ☐ No ☐ It "yes", go *to* D

2. **A** loss of safety function to the extent that if the component was installed in the plant there would be *a* major reduction in the degree of protection provided to the public health and safety? Examples would include moderate exposure to or release of licensed material or major degradation of essential safety-reiated equipment or major deficiencies involving design, construction, inspection, test, or use. Yes ☐ No ☐ If "yes", go to D

3. A departure from the technical requiremerits for a delivered component or service as set forth in a procurement document? Delivery occurs upon acceptance by TVA (*e.g.*, at receipt inspection). Yes ☐ No ☐ If "yes". go to D

Answer ali three questions under C above. If all "No's", this deficiency is not reportable. Go to "E". If any Yes's", continue as indicated **by** the applicable 'GO TO" statement.

D. Could the deviation or noncompliance have caused a substantial safety hazard (By definition, item C2 constitutes a substantial safety hazard.)? Yes ☐ No ☐

(If "No," this item is not reportable - If "Yes." this item is reportable; under Part 21.

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PART 21 EVALUATION SHEET

E. Is this item potentially reportable by TVAN under Part 21? Yes ☐ No ☐

Prepared by: ______________________ Date ____________

Responsible Manager: ______________________ Date ____________

SITE LICENSING MANAGER REVIEW (OR DESIGNEE)

F. *Has* this item been reported by TVA, another licensee. or by a vendor? Yes ☐ No ☐

If "yes," a separate 10 CFR 21 report is not required.

If "no," complete a Part 21 Report (App. F) and notify Site Licensing or Senior Site Management.

Tracking Data:

Date forwarded to Site Licensing ____________

Date received by Site Licensing ____________

Date Senior Site Management informed ____________

Date NRC notified (initial notification) ____________

Date NRC notified (written report) ____________

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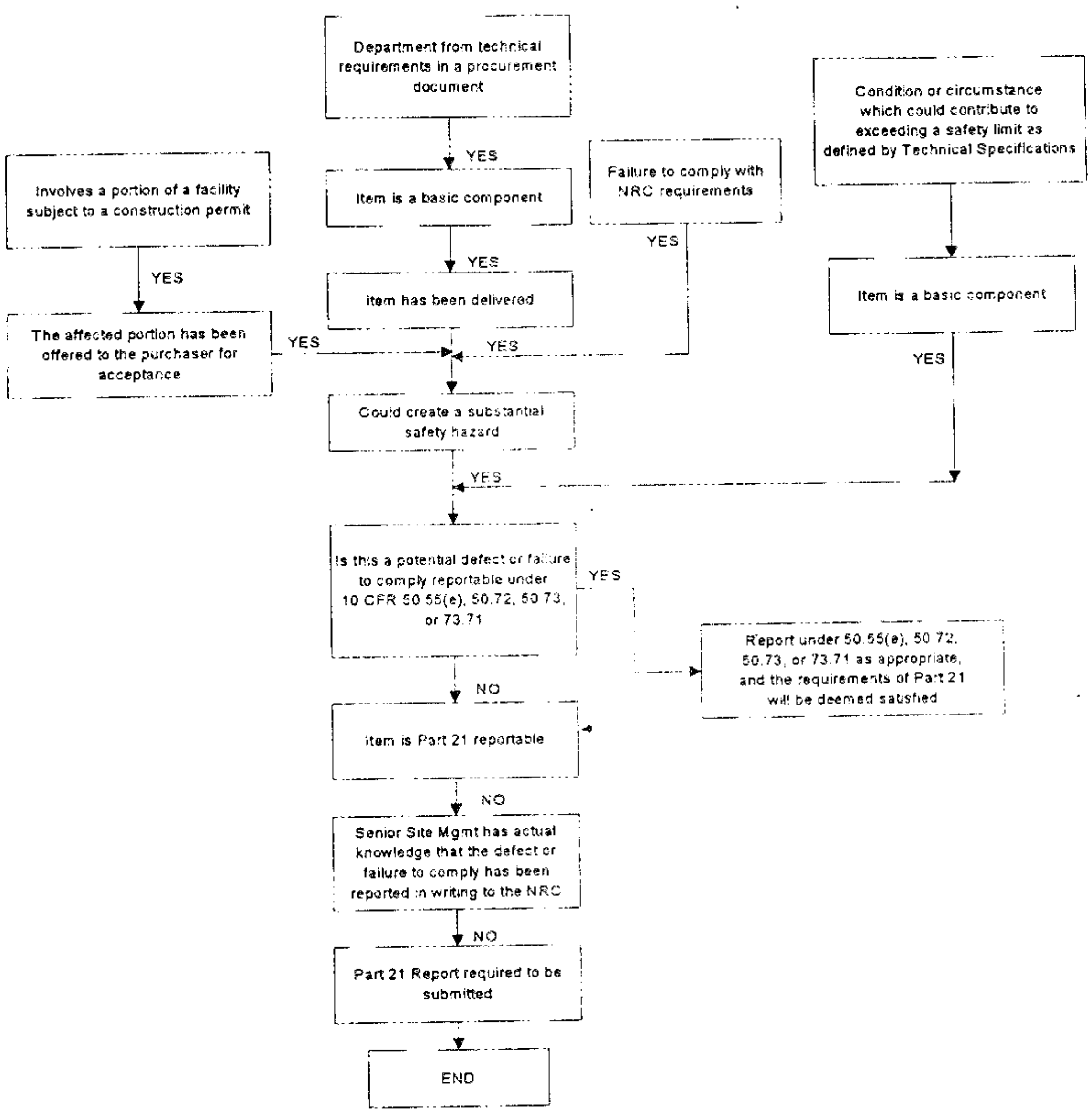
EVALUATION LOGIC FOR PART 21
Figure 1



APPENDIX H
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REPORTING OF DECOMMISSIONING FUNDING

1.0 PURPOSE

The purpose *of* this Appendix is to identify the minimum requirements for submitting a decommissioning funding report to the NRC as required by 10 CFR 50.75, "Reporting and Recordkeeping for Decommissioning Planning." This Appendix also identifies requirements for notifying NRC when permanently shutting down the operation of a reactor, as required by 10 CFR 50.54(bb).

2.5 **SCOPE**

TVA shall report decommissioning funding plans for its nuclear plants as required by 10 CFR 50.33(k), "Contents of Applications; General information." 10 CFR 58.75 establishes requirements for: 1) indicating to NRC how a licensee will provide reasonable assurance that funds are available for the decommissioning process, and 2) reporting tirneframes. This Appendix provides both the minimum information that is required to **be** submitted in the deconimissioning funding report and the required reporting timeframes. Additionally, this Appendix provides the minimum reporting requirements (e.g., reporting timeframes, description of the program that TVA intends to implement for managing all irradiated fuel until it is transferred to the Department of Energy) for notifying NRC when permanently shutting down the operation of a reactor, as required by 10 CFR 50.54(bb).

3.0 **REQUIREMENTS**

3.1 **General 10 CFR 50.35 Reporting Requirements**

10 CFR 50.33(k) and 10 CFR 50.75(b) require nuclear plant license applicants and licensees to submit decommissioning funding reports. License applicants are required to submit information regarding how reasonable assurance will be provided that funds will be available for decommissioning pursuant to 10 CFR 50.33(k) and 10 CFR 50.75. License holders are required to report periodicaliy on the status of their decommissioning funding pursuant to 10 CFR 50.75(f)(1) as described below.

A. Written Report Content

The information in this report must include, at a minimum:

1. The amount of decommissioning funds estimated to be required pursuant to 10 CFR 50.75(b) and (c);

2. The amount accumulated to the end of the calendar year preceding the date of the report:

3. A schedule of the annual amounts remaining to **be** collected;

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4. The assumptions used regarding:

   a) Rates of escalation in decommissioning *costs;*
   b) Rates of earnings on decommissioning funds; and
   c) Rates of other factors used in funding projections;

5. Any modifications occurring to a licensee's current method of providing financial assurance since the last submitted report; and

6. Any material changes to trust agreements.

B. Responsibilities

1. The Treasurer's Office is responsible for providing input to Corporate Licensing for the decommissioning report for items 2, 3, 4.b, 5, and 6 prescribed in Section 3.3.

2. Corporate Nuclear Engineering is responsible for providing input to Corporate Licensing for the decommissioning report for items 1, 4a, and 4c. prescribed in Section 3.2.

3. Corporate Licensing is responsible for preparing and submitting the final decommissioning report.

C. Required Reporting Pimeframes

1. TVA shall report. on a calendar-year basis, to NRC by **March 31, 1999,** and at least once every 2 years thereafter on the status of its decommissioning funding for each licensed reactor that it owns.

2. TVA shall submit a decommissioning report annually, when:

   a) A reactor power plant is wiihin 5 years of the projected end of its operation;

   b) Conditions have changed such that a reactor power plant will close within 5 years (before the end of its licensed life);

   c) A reactor power plant has already closed (before the end of its licensed life); or

   d) A plant is involved in a merger or acquisition

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3.2 Permanent Reactor Shutdown Reporting Requirements

TVA shall, within *2* years following permanent cessation of operation of reactor or **5** years before expiration of the reactor operating license, which ever occurs first, submit written notification to the Commission in accordance with 10 CFR 50.54(bb).

A. This written notification shall:

1. Include a description of TVA's program for managing and funding the irradiated fuel at the reactor until title to and possession of the irradiated fuel is transferred to the Secretary of Energy.

2. Demonstrate to NRC that the elected actions are consistent with NRC requirements for licensed possession of irradiated nuclear fuel and that the actions will be implemented in a timely basis.

3. Verify, for actions requiring NRC prior approval. that submittals have been or will be made to NRC.

4. **Be** retained as a record until expiration of the reactor operating license

B. **TVA** shall notify the NRC of any significant changes in the proposed waste management program as described in the initial notification.

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COMMUNICATION WITH THE NRC FOLLOWING A SIGNIFICANT OPEKATIONAL EVENT

1.0 PURPOSE

The purpose of this Appendix is to define the communications that needs to be established with the site resident inspectors and the regional adniinistrator staff within *24* to 36 hours following a significant operational event that could result in an incident investigation by the NRC.

2.0 SCOPE

This appendix briefly discusses the NRC decision making process for determining their inspection response following notification of a significant operational event and provides guidance on the need for and types of information required by the NRC during this decision making process.

During the site investigation phase following a 10 CFR 50.72 notification for a plant trip or another significant equipment malfunction or failure, a clear communication path needs to be maintained with the resident inspector.

Background

Upon notification of a significant operational event, the regional administrator and his staff will perform the initial review to assess the safety significance of the event. The guidance provided in NRC's Management Directive MD-8.3 "NRC Incident Investigation Program" is used lo assess the level of response required. The criteria for determining between an Incident investigation TEam (IIT), an Augmented Inspection Team (AIT) or a special inspection is based on a combination of deterministic criteria and an estimation of the Conditional Core Damage Frequency (CCDF) of the actual plant configuration at the time of the significant operational event.

in determining the risk significance of the event the NRC is instructed to assess 1) the potential influence on risk of the dominant core damage sequences, 2) level of confidence in failure Or unavailability values assumed for these sequences, and 3) level of confidence of equipment failure/recovery and their influence on the CCDF.

An accurate estimated CCDF is crucial to the determination of the inspection response. With the high reliance of an estimated CCDF on the level of confidence of equipment recovery and satisfactory completion of operator actions, an accurate CCDF requires a thorough understanding of the status of equipment at the time of the event and the causes of failures for any equipment that did not perform as designed.

3.0 REQUIREMENTS

A communication path could be accomplished through normal resident to licensee manager communication or through the trip response team licensing representative. This decision should be based on the complexity of the event and the event investigative team structure. The communication process must assess that sufficient and timely information is being provided to the NRC decision makers.

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As the site investigation evolves and the failure modes and causes of equipment and performance issues are identified, it is critical that the NRC be kept informed of the evolving analysis findings as they pertain to the following crucial estimated CCDF inputs:

- Clear understanding of the event sequences, equipment failures, equipment not available for mitigation, and operator performance problems;
- Perspectives on equipment failures that address suspected causes, potential extent of condition, arid ability of operators to recover failed equipment:
- Perspectives on unavailable equipment and ability to restore functions to **support** risk significant scenarios;
- Perspectives on operator performance problems and their ability to recover or restore critical functions.

NOTE It is important to recognize that the MD-8.3 decision process not only assesses what happened, it also assesses what might have happened with respect to risk Significant scenarios.

APPENDIX J
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INTERNAL NOTIFICATION OF EVENTS REQUIRING SERIOUS ACCIDENT INVESTIGATIONS

1.0 PURPOSE

The purpose of this appendix is to provide the internal management notification requirements for serious accidents, as prescribed in TVA-SPP-18.010, "Conduct Serious Accident Investigation." This appendix applies to all serious accidents that result in any of the following occurrences. EXCEPTION--Radiological control and nuclear operational safety incidents subject to other specific reporting and investigation requirements are investigated by TVA Nuclear (TVAN) as required by applicable procedures.

2.0 REQUIREMENTS

2.1 Serious Accidents include any of the following occurrences:

A. A fatality or in-patient hospitalization of three or more TVA employees within 30 days of an accident.

6. Other events (including property damage only) which under slightly different circumstances would have met or may meet the following provisions:

1. Falls (usually from elevation) causing head injury, broken bones, and/or other serious injury.
2. Electric contact resulting in current flow through the body andior loss of consciousness.
3. Electric are causing second- and/or third-degree burns to the body.
4. Being caught in or by equipment/machinery causing head injury, broken bones, and/or other serious injury.
5. Thermal burns causing second and/or third degree burns to the body
6. Being struck by equipment/machinery or falling objects causing head injury, broken bones: and/or other serious injury.
7. Overpressure resulting in component failure (gas, hydraulic. or air! causing head injury, broken bones, and/or other significant traumatic injury.
8. Release of latent or kinetic energy (tension [objects under compression] or projectiles [objects being thrown]) causing head injury broken bones. andior other serious injury.

2.2 Manager in Charge of Workplace, or designee, performs the following initial actions:

NOTE Checklist items are listed in the general order of preference: however, many of these can occur simultaneously.

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A. Notifies the TVA Police for assistance.

B. Gathers the following notification information:

1. Accident location:
2. Time of the accident:
3. Name(s) of the individual(s) involved:
4. Extent of injuries:
5. Contact person (name and telephone number):
6. Brief description of the occurrence:

C. Immediately notifies the following persons and provides the information listed in Section 2.2.B.

NOTE If either of the individuals listed in Sections 2.2.C.1. or 2.2.C.2. is unavailable, do not delay in contacting the TVA Operator or the Program Manager, Corporate Safety [Refer to Section 2.2.C.3].

1. Responsible executive vice president of the organization where the accident occurred.
2. Safety manager of the organization.
3. TVA Operator at (865)-632-2101. Inform the operator that this call is to report a serious accident to the Program Manager, Corporate Safety. Alternate numbers for Corporate Safety are:

   a) Office: (865)-632-7753 or 7756 (during business hours).
   b) Cell phone: (865)-414-8819.
   c) Pager: (800)-201-8139.

2.3 Licensing will evaluate those serious accidents that Operations determined were not reportable per 10CFR50.72(b)(2)(xi) [refer to Appendix A, Section 3.1.C.4.] to determine if a "courtesy" phone call to NRC is appropriate.

NRC EVENT NOTIFICATION WORKSHEET
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U.S. NUCLEAR REGULATORY COMMISSION
OPERATIONS CENTER

NRC EVENT NOTIFICATION WORKSHEET

EN # ____________

NOTIFICATION TIME	FACILITY OR ORGANIZATION	UNIT	CALLER'S NAME	CALL BACK #: or ( ) ___ - ______

EVENT TIME & ZONE	EVENT DATE
	/ /
POWER/MODE BEFORE	POWER/MODE AFTER

Event Classifications	
General Emergency	GEN/AAEC
Site Area Emergency	SIT/AAEC
Alert	ALE/AAEC
Unusual Event	UNU/AAEC
50.72 Non-Emergency	*(see next columns)*
Physical Security (73.71)	DDDD
Material/Exposure	B???
Fitness For Duty	HFIT
Other Unspecified Reqmt.	(see last column)
Information Only	NNF

1-Hr Non-Emergency 10 CFR 50.72(b)(1)	
TS Deviation	ADEV

4-Hr Non-Emergency 10 CFR 50.72(b)(2)	
(i) TS Required S/D	ASHU
(iv)(A) ECCS Discharge to RCS	ACCS
(iv)(B) RPS Actuation (scram)	ARPS
(xi) Offsite Notification	APRE

8-Hr Non-Emergency 10 CFR 50.72(b)(3)	
(ii)(A) Degraded Condition	ADEG
(ii)(B) Unanalyzed Condition	AUNA
(iv)(A) Specified System Actuation	AESF
(v)(A) Safe S/D Capability	AINA
(v)(B) RHR Capability	AINB
(v)(C) Control of Rad Release	AINC
(v)(D) Accident Mitigation	AIND
(xii) Offsite Medical	AMED
(xiii) Lost Comm/Asmt/Resp	ACOM
60-Day Optional 10 CFR 50.73(a)(1)	
Invalid Specified System Actuation	AINV
Other Unspecified Requirement (Identify)	
	NONR
	NONR

DESCRIPTION

Include: Systems affected, actuations & their initiating signals, causes, effect of event on plant, actions taken or planned, etc. (Continue on back)

NOTIFICATIONS	YES	NO	WILL BE
NRC RESIDENT			
STATE(s)			
LOCAL			
Other Gov Agencies			
Media/Press Release			

Anything Unusual or Not Understood?	Yes *(Explain above)*	No
Did All Systems Function As Required?	Yes	No *(Explain above)*
Mode of Operation Until Corrected:	Estimated Restart Date	Additional INFO on Back? ☐ Yes ☐ No

NRC EVENT NOTIFICATION WORKSHEET
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USNRC Operations Center

RADIOLOGICAL RELEASES: CHECK OR FILL IN APPLICABLE ITEMS (specific details/explanations should be covered in event description)

	Liquid Release		Gaseous Release		Unplanned Release		Planned Release		Ongoing		Terminated
	Monitored		Unmonitored		Offsite Release		T.S. Exceeded		RM Alarms		AREAS Evacuated
	Personnel Exposed or Contaminated				Offsite Protective Actions Recommended			*State release path in description.			

	Release Rate (Ci/sec)	% T. S. Limit	HOO Guide	Total Activity (Ci)	% T.S. Limit	HOO Guide
Noble Gas			0.1 Ci/sec			1000 Ci
Iodine			10 uCi/sec			0.01 Ci
Particulate			1 uCi/sec			1 mCi
Liquid *(excluding tritum & dissolve noble gases)*			10 uCi/min			0.1 Ci
Liquid (tritium)			0.2 Ci/min			5 Ci
Total Activity						

	Plant Stack	Condenser/Air Ejector	Main Steam Line	SG Blowdown	Other
RAD Monitor Readings:					
Alarm Setpoints:					
% T.S. Limit (if applicable)					

RCS or SG Tube Leaks: Check or Fill in Applicable Items: (specific details/explanations should be covered in event description)

Location of the Leak *(e.g., SG #, valve, pipe, etc.)*:

Leak Rate:	Units: gpm/gpd	T. S. Limits:	Sudden or Long Term Development:
Leak Start Date:	Time:	Coolant Activity & Units:	Primary - Secondary -

List of safety Related Equipment not Operational:

EVENT DESCRIPTION *(Continued from front)*

BROWNS FERRY NUCLEAR PLANT
JOB PERFORMANCE MEASURE

JPM NUMBER: A3.1R

TITLE: REVIEW A RADIOLOGICAL SURVEY MAP

TASK NUMBER: N/A

SUBMITTED BY:______________________ DATE:

VALIDATED BY:______________________ DATE,

APPROVED: ______________________ DATE:
TRAINING

PLANT CONCURRENCE: ______________________ BATE:
OPERATIONS

* Examination JPMs Require Operations Training Manager or Designee Approval and Plant Concurrence

BROWNS FERRY NUCLEAR PLANT
JOB PERFORMANCE MEASURE

JPM NUMBER: A2.1R

TITLE: CORE SPRAY SYSTEM VALVE TIMING

TASK NUMBER:

SUBMITTED BY: ______________________ DATE: ________

VALIDATED BY: ______________________ DATE: ________

APPROVED: ______________________ DATE: ________
TRAINING

PLANT CONCURRENCE: ______________________ DATE: ________
OPERATIONS

Examination JPMs Require Operations Training Manager or Designee Approv
Plant Concurrence

BROWNS FERRY NUCLEAR PLANT
JOB PERFORMANCE MEASURE

REVISION LOG

Revision Number	Date	Pages	Description of Revision
0			New

BROWNS FERRY NUCLEAR PLANT
JOB PERFORMANCE MEASURE

OPERATOR: ______________________

RO ______ SRO ______ DATE: ______________

JPM NUMBER: A2.1R

TASK NUMBER:

TASK TITLE: CORE SPRAY SYSTEM II VALVE TIMING

K/A NUMBER: 2.2.2 K/A RATING: RO 4.0 SRO: 3.5

****

TASK STANDARD: DETERMINE 2-FCV-75-50 stroke time is within its maximum closure time but outside its normal range and DIRECT an engineering evaluation be performed.

LOCATION OF PERFORMANCE: SIMULATOR X PLANT ___

REFERENCES/PROCEDURES NEEDED: 2-SR-3.5.1.6(CS II) (**Completed to Step 7.24, 7.24 is next step to perform**)

VALIDATION TIME: CONTROL ROOM: X LOCAL: ______

MAX. TIME ALLOWED: ______ (Completed for Time Critical JPMs only)

PERFORMANCE TIME: ______ CONTROL ROOM ______ LOCAL

COMMENTS: ____

Additional comment sheets attached? YES ____ NO ____

RESULTS: SATISFACTORY ______ UNSATISFACTORY ______

EXAMINER SIGNATURE: ______________ DATE: ______

BROWNS FERRY NUCLEAR PLANT
JOB PERFORMANCE MEASURE

****

SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. Ensure you indicate to me when you understand your assigned task and when you have completed the assigned task.

****

INITIAL CONDITIONS: You are a Unit 2 Operator. Core Spray System II is inoperable due to maintenance on 2-FCV-75-50, Core Spray System II Test Valve. You ~~is~~ are performing 2-SR-3.5.1.6 (CS II), Core Spray Flow Rate Loop II which is the specified PMT for the maintenance that was performed on the valve. All steps through step 7.23 are complete ~~[illegible]~~.

INITIATING CUES: You are to continue the surveillance, starting at step 7.24 and continuing until complete with the surveillance ~~[illegible] stop by the examiner~~. You may take the time necessary to ensure you understand the task you are given before you start.

(Stop watch provided)

Need APP. HAND OUT PAGE

START TIME___________

Beginning at Step 7.24

****

Performance Step: Critical X Not Critical__

7.24 FULLY **OPEN** 2-FCV-75-50 using CORE SPRAY SYS II TEST VALVE, 2 HS-75-50A.

Standard:

PLACES HANDSWITCH in open position and fully **opens valve as inidicated by RED LIGHT only..**

SAT____UNSAT___ N/A____ COMMENTS:______________________

____

****

Examiner Note: Valve in next step should close within 19-23 seconds which is outside the normal range and within maximum.

MOTES:
REFER TO Illustration 1, Process for Stroke Timing Valves per the ASME OM Code.

****

Performance Step: Critical X Not Critical

7.25 CLOSE and TIME 2-FCV-75-50 using CORE SPRAY SYS PI TEST VALVE, 2-HS-75-50A and RECORD the stroke time below.

Standard:

Records closure time of valve in table for step 7.25. Closure time should be below the normal range and maximum value.

SAT___ UNSAT___ N/A___ COMMENTS:___________________________

****

****

Performance Step: Critical X Not Critical __

7.25.1 VERIFY the time recorded is within the maximum value listed.

Standard:

Notes that time *recorded* in *step* 7.25 is within the maximum value listed as Acceptance Criteria.

SAT ___ UNSAT ___ N/A ___ COMMENTS: ________________

****

****

Performance Step: Critical X Not Critical

7.26 IF the stroke time measured in step 7.25 is within the maximum value listed and outside the normal range, then

Perform the following (otherwise NA this ection) (BFPER971386)

7.26.1 OPEN the 2-FCV-75-50 using CORE SPRAY SYS II TEST VALVE, 2-HS-75-50A.

Standard:

DETERMINES time <u>outside the normal range</u> and OPENS 2-FCV-75-50 to allow re-stroking and retiming the valva.

SAT ___ UNSAT ___ N/A ___ COMMENTS: ________________

****

****

Performance Step: Critical X Not Critical __

7.26.2 CLOSE **and *TIME* 2-FCV-75-50, using CORE SPRAY SYS II TEST VALVE, 2-HS-75-5QA and RECORD the restroke time** on **Attachment 7.**

Standard:

PLACES AND HOLDS handswitch in *close* and times valve closure from movement of HS to Green Light Only. Notes time is again below normal expected range. Records Information on Attachment 7 of SR.

SAT____ UNSAT____ N/A____ COMMENTS:________________

____

****

****

Performance Step: Critical X Not Critical ___

7.26.3 VERIFY the restroke time recorded on Attachment 7 is within the maximum values *listed.* [BFPER971386].

Standard:

DETEMINES re-stroke time is below the maximum value listed. *Acceptance* criteria met

SAT____ UNSAT____ N/A____ COMMENTS:________________

____

****

EXAMINERS NOTE;

IF STEP 7.27 IS STARTED THEN ASK STUDENT **TO STOP SR** PERFROMANCE, **EVALUATE THE ITEMS FOLLOWING;**

****

EXAMINERS NOTE: This Illustration is a flowchart that incorporates the restroking of the valve when outside the normal range and provides additional directions for requesting an engineering evaluation due to this valve being outside the normal stroke times. This illustration may have been referred too during the performance of the surveillance. If not provide the following cue if the student was successful in completing the initial timing and restroke timing of the valve.

CUE: Determine any needed evaluations as a result of the valve being outside the normal stroke times.

Performance Step: Critical_X__ Not Critical_

Utilizes Illustration 1 flow chart to determine required action.

Enters flow chart at. START		___
Stroke Time Valve Per SR		___
Stroke 'Time Within Maximum Limit?	YES	___
Stroke Time within Normal Range	NO	___
Restroke Valve		___
Restroke Time within Maximum Limit?	YES	___
Restroke Time within Normal Range	NO	___
Provide Copy of Attachment to Engineering		___
Engineering Evaluates Stroke Times within 96 Hours		___

Standard:

Contact Engineering and directs them to evaluate the stroke times within 96 hours for 2-FCV-75-50.

SAT___ UNSAT___ N/A___ COMMENTS:______________________________

____

****

****
NON CRITICAL

Followup Question:

If this were the last PMT item for the maintenance on this valve, would you declare the PMT complete Sat and declare the system operable for TS prior to evaluation by Engineering being complete.

Expected Response;

Since the work on this system was directly related to 2-FCV-75-50, the maintenance that was performed on this valve has evidently changed the stroke characteristics of the valve or limit switch settings. There may be a reliability issue or require re-baselining.

However, all Acceptance Criteria was met and nothing procedurally requires the stroke time to be within normal values prior to making an operability call, or successful completion of SR.

****

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT

SURVEILLANCE PROCEDURE

2-SR-3.5.1.6(CS II)

CORE SPRAY FLOW RATE LOOP II

REVISION 12

QUALITY RELATED

PREPARED BY: Keith Smith

RESPONSIBLE ORGANIZATION: OPERATIONS

APPROVED BY: PHILLIP CHADWELL

EFFECTIVE DATE 03/05/2003

LEVEL OF USE: **CONTINUOUS USE**

PAGES AFFECTED 61

REVISION DESCRIPTION IC-13 ENHANCEMENT

Page 61, Replaced illustration 1 that referenced attachment 3 for the restroking of any valves, with a new illustration. The original iilustration could not be changed without special software program. The New illustration is a generic restroke illustration that may can be modified using the current Microsoft Word program

The actual illustration being deleted does not show as a revision and only the illustration title appears as a change.

1.0 INTRODUCTION

1.1 Pu se

This surveillance procedure is performed to determine the operability of Core Spray (CS) System LOOP II in conformance with requirements specified in Technical Specification (TS) Surveillance Requirements (SRs) 3.5.1.6and 3.5.2.4, and to functionally test the CS LOOP II minimum flow valve as required by SR 3.3.5.1.2. This SR also tests CS Pumps 2B and 2D, and associated valves, to the ASME OM Code Program of TS 5.5.6. The equipment area coolers associated with Core Spray Pump 2B and 2D as specified in the Technical Requirements Manual (TRM – Sections 3.3.3 and 3.5.3) will also be verified.

7.2 Scope

1.2.1 This procedure verifies CS Pumps 2B and 2D can be operated from the main control room to deliver rated flow and pressure against a simulated reactor pressure. This satisfies TS SRs 3.5.1.6 and 3.5.2.4 for CS Loop II.

1.2.2 This procedure also serves to demonstrate compliance with the TS SRs, TRM Surveillance Requirements (TSRs), and ASME OM Code program requirements indicated below:

CORE SPRAY SYS II TEST VALVE 2-FCV-75-50 is exercised (i.e., open and closed) by this procedure to demonstrate its operability per the ASME OM Code Program.

The CS pump room cooler located is the NE Corner Room at Elevation 541 5' of the Reactor Building is operated during performance of this procedure to demonstrate its cooling ability. Specifically, the room cooler must start and operate upon start of either CS Pump 2B or 2D. This procedure satisfies TSR 3.3.3.2.2(for Table 3.3.3.2-1function 4) and TSR 3.5.3.1 for CS Loop II.

The CS Loop Minirnum Flow Valve, 2-FCV-75-33, is verified to open and close as system parameters require to satisfy TS SR 3.3.5.1.2 for Table 3.3.5.1-1 Function 1.d.

The CS System LOOP II discharge piping venting is verified by this procedure This satisfies TS SR 3.5.11 and SR 3 5 2 2 testing requirenents for CS Loop II.

1.2.3 This procedure is also used to collect applicable data as required by 2-SI-3.1.1 or 2-SI-3.1.8. This procedure in conjunction with SRs/SIs listed as being ASME type in Surveillance Program Matrix. fully implements the ASME OM Code Program of TS 5.5.6 and 0-TI-362.

1.2 Scope (Continued)

1.2.4 Satisfactory completion of this procedure meets the *ASME* OM Code requirements as specified in 2-SI-3.2.1 for cycling the following valves:

2-FCV-95-50
2-CKV-75-537B (cycle valve open and closed)
2-CKV-751-5375 (cycle valve open and closed)
2-CKV-75-570B (cycle valve open and closed)
2-CKV-75-57OD (cycle valve open and closed)

1.2.5 This procedure and 2-SR-3.5.1.6(CS I) fully satisfy SR 3.3.5.1.2 for Table 3.3.5.1-1 Function 1.d.

1.2.6 This procedure and 2-SR-3.5.1.6(CS I) fully satisfy SRs 3.5.1.6 and 3.5.2.4 for CS pumps.

1.2.7 This procedure may be performed in lieu of 2-SR-3.5.1.1(CS II) to satisfy SRs 3 5.1 1 and 3.5.2.2 for CS Loop II. Note that the frequency of SRs 3.5.1 1 and 3.5.2.2 is once per 32 days.

2.3 Frequencv

This procedure is to be performed once per 92 days. This satisfies the requirements of SR 3.3.5.1.2 (once per 92 days), and SRs 3.5.1.6 and 3.5.2.4 (in accordance with the Inservice Testing Program).

1.4 Applicability

LCO 3.3.5.1 Functions: MODES 1, 2, 3, 4[(a)], and 5[(a)].

LCO 3.5.1 Functions: MODES 1, 2, and 3.

LCO 3.5.2 Functions: MODE 4, and MODE 5 except with the spent fuel storage pool gates removed and water level ≥ 22 ft over the top of the reactor pressure vessel flange

TRM LCO 3.3.3.2 Functions: MODES 1, 2, 3, 4[(a)], and 5[(a)]

TRM LCO 3.5.3 Functions: (a).

(a) When associated subsystem(s) are required to De operable

2.0 REFERENCES

2.1 BFN Unit 2 Technical Specifications

Section 3.3.5.1, Emergency Core Cooling System (ECCS) Instrumentation

Section 3.5.1, ECCS-Operating

Section 3.5.2, ECCS-Shutdown

Section 5.5.6, Inservice Testing Program

2.2 Technical Requirements Manual

Section TI? 3.3.3.2, Low Pressure Area Cooler Instrumentation.

Section TR 3.5.3, Equipment Area Coolers.

2.3 BFN UFSAR

Section 6.4.3, Core Spray System Description

Section 6 5.2.4, Core Spray System

Section 7.4, Core Standby Cooling System Control and Instrumentation

2.4 Plant Instructions

0-GOI-300-2, Electrical

0-OI-23, Residual Heat Removal Service Water System

0-01-67, Emergency Equipment Cooling Water System

2-01-75, Core Spray System

2-SI-3.1.1. Core Spray Pump Performance

2-SI-3.1 8, Core Spray System Baseline Data Evaluation

2-SI-3.2.1, ASME Section XI Valve Performance

2-SI-4.2.B-60FT(II) Core Spray Pump Area Cooler Fan Thermostat Functional (2-TS-64-73).

2-SIMI-75B, Core Spray System Scaling and Setpoint Documents

2.4 Plant Instructions (Continued)

0-TI-230, Vibrating Monitoring and Diagnostics

0-TI-280, Calculations of Flow Transmitter Output for Use with ASME Section XI

0-TI-362, Inservice Testing of Pumps and Valves

SPP-8.1, Conduct of Testing

SPP-3.2, Electrical Equipment Environmental Qualification (EQ) Program

SPP-10.3, Verification Program

2.5 Plant Drawings

0-730E930 Sheets 1 and 13 through 20, CS System Elementary Diagrams

0-731E761 Sheets 10 and 11, Emergency Equipment Elementary Diagrams

2-45N2750-12, 480V Reactor MOV Board 2B Connection Diagram

2-45E779-8, 480V Shutdown Aux Power Schematic Diagram

2-45E779-10, 480V Shutdown Aux Power Schematic Diagram

2-45E779-16, 480V Shutdown Aux Power Schematic Diagram

2-47E814-1, CS System Flow Diagram

2-47E610-75-1, Mechanical Control Diagrams CS System

2-47E611-75-1 through -3, CS System Mechanical Logic Diagrams

2.6 Vendor Manuals

BFN-0-CVM-G080-2105-07, GEK-779A, Controls Sys Manuals Vol 7 [Unit 2 only)

BFN-2/3-VTM-B260-0010, Vendor Technical Manual for Bingham-Williamette 12X16 14 1/2 Single Stage CVDS Pumps.

2.7 Other Documentation

GE Services Information Letter (SIL) No 93, System Protective Devices.

MD-Q2075-890109, Core Spray Acceptance Criteria for Technical Specification Operability Surveillance (B22 900118 174)

NESSD 2T-064-0072-00-01, Nuclear Engineering Setpoint and Scaling Document

ED-Q2075-880532 Setpoint and Scaling and Calculation for 2-F-75-21 and -49

3.0 PRECAUTIONS AND LIMITATIONS

3.1 This test manually starts CS Pumps 2B and 2D and measures system pressure with various pump configurations.

3.2 If maintenance other than what is provided in this procedure becomes necessary, a work order should be generated.

3.3 [NRC/C] The CS pump motor nameplate full load current is 80 amperes. The maximum allowable continuous running current is 92 amperes which is based on full load current multiplied by the motor service factor of 1.15. [SLT 861087005]

3.4 The stroke time for all motor-operated valves shall be measured to the nearest tenth of a second. Stroke time is defined as the period from initial switch movement to Panel indication of completed valve travel.

3.5 ASME OM Code information should be reviewed and recorded in accordance with 2-SI-3.1.1, 2-SI-3.1.8, and 2-SI-3.2.1 as appropriate, within 4 days of the completion of this procedure.

3.6 Annunciators which will alarm during performance of this procedure are specified on Attachment 5 and within the procedure text as notes.

3.7 When starting pumps with the injection valves and test valve closed, the CORE SPRAY SYS II DISCH PRESS HIGH (2-XA-55-3F, window 30) annunciator may alarm momentarily.

3.8 This procedure lifts the power lead (Cable 2ES3308-II, Wire No. 8BX) at Terminal 2 in JB 8790 for 2-TS-64-73 to enable testing of the CS pump room cooler fan due to CS pump operation regardless of area temperature. This operation disables the CS pump room cooler fan thermostat. 2-SI-4.2.B-60FT(II) must be performed on the thermostat to return it to operable status.

All wire lifts shall be taped or covered in such a manner as to prevent personnel or equipment hazards. Wire lifts performed in conjunction with this procedure shall be restored to their as-found configuration (e.g., bend radius).

3.9 If the CS pump room cooler fan thermostat is disabled for greater than 24 hours, a TRM LCO may result.

3.10 A radiation work permit (RWP) may be required to perform this procedure.

3.11 The motor start limitations of 0-GOI-300-2 limit CS pump starts to two starts in succession coasting to rest between starts with the motor initially at ambient temperature or one start with the motor initially at normal operating temperature.

3.0 PRECAUTIONS AND LIMITATIONS (Continued)

3.12 Du · performance of this test, RHRSW Pumps A I, C1, B3, and D3 may start if they are valid in to Emergency Equipment Cooling Water (EECW) System, operable, and NOT running. This constitutes a planned actuation of the EECW System, an Engineered Safeguards Feature (ESF) and, therefore, is NOT reportable.

3.13 The BFN ASME OM Code Ten Year Program for monitoring pump flow and pressure requires that measuring instruments are accurate within two percent. This accuracy requirement is implemented in this procedure by using temporary pressure gauges with this accuracy to monitor pump suction pressure and by directly measuring flow modifier input/output using either a digital volt-ohm meter (BVOM) or the Integrated Computer System (ICS). Existing discharge pressure gauges satisfy the accuracy requirement and do NOT require substitution with temporary measuring instruments.

The ASME OM Code also provides the full scale (FS) range of compliance instrumentation be three times (3X) the expected process value or less to ensure an accurate measurement since compliance instrument accuracy requirements are based on the full scale range of the instrument. The full scale range of the temporary suction pressure gauges specified by step 5.2.2 account for this requirement. The existing discharge pressure gauges comply with the ASME OM Code range requirement and do NOT require substitution with temporary measuring instruments.

3.14 Corrective Actions shall be dispositioned in accordance with SPP-8.1, Conduct of Testing.

3.15 ASME OM Code data collection requires the CS pumps be operated at a repeatable reference value when discharge presstire readings are taken. While the flow rate may be an average of the required value (3200 gpm), the UO should attempt to maintain the flow rate as close as possible to the stated value. This helps to ensure that discharge pressure readings do not vary significantly due to operating point changes from performance to performance of this procedure unless an actual problem exist. [BFPER98-004734-000]

3.16 Any ICS console in the main control rocm may be used for collecting ICS data specified by this procedure. If the ICS console originally selected fails io operate properly during performance of this procedure, another ICS console(s) may be used for completion of test activities provided the failure is isolated to the console in use If an alternate ICS console(s) is used, then the change(s) shall be noted in the post-test remarks section of Attachment 1

3 0 PRECAUTIONS AND LIMITATIONS (Continued)

3 17 The suction pressure gauges used for ASME OM Code data measurements will be measuring a nominal pressure of approximately 5 to 8 psig. The gauges shall have a minimum range of 0 to 10 psig and a maximum range of 0 to 30 psig to ensure compliance with the ASME OM Code The gauges need NOT have the same range. For example, Gauge A may have a range of 0 to 10 psig while Gauge B may have a range of 0 to 30 psig.

The process connection fittings used for gauge installation shall have a minimum withstand pressure rating of 450 psig. The pressure gauges are NOT reqtiired to meet this withstand requirement due to the sense line sine and root valve isolation capability.

3 18 Vibration readings (obtained by Electrical Maintenance using the portable vibration instrument) should be verified to not be in ALARM status. This is NOT acceptance criteria, but is a means of verifying vibration data obtained is valid. Any vibraticn reading exceeding an ALARM limit should be remeasured to verify the reading is valid.

3 19 The observers who verify lights should have spares available to replace burned out. bulbs.

a 20 REFER TO Illustration 1, Process for Stroke Timing Valves per the ASME OM Code Valves that exceed the Maximum allowed stroke time shall not be restroked and must be declared inoperabie.

Date 5-8-03

INITIALS

NOTE:

Prerequisite steps may be performed in any order at the discretion of the Unit Supervisor.

4.0 PREREQUISITES

4.1 This copy of 2-SR-3.5.1.6(CS II) is verified the most current revision. ____

4.2 EECW is in service to support CS NE pump room cooler operation ____

4.3 CS System LOOP II is available for testing ____

4.4 Qualified personnel listed below are available to perform this procedure

UO 1 IM 2 AUO 2 EM 2 ____

4.5 Electrical Maintenance (EM) has been notified of this procedure performance and is ready to perform the applicable pump vibration monitoring sections of this procedure and 3-TI-530. ____

4.6 Instrument Maintenance (IM) has been notified of this procedure performance and is ready to perform the applicable portions of this procedure and 2-SI-4.2.B-60FT(II).

Date 5-8-03

INITIALS

4.0 PREREQUISITES (Continued)

NOTE:

Separate equipment qualification work record packages are required for each maintenance discipline (i.e., EM and IM) to facilitate review and approval.

4.7 Computer-generated environmental qualification work record forms (SPP-9.2-26) have been obtained from Work Planning and will be completed by IM during performance of this procedure for the following terminal block and cable:

2-TB-64-8790
2-@ES-064-3308/II ______

4.8 Computer-generated environmental qualification work record forms (SPP-9.2-26) have been obtained from Work Planning and will be completed by EM to document CS pump motor vibration levels during performance of this procedure:

2-MTR-075-0033
2-MTR-075-0042 ______

Date 5-8-03

INITIALS

4.0 PREREQUISITES (Continued)

NOTE:

The CS LOOP II flow rate may be monitored from an ICS console by requesting either a single value display (SVD 75-43) or the CS mimic.

4.9 **IF** the ICS is used *to* collect CS LOOP II flow rate data, **THEN**

CHECK no gross instrument channel failures have occurred by noting the ICS display to be used indicates within 100 gpm of CORE SPRAY SYS II FLOW Indicator 2-FI-75-49 on Panel 2-9-3; Otherwise **N/A**. ______

4.10 **IF** the ICS is used to collect CS LOOP II flow rate data, **THEN**

PLACE NIA in the following steps;:

7.5.5, 7.15.2, 7.32.2,7.48, and 7.52.8 (Otherwise N/A this step.)

4.11 **IF** the ICS is NOT used to collect CS LOOP II flow rate data, THEN

PLACE NIA in the following steps:

7.5.1 and 7.32.1;(Otherwise NIA this step.) ______

5.0 SPECIAL TOOLS AND EQUIPMENT RECOMMENDED

5.1 Recommended Tools

5.1.1 Lead seals and crimping tool

5.1.2 Slotted head screwdriver set

5.1.3 Crescent wrench

5.1.4 Needle nose pliers

5.2 Recommended Measuring and Test Equipment (M&TE)

5.2.1 Enter information where reqtiired. Vibration M&TE accuracy and frequency response range are controlled by the BFN Vibration Program and have been verified to meet the listed requirements. Verify required range and accuracy for remaining M&TE by reviewing calibration sheets. N/A blanks for instruments not used. For example, only two suction pressure gauges are required and they are usually analog. in this case the digital gauge blanks would be N/A. Likewise. the DVOM is only required if the ICS is no! available, and those blanks will usually be N/A.

Parameter Measured	Recommended Instrument	Required Range	Required Accuracy	Frequency Response Range	Calibration Due date	M&TE ID
Vibration	CSI Model 2100 series or equal	NIA	± 5% of calibrated range	19.89-1000 Hz minimum		
Suction Pressure	Analog gauge A	0-30 psig maximum	± 2% of full scale	N/A		
Suction Pressure	Analog gauge B	0-30 psig maximum	± 2% of full scale	N/A		
Suction Pressure	Digital gauge A	14.2 psig minimum	± 2% of calibrated range	N/A		
Suction Pressure	Digital gauge B	14.2 psig minimum	± 2% of calibrated range	N/A		
Time	Stopwatch	N/A	± 1 second	N/A	N/A	
Voltage	Keithley Model 197 DVOM or equal	498.5 mV minimum	± 2% of calibrated range	N/A		

Date 5/3/03

5.0 SPECIAL TOOLS AND EQUIPMENT RECOMMENDED (Continued)

NOTES:

(1) Accuracy ratios should be calculated in the space provided before leaving the IM Shop area

(2) M&TE with an accuracy ratio less than 1.0 shall NOT be used for procedure test record purposes

5.2.2 Two analog or digital pressure gauges (as listed in Section 5.2.1) and associated process connection fittings (e.g., Tyyon tubing) per Section 3.17.

Gauge A

Accuracy Ratio = Required Accuracy / M&TE Accuracy = 0.6 psig / ( ) psig = ____

Gauge B

Accuracy Ratio = Required Accuracy / M&TE Accuracy = 0.6 psig / ( ) psig = ____

6.0 ACCEPTANCE CRITERIA

6.1 Responses which fail to meet the acceptance criteria stated in this section shall constitute unsatisfactory SR results and require immediate notification of the Unit Supervisor at the time of failure.

The following acceptance criteria shall be demonstrated as required by this procedure:

NOTE:

Step 6 1.1 is used to satisfy the Tech Spec requirement to provide 6258 gpm flow at 105 psid between the reactor pressure vessel (RPV) and primary containment.

6.1.1 CS LOOP II provides at least 6250 gpm flow at 228 psig or greater discharge pressure.

6.1.2 The operating point for CS Pump 2B and 2D shall be within the acceptance criteria range determined by the BFN ASME OM Code Inservice Test (IST) Program. Specifically, CS Pump 2B differential pressure shall lie within ihe range 229 to 279.8 psid and CS Pump 2D differentia! pressure shall lie within the range 220.6 to 269.6 psid while either pump is operating within the ASME OM Code-specified discharge flow rate range.

6 1 3 CQRE SPRAY SYS II PEST VALVE 2-FCV-75-50 closing stroke time does <u>N</u>OT exceed 30 seconds.

6.1.4 CS LOOP II discharge piping is vented

6.1.5 The CORE SPRAY MINIMUM FLOW VALVE. 2-FCV-75-37, opens on lowering flew and closes on rising flow.

6.0 ACCEPTANCE CRITERIA (Continued)

6.1.6 The following valves shall comply with the ASME OM Code IST Program acceptance criteria stipulated below:

	intended function by noting at least a 10 psi drop in CS Pump 2B discharge pressure when CORE SPRAY SYS II MIN FLOW VALVE 2-FCV-75-37 is opened while CS Pump 28 is operating at or near rated conditions.
-CKV-75-570D	delivering 3200 gpm. Valve shall open Sufficiently to perform its intended function by noting at least a 10 psi drop in CS Pump 2D discharge pressure when CORE SPRAY SYS II MIN FLOW VALVE 2-FCV-75-37 is opened while CS Pump 2D is operating at or near rated conditions. Valve shall close sufficiently to prevent backflow by noting CS Pump 2B alone is capable of delivering 3200 gpm

6.2 Responses which fail to meet the acceptance criteria stated in this section shall constitute unsatisfactory TSR results and require irnrnediate notification of the Unit Supervisor at the time of failure.

6.2.1 Core Spray NE Room Cooler automatically starts when either Core Spray Pump 2B or 2D start.

6.3 Steps which determine the above criteria are designated by (AC) next to the initials blank.

Date 5/8/03

INITIALS

7 0 PROCEDURE STEPS

NOTE:

1) All checks, verifications or manipulations by UO will be performed at Panel 2-9-3 unless otherwise noted.

2) [NRCIC] The area fan cooler thermostat is disabled during this procedure and a TWM LCO may be entered

7.1 **VERIFY** the following initial conditions are satisfied:

7.1.1 Precautions and limitations in Section 3.0 have been reviewed. ______

7.1.2 Prerequisites listed in Section 4.0 2re met. ______

7.2 **OBTAIN** permission from Unit Supervisor to perform this test. [NCO 89-0216-002] ______ US

7 3 [NRCIC] **NOTIFY** Unit 1, Unit 2, and Unit 3 Unit Operators (UO) prior to the start of this procedure [RPT 82-16, LER 259/8232] ______

7.4 **RECORD** the date and time started, reason for test, and plant conditions on Attachment 1, Surveillance Procedure Review Form. ______

Date 5/11/03

INITIALS

7.0 PROCEDURE STEPS (Continued)

NOTES:

(1) Lifting of the CS pump room thermostat power lead will enable testing of the NE Corner Room CS pump room cooling fan operation due to CS pump operation regardless of room temperature, This operation disables the area fan cooler thermostat. 2-SI-4.2.B-60FT(II) must be performed on the thermostat to return it to an operable status.

(2) [NRC/C] When the CS pump room cooler thermostat is disabled, an LCO associated with the area cooler fan logic may be entered (TR Table 3.3.3.2-1). If the CS pump room cooler thermostat is disabled for greater than 24 hours, a CS System LCO may result if CS Loop II is required to be operable. Step 7.51 verifies 2-SI-4.2.B-60FT(II) was performed and records the time. The performer shall notify the Unit Supervisor prior to exceeding 24 hours or if it becomes apparent the LCO time limit cannot be met. [NCO 89-0216-002]

(3) JB 8790 is located in the NE Corner Room at Elevation 541.5'

(4) Steps 7.5, 7.6, and 7.7 may be performed concurrently

7.5 **PERFORM** the following:

7.5.1 LIFT, **TAPE,** and **LABEL** the power lead (Cable 2ES3308-II, Wire No 8BX) at Terminal 2 in JB 8790 for 2-TS-64-73

_______ 1st

_______ 2nd

7.5.2 [NRC/C] RECORD below the time step 7.5.1 is completed: [NCO 89-0216-002]

Time ____________

_______ IM

Date 5/19/03

INITIALS

7 0 PROCEDURE STEPS (Continued)

NOTES:

(1) The work described in the following step shall be performed using skill of craft.

(2) The pressure gauges installed in the following step are located at Instrument Rack 2-LPNL-25-60 in the Northeast Corner Room at Elevation 519' of the Reactor Building.

(3) The centerline of Pressure Gauge A must be at the same elevation as 2-PI-75-32 centerline when readings are taken.

(4) The centerline of Pressure Gauge E must be at the same elevation as 2-PI-75-41 centerline when readings are taken.

7.5.3 **INSTALL** a temporary pressure gauge and associated process connection hardware at each test tee connection for 2-PI-75-32 and 2-PI-75-41 per Attachment 6, Section A. ______ IM

7 5 4 **TAG** the gauge on 2-PI-75-32 test tee as PRESSURE GAUGE A and the gauge on 2-PI-75-42 test tee as PRESSURE GAUGE B ______ IM

NOTE:

The BVOM test connections which may be installed in the following step are located at Panel *2-9-19* in the Auxiliary Instrument Room.

7.5.5 **IF** the ICS is NOT used to collect CS LOOP II flow rate data. THEN

TEMPORARILY INSTALL DVOM test connections at Flow Modifier 2-FM-75-49 per Attachment 3; (Otherwise N/A.) ______ IM

Date 5/8/03

INITIALS

7.0 PROCEDURE STEPS (Continued)

NOTE:

1) Section 7.6 for CS Loop II venting is not required to be performed if venting was performed by this procedure, 2-01-75, or 2-SR-3.5.1.1(CS II) within the previous 24 hours, and no activities have occurred to invalidate the venting. If this is applicable: Section 7.6 can be **N/A** with performance of the applicable 2-SR-3.5.1.1(CS II), 2-OI-75, or 2-SR-3.5.1.6(CS II) documented in the post-test remarks.

2) 2-HS-75-72 and 2-SHV-75-92 are located on the north wall of Elevation 593 of the Unit 2 Reactor Building near column lines R11-P.

7.6 **VERIFY** CS LOOP II discharge piping is vented as follows:

7.6.1 **OPEN** CS SYSTEM II HP VENT TELL-TALE FLOW SOLENOID VALVE 2-FSV-75-72 by depressing and holding push-button 2-HS-75-92. ______

7.6.2 **SLOWLY** OPEN CS SYSTEM II TELL-TALE VENT SHUTOFF VALVE %-SHV-75-72. ______

7.6.3 **VISUALLY VERIFY** a steady stream of water exiting the drain side of CS SYSTEM II HP VENT TELL-TALE FLOW SOLENOID VALVE 2-FSV-75-72 after venting four minutes. ______ (AC)

7.6.4 **CLOSE** CS SYSTEM II HP VENT TELL-TALE FLOW SOLENOID VALVE 2-FSV-75-72 by releasing push-button 2-HS-75-72 ______

7.6.5 **CLOSE** CS SYSTEM II TELL-TALE VENT SHUTOFF VALVE 2-SH . 75-72 ______

7.6.6 VERIFY CS LOOP II discharge piping static pressure is greater than 39 psig as indicated by CORE SPRAY SYS II DISCH PRESS Indicator 2-PI-75-49 on Panel 2-9-3. ______ (AC)

Date 5/8/03

INITIALS

7 0 PROCEDURE STEPS (Continued)

NOTE:

The RHR SEVICE WATER PUMP RUNNING annunciators (Reference Attachment 5) will alarm in steps 7 7 1 through 7 7 4 for those pumps not already running and will collectively be verified in step 7 7 5

7.7 **PLACE** RHRSW pumps in operation as fellows to prevent an RHRSW automatic start:

7.7.1 **IF** RHRSW Pump 83 is capable of being started AND is NOT operating, **THEN**

START WHRSW Pump B3 using hand-switch 0-HS-23-88A/2 (Othenvise MIA.) ______

7 7 2 **IF** RHRSW Pump B3 is capable of being started AND is NOT operating. **THEN**

START RHRSW Pump D3 using hand-switch 0-HS-23-94A/2 (Otherwise N/A) ______

Date 5/6/03

INITIALS

7.0 PROCEDURE STEPS (Continued)

7.7.3 **IF** RHRSW Pump A1:

(1) Is capable of being started.

AND

(2) Is lined up to supply EECW.

AND

(3) Is NOT operating, THEN

START RHRSW Pump A1 using hand-switch 0-HS-23-1A/2. (Otherwise **N/A**.) ______

7.7.4 **IF** RHRSW Pump C1

(1) Is capable of being started.

AND

(2) Is lined up to supply EECW

(3) Is NOT operating. **THEN**

START RHRSW Pump C1 using hand-switch 0-HS-23-8A/2. (Otherwise **N/A**.) ______

7.7.5 VERIFY and **ACKNOWLEDGE** alarms received on Panel 0-9-23 ______

Date 5/8/03

INITIALS

7.0 PROCEDURE STEPS (Continued)

CAUTION

The static pretest suction pressure must be 1 psig before a **CS** pump can be operated

NOTES:

(1) Pressure indicators are located on instrument Rack 2-LPNL-25-60 which is in the NE Corner Room at Elevation 519'.

(2) Ensure Pressure Gauges A and B are mounted at the same elevation as 2-PI-75-32 and 2-PI-75-47 when readings are taken.

7.8 **VERIFY** CS Pump 2B and 2D pretest static suction pressures are adequate and **RECORD** below.

Parameter/Indicator	Measured	Required
CS Pump 2B Suction Pressure 2-PI-75-32	7 psig	≥ 1 psig
CS Pump 2D Suction Pressure 2-PI-75-41	7 psig	≥ 1 psig
Pressure Gauge A	7 psig	≥ 1 psig
Pressure Gauge B	7 psig	≥ 1 psig

Date 5/3/05

INITIALS

7.0 PROCEDURE STEPS (Continued)

7.9 **VERIFY** Core Spray NE Room Cooler Fan is OFF by observing fan motor power on light at 480V RMOV Board 2B, Compartment 8B, Electrical Board Room 2B (El. 593') is extinguished. ______

7.10 **CHECK** Core Spray NE Room Cooler Fan is NOT operating by noting no air flow from the duct louvers above CS Pump 2B and 2D can be felt while standing next to each pump. ______

7.11 **START** CS Pump 2B using hand-switch 2-HS-75-33A ______

7.12 **CHECK** the following annunciators on Panels 2-9-3 and 0-9-23-8 are in alarm

- CORE SPRAY SYS II PUMP B START (2-XA-55-3F, window 1) ______
- RHR OR CS PUMPS RUNNING ADS BLOWDOWN PERMISSIVE (2-XA-55-3C, window 10) ______
- CORE SPRAY PUMP 2B RUNNING (0-XA-55-23C, window 37) ______

NOTE:

Indication of the required flow in the following step demonstrates 2-CKV-75-537B is open and 2-CKV-75-537D and 2-CKV-75-570D are closed sufficiently to allow CS Pump 2B to operate at rated conditions.

7.13 **THROTTLE** 2-FCV-75-50 using CORE SPRAY SYS II TEST VALVE. 2-HS-75-50A to obtain an average CS LOOP II flow of 3200 gpm as indicated by 2-FI-75-49. [BFPER 98-0004734-000] ______ AC)

Date ~~5/8/03~~

INITIALS

7 0 PROCEDURE STEPS (Continued)

NOTES:

(1) Satisfactory completion of step 7.14 verifies 2-CKV-75-570B has opened sufficiently to perform its intended function.

(2) Step 7.14 *may* be repeated as required if data acquisition steps are incomplete due to timing/coordination problems while observing gauges and indicating lights.

7.14 **PERFORM** the following to verify 2-CW-75-578B operation:

7.14.Z **THROTTLE** 2-FCV-75-50 using CORE SPRAY SYS II TEST VALVE, 2-HS-75-50A to obtain a CS LOOP II flow of approximately 1800 gprn as ind cated by 2-FI-75-49 ______

7.14.2 **VERIFY** COKE SPRAY SYS II MIN FLOW VALVE 2-FCV-75-37 opens ______ (AC)

7.14.3 **THROTTLE** 2-FCV-95-50 using CORE SPRAY SYS II TEST VALVE 2-HS-75-50A to obtain a CS LOOP II flow of approximately 2800 gpm as indicated by 2-FI-75-49 ______

7.14.4 **VERIFY** CORE SPRAY SYS II MIN FLOW VALVE, 2-FCV-75-37 is closed ______ (AC)

7.14.5 **THROTTLE** 2-FCV-75-58 using CORE SPRAY SYS II TEST VALVE. 2-HS-75-50A to obtain a CS LOOP II flow of approximately 3200 gpm as indicated by 2-FI-75-49. ______

Date 5/8/03

INITIALS

7.0 PROCEDURE STEPS (Continued)

7.14.6 **RECORD** below CS Pump 2B discharge pressure measured locally by 2-PI-75-35 on 2-LPNL-25-60:

CS Pump 2B Diseh Press ________ psig

7.14.7 **NOTIFY** Operations personnel to monitor CS Pump 28 discharge pressure measured locally by 2-PI-75-35 on 2-LPNL-25-60 for minimum reading when CORE SPRAY SYS II MIN FLOW VALVE 2-FCV-75-37 reaches open position

7.14.8 **CONTINUOUSLY HOLD** the CORE SPRAY SYS II MIN FLOW VALVE 2-FCV-75-37, 2-HS-75-37A in the OPEN position until step 7.14.10.

7.14.9 **RECORD** below the lowest CS Pump 2B discharge pressure measured locally by 2-PI-75-35 on 2-LPNL-25-60

CS Pump 28 Disch Press ________ psig

7.14.10 **RELEASE** hand-switch 2-HS-75-37A to the AUTO position

Date 5/8/03

INITIALS

7.0 PROCEDURE STEPS (Continued)

7.14.11 **CALCULATE** the change in CS Pump 2B discharge pressure as stipulated below:

Initial Discharge Pressure		300 psig (Step 7.14.6)	
Lowest Discharge Pressure	-	170 psig (Step 7.14.9)	
Discharge Pressure Change	=	130 ~~psid~~	

NOTE:

Verification the discharge pressure change meets the acceptance criteria stipulated in the following step provides positive confirmation 2-CKV-75-570B ias opened sufficiently to perform its intended design function.

7.14.12 **VERIFY** the discharge pressure change recorded in step 7.14.11 is greater than or equal to 10 psid. ______ (AC)

7.14.13 **CHECK** CORE SPRAY SYS II MIN FLOW VALVE 2-FCV-75-39 is closed by noting valve position indicating lights above hand-switch 2-HS-75-37A. ______

Date 5/8/03

INITIALS

7 0 PF EDURE STEPS (Continued)

NOTES:

(1) Pressure indicators 2-PI-75-32 and 2-PI-75-35 are located in the NE Corner Room at Elevation 519' on local instrument Rack 2-LPNL-25-60

(2) The centerline of Pressure Gauge A must be at the same elevation as 2-PI-75-32! centerline when readings are taker,.

(3) CS LOOP II flow rate data may be obtained by using the ICS (e.g., ICS console display, Panel 2-9-5 ICS digital display) or by manual means using a DVOM to locally monitor Flow Modifier 2-FM-75-49 input. If the ICS is used, the average flow rate shall be determired by a visual estimate of the average of the ICS display. A numerical calculation is not required.

(4) ASME OM Code data collection requires the CS pumps be operated at a stable reference flow rate when discharge pressure readings are taken. While the flow rate may be an average value, the Unit Operator (UO) should maintain the flow rate as close as possible to the required 3200 gum (349 mV) flow rate.

7 15 **PERFORM** the following ASME OM Code pump flow and pressure measurements for CS Pump 28 operation

7.15.1 ***IF*** the ICS is available to obtain CS LOOP II flow rate data, THEN

PERFORM the following, (Otherwise N/A)

7.15.1.1 **CHECK** no gross instrument channel failures have occurred by noting the ICS-displayed flow rate is within 100 gpm of the flow rate shown on CORE SPRAY SYS II FLOW Indicator 2-FI-75-49. ____

7 15 1 2 THROTTLE CORE SPRAY *SYS* II PEST VALVE 2-FCV-75-50 using hand-switch 2-HS-75-50A to obtain an average ICS display reading of 3200 gpm [BFPER98-004734-000] ____

Date 5/8/02

INITIALS

7.0 PROCEDURE STEPS (Continued)

7.15.2 **IF** the ICS is NOT available to obtain CS LOOP II flow rate data, ***THEN***

THROTTLE 2-FCV-75-50 using CORE SPRAY SYS II TEST VALVE, 2-HS-75-50A to obtain an average reading of 349 mV at the DVOM installed at Panel 2-9-19. (Otherwise N/A.) [BFPER98-004734-000] N/A

7.15.3 **AFTER** stable conditions are obtained, **THEN**

RECORD CS Pump 2B suction pressure from Pressure Gauge A:

CS Pump 2B suction pressure (M&TE) -1 psig

7.15.4 **RECORD** the pressure reading at 2-PI-75-35 below:

CS Pump 2B discharge pressure 247 psig

7.15.5 **CALCULATE** CS Pump 2B differential pressure as follows and **VERIFY** the differential pressure meets the acceptance criteria:

Discharge Pressure	247 psig (Step 7.15.4)
Suction Pressure	-1 psig (Step 7.15.3)
Differential Pressure	= [illegible] psid
Acceptance Criteria	229 to 279.8 psid

(AC)

Date 4/8/07

INITIALS

7.0 PROCEDURE STEPS (Continued)

7.15.6 **RECORD** the following data for CS Pump 2B:

Parameter/Indicator	Measured	Acceptance Criteria
Core Spray Sys II Flow, or ICS Display	3[illegible] gpm (average)	3200 gpm (average)
Core Spray Sys II Disch Pressure 2-PI-75-48 (Panel 2-9-3)	[illegible]	N/A
Core Spray Sys II Flow 2-FM-75-49 (Panel 2-9-19) (Flow Transmitter Input)	[illegible] mV (average) (Note 1)	349 mV (±0.5 mV) (average)
Core Spray pump 2B Motor Current 2-EI-75-33 (Panel 2-9-3)	[illegible] amps	N/A
4kV Shutdown Bd C Voltage (Panel 9-23-8)	4[illegible] VAC	N/A

NOTE:

1) N/A reading for 2-FM-75-49, CS SYS II FLOW on step 7.15.6 if DVOM on step 7.5.5 was NOT installed

Date 5/8/03

INITIALS

7.0 PROCEDURE STEPS (Continued)

NOTE:

Verify vibration readings obtained in the next step are not in ALARM status on the portable M&TE before shutting down the CS Pump. This will preclude having to restart the pump in case of questionable data. The ALARM limits are NOT acceptance criteria, but are used to flag questionable data.

7.16 [QMDS] **NOTIFY** EM to perform 0-TI-230 vibration measurements as indicated on Attachment 4 for CS Pump 28.

7.17 **RECORD** CS Pump 28 vibration readings below:

VIBS POINT	MEASURED VALUE
AA	inisec
AH1	in/sec
AH2	in/sec
BH	in/sec
CH1	in/sec
CH2	in/sec

EM

7.18 **VERIFY** the Core Spray NE Room Cooler Fan is **ON** by

OBSERVING fan motor power on light at 480V RMOV Board 28, Compartment 8B, Electrical Board Room 2E (El 593') is illuminated

7.19 **CHECK** the Core Spray NE Room Cooler Fan is operating by:

NOTING air flow from the duct louvers above CS Pump 2B and 2D can be felt while standing next to each pump.

Date 5/1/03

INITIALS

7.0 PROCEDURE STEPS (Continued)

7.20 **START** CS Pump 2D using hand-switch 2-HS-75-42A. ML

7.21 **CHECK** the following annunciators on Panels 2-9-3 and 0-9-23-8 are in alarm:

CORE SPKAY SYS II PUMP D START (2-XA-55-3F, window 2).

e CORE SPRAY PUMP 2D RUNNING (0-XA-55-23D, window 37)

7.22 **THROTTLE** 2-FCV-75-50 using CORE SPRAY SYS II TEST VALVE, 2-HS-75-50A as necessary to obtain a CS LOOP II flow of 6250 to 6350 gpm as indicated by 2-FI-75-49 or ICS display

7.23 **RECORD** the following CS LOOP II parameters and **VERIFY** parameter values are within acceptance band

Parameter/Indicator	Minimum	Measured	Maximum
Care Spray Sys II Flow 2-FI-75-49 (Panel 2-9-3) or ICS	6250 gpm	63__ gpm	6350 gpm
Core Spray Sys II Disch Pressure 2-PI-75-48 (Panel 2-9-3)	228 psig	28_ psig	N/A

(AC)

7.24 **FULLY** OPEN 2-FCV-75-50 using CORE SPRAY SYS II TEST VALVE 2-HS-75-50A.

Date ____________

INITIALS

7 0 PR EDURE STEPS (Continued)

NOTES:

REFER TO Illustration 1 Process for Stroke Timing Valves per the ASME OM Code

7 25 **CLOSE** and **TIME** 2-FCV-75-50 using CORE SPRAY SYS II TEST VALVE, 2-HS-75-50A and RECORD the stroke time below

2-FCV-75-50 Closure Time (Seconds)		
Normal	Measured	Maximum
23.0 to 30.0		30.0

7.25.1 **VERIFY** the time recorded is within the maximum value listed ___ (AC)

7.26 **IF** the stroke time measured in step 7 25 is within the maximum value listed and outside the normal range, THEN

PERFORM the following; (Othenvise N/A this section) [BFPER971386]

7 26 1 **OPEN** the 2-FCV-95-50 using CORE SPRAY SYS II TEST VALVE, 2-HS-75-50A ______

7 26 2 **CLOSE** and **TIME** 2-FCV-75-50 using CORE SPRAY SYS II TEST VALVE 2-HS-75-50A and **RECQRD** the restroke time on Attachment 7 [BFPER971386] ______

7.26.3 **VERIFY** the restroke time recorded on Attachment 7 is within the maximum values iisted. [BFPER971386] ___ (AC)

7.27 **CHECK** CORE SPRAY SYS II MIN FLOW VALVE 2-FCV-75-37 is open by noting valve position indicating lights above hand-switch 2-HS-75-37A. ______

Date ______________

INITIALS

7 0 PROCEDURE STEPS (Continued)

NOTE:

The following steps requires the operator to hold until conditions have stabilized as determined by Engineering or Components Organization for obtaining Temperature profiling and Thermography.

7.28 **IF** Temperature profiling/Thermography is being performed on the 28 Core Spray Pump. **THEN**

PERFORM the following: (Otherwise N/A this section.)

7.28.1 **THROTTLE** 2-FCV-75-50 using CORE SPRAY SYS II TEST VALVE, 2-HS-75-50A as necessary to obtain a CS LOOP II flow of 6250 to 6350 gpin as indicated by 2-FI-75-49 or ICS display

7.28.2 **HOLD** until Engineering/ Components group obtains steady state Temp profile/Thermography for the 2B Core Spray pump bearings/components. ______

7.28.3 **WHEN** Notified by Engineering/ Components group that steady state Temp profile/Thermography data is obtained for the 28 Core Spray pump, **THEN**

CONTINUE with this procedure ______

7.28.4 **CLOSE** -FCV-75-50 using COKE SPRAY SYS II TEST VALVE. 2-HS-75-50A ______

7.28.5 **CHECK** CORE SPRAY SYS II MIN FLOW VALVE 2-FCV-75-37 is open by noting valve position indicating lights *above* hand-switch 2-HS-75-37A. ______

Date ________________

~~INITIALS~~

7 0 PROCEDURE STEPS (Continued)

7 29 **STOP** CS Pump 28 using hand-switch 2-HS-75-33A. ______

7.30 **RESET** and **CHECK** the following annunciators on Panels 2-9-3 and 0-9-23-8 are reset:

- CORE SPRAY SYS II PUMP B START (2-XA-55-3F, window 1). ______
- CORE SPRAY PUMP 2B RUNNING (0-XA-55-23C, window 37). ______

NOTE:

Indication of required flow in the following step demonstrates 2-CKV-75-537D is open and 2-CKV-75-537B and 2-CKV-75-570B are closed sufficiently to allow CS Pump 2D to operate at rated conditions

7.31 **THROTTLE** 2-FCV-75-53 using CORE SPRAY SYS II TEST VALVE, 2-HS-75-50A to obtain a CS LOOP II flow of 3200 gpm as indicated by 2-FI-75-49. ______ (ACj

Date ________________

INITIALS

7 0 PROCEDURE STEPS (Continued)

NOTES:

(1) Pressure indicators 2-PI-75-41 and 2-PI-75-44 are located in the NE Corner Room at elevation 519 *on* local instrument Rack 2-LPNL-25-60

(2) The centerline of Pressure Gauge B must be at the same elevation as 2-PI-75-41 centerline when readings are taken.

(3) ASME OM Code data collection requires the CS pumps be operated at a stable reference flow rate when discharge pressure readings are taken. While the flow rate may be an average value, the Unit Operator (UO) should maintain the flow rate as close as possible to the required 3200 gpm (349 mV) flow rate.

7.32 **PERFORM** the following ASME OM Code pump flow and pressure measurements for CS Pump 2D operation:

7 32 1 **IF** the ICS is available to obtain CS LOOP II flow rate data, **THEN**

THROTTLE: 2-FCV-75-50 using CORE SPRAY SYS II TEST VALVE, 2-HS-75-50A to obtain an average ICS display reading cf 3200 gpm. (Otherwise N/A) [BFPER98-004734-000] ______

7.32.2 **IF** the ICS is NOT available to obtain CS LOOP II flow rate data, **THEN**

THROTTLE 2-FCV-75-50 using CORE SPRAY SYS II PEST VALVE 2-HS-75-50A to obtain an average reading of 349 mV (±0 5 mV) at the DVOM installed at Panel 2-9-19 (Otherwise N/A) [BFPER98-004734-000]

Date ________________

INITIALS

7 0 PROCEDURE STEPS (Continued)

7.32.3 **AFTER** stable conditions are obtained, **THEN**

RECORD CS Pump 2B suction pressure from Pressure Gauge B:

CS Pump 2D suction pressure (M&TE) ________ psig ________

7.32.4 **RECORD** the pressure reading at 2-Pi-75-44 below:

CS Pump 2D discharge pressure ________ psig ________

7 32 5 **CALCULATE** CS Pump 2D differential pressure as follows and **VERIFY** the differential pressure meets the acceptance criteria

Discharge Pressure	________ psig (Step 7.32.4)
Suction Pressure	________ psig (Step 7.32.3)
Differential Pressure	= ________ psid
Acceptance Criteria	220.6 to 269.6 psid

________ (AC)

Bate ________________

INITIALS

7 0 PR(:DURE STEPS (Continued)

7 32.6 **RECORD** the following data for CS Pump 2D:

Parameter/Indicator	Measured	Acceptance Criteria
Core Spray Sys II Flow, or ICS Display	_______ gpm (average)	3200 gpm (average)
Core Spray Sys II Disch Pressure 2-PI-75-48 (Panel 2-9-3)	_______ psig	N/A
Core Spray Sys II Flow 2-FM-75-49 (Panel 2-9-19) (Flow Transmitter Input)	_______ mV (average) Note 1	349 mV (±0.5 mV) (average)
Core Spray Pump 2D Motor Current 2-EI-75-42 (Panel 2 - 9 3	_______ amps	N/A
4kV Shutdown Bd D Voltage (Panel 9-23-8)	_______ VAC	N/A

____

NOTE:

1) N/A reading for 2-FM-75-49, CS SYS II FLOW on step 7 32 6 if DVOM on step 7 5 5 was NOT installed

Date ________________

INITIALS

7.0 PROCEDURE STEPS (Continued)

NOTE:

Verify vibration readings obtained in the next step are not in ALARM status on the portable M&TE before shutting down the CS Pump. This will preclude having to restart the pump in case of questionable data. The ALARM limits are NOT acceptance criteria, but are used to flag questionable data.

7.33 [QMDS] **NOTIFY** EM to perform 0-TIL230 vibration data measurements as indicated on Attachment 4 for CS Pump 2D ______

7.34 **RECORD** CS Pump 2D vibration *reading* below

VIBS POINT	MEASURED VALUE
AA	in/sec
AH1	in/sec
AH2	in/sec
BH	in/sec
CH1	in/sec
CH2	in/sec

______
EM

Bate ________________

INITIALS

7.0 PROCEDURE STEPS (Continued)

NOTES:

(1) Satisfactory completion *of* step 7.35 verifies 2-CKV-75-570D has opened sufficiently to perform its intended function.

(2) Step 7.35 may be repeated as required if data acquisition steps are incomplete due to timing/coordination problems while observing gauges and indicating lights.

7.35 **PERFORM** the following to verify 2-CW-75-570D operation:

7.35.1 **RECORD** below the CS Pump 2D discharge pressure rneasured locally using 2-PI-75-44 on 2-LPNL-25-60

CS Pump 2D Disch Press __________ psig ________

7 35 2 **NOTIFY** Operations personnel to monitor CS Pump 2D discharge pressure measured locally rising 2-PI-75-44 on 2-LPNL-25-60 for minimum reading when CORE SPRAY SYS II MIN FLOW VALVE 2-FCV-75-37 reaches open position. ________

7.35.3 CONTINUOUSLY **HOLD** the CORE SPRAY SYS II MIN FLOW VALVE, 2-US-75-37A in the OPEN position until step 7 35 5 ________

7.35.4 **RECORD** below the lowest CS Pump 2D discharge pressure measured locally by 2-PI-75-44 on 2-LPNL-25-60:

CS Pump 2D Disch Press _________ psig

7.35.5 **RELEASE** hand-switch 2-HS-75-37A to the AUTO position ________

Date ________________

INITIALS

7.0 PROCEDURE STEPS (Continued)

7.35.6 **CALCULATE** the change in CS Pump 2B discharge pressure as stipulated below:

Initial Discharge Pressure		___________psig (Step 7.35.1)	
Lowest Discharge Pressure	-	___________psig (Step 7.35.4)	
Discharge Pressure Change	=	___________psid	_______

NOTE:

Verification the discharge pressure change meets the acceptance criteria stipulated in the following step provides positive confirmation 2-CKV-75-570D has opened sufficiently to perform its intended design function.

7.35.7 **VERIFY** the discharge pressure change recorded is greater than or equal to 10 psid. _______ (AC)

7.35.8 **CHECK** CORE SPRAY SYS II MIN FLOW VALVE 2-FCV-75-37 is closed by noting valve position indicating lights above hand-switch 2-HS-75-37A. _______

Date ________________

INITIALS

7.0 PROCEDURE STEPS (Continued)

NOTE:

The following steps requires the operator to hold until conditions have stabilized as determined by Engineering or Components Organization for obtaining Temperature profiling and Thermography.

7.36 **IF** Temperature profiling/Thermography is being performed on the 2D Core Spray Pump, **THEN**

PERFORM the following: (Otherwise N/A this section.)

7 36 1 **THROTTLE** 2-FCV-75-50 using CORE SPRAY SYS II PEST VALVE, 2-HS-75-50A as necessary to obtain a CS LOOP II flow of 3200 gpm as indicated by 2-FI-75-49 or ICS display

7 36 2 **HOLD** until Engineering/ Components group obtains steady state Temp profile/Thermography for the 2D Core Spray pump bearings/components. ______

7 36 3 **WHEN** Notified by Engineering/ Components group that steady state Temp profile/Thermography data is obtained for the 2B Core Spray pump, **THEN**

CONTINUE with this procedure ______

Date ________________

INITIALS

7.0 PR(:DURE STEPS (Continued)

7.37 **THROTTLE CLOSE** 2-FCV-75-50 using CORE SPRAY SYS II TEST VALVE, 2-HS-75-50A. ________

7.38 **CHECK** the Core Spray NE Room Cooler is operating by noting air flow from the duct louvers above CS Pump 2B and 2D can be felt while standing next to each pump ________ (AC)

7.39 **STOP** CS Pump 2D using hand-switch 2-HS-75-42A. ________

7.40 **CHECK** CORE SPRAY SYS II MIN FLOW VALVE 2-FCV-75-37 is open by noting valve position indicating lights above 2-HS-75-37A ________

7.41 **VERIFY** the Core Spray NE Room Cooler Fan is OFF by observing fan motor on light at 480V RMOV Board 28, Comoartment 8B, Electrical Board Room 2B (El 593') is extinguished ________

7.42 **CHECK** the Core Spray NE Room Cooler has stopped operating by noting no air flow from the duct louvers above CS Pump 2B and 2D can be felt while standing next to each pump ________

7.43 **RESET** and **CHECK** the following annunciators on Panel 2-9-3 and 0-9-23-8 are reset:

- CQRE SPRAY SYS II PUMP D START (2-XA-55-3F, window 2). ________
- RHR OR CS PUMPS RUNNING ADS BLOWDOWN PERMISSIVE (2-XA-55-3C, window IO) ________
- CORE SPRAY PUMP 2D RUNNING (0-XA-55-23D, window 37) ________

Date ________________

INITIALS

7.0 PROCEDURE STEPS (Continued)

NOTE:

The NE Corner Room cooling fan will automatically start if area temperature is greater than 90 degrees Fahrenheit

7 44 **RETERMINATE** the power lead (Cable 2ES3308-II, Wire No 8BX) at Terminal 2 in JB 8790 for 2-TS-64-73 and **REINSTALL** JB cover

______ 1st

______ 2nd

7.45 **COMPLETE** applicable environrnentai qualification work record entries *for* terminal block and cable.

______ IM

7.46 **COMPLETE** applicable environmental qualification work record entries for CS pump motor vibration measurements

______ EM

NOTE:

The following step shall be performed using skill of craft.

7.47 REMOVE Pressure Gauges A and B installed at test tees on Instrument Rack 2-LPNL-25-60 and **RETURN** instrument sense lines to their as-found condition per Attachment 6, Section B

IM

7.48 IF a DVOM was used to collect CS Loop II flow rate data, THEN

REMOVE the DVOM and associated test probe leads installed at Panel 2-9-19 on Flow Modifier 2-FM-75-49 and RETURN the flow modifier to its as-found condition; Otherwise N/A.

IM

7.49 **RETURN** RHRSW Pumps A1, C1, E3 and D3 to their pretest alignment in accordance with 0-OI-67 where applicable (Reference step 7 7)

______

Date ________________

INITIALS

7 0 PROCEDURE STEPS (Continued)

NOTES:

(1) Lifting of thermostat power lead enabled testing of NE Corner Room cooling fan operation due to CS pump operation regardless of room temperature. This operation disabled the area fan cooler thermostat 2-SI-4.2.B-60FT(II) must be run on the thermostat to return it to an operable status.

(2) [NRC/C] When the CS pump Room cooler thermostat was disabled, an LCO on the thermostat may have been entered (TR Table 3.3.3.2-1). If the CS pump room cooler thermostat was disabled for greater than 24 hours, a CS System LCO may have resulted. Step 7.5 lifted the lead and recorded the time. If the time the thermostat was disabled approaches *24* hours, the performer shall notify the Unit Supervisor immediately. [NCO 89-0216-002]

7.50 **REQUEST** IM to perform portions of 2-SI-4.2 B-60FT(II) which are applicable to 2-TS-64-73 functional testing ________

7.51 [NRC/C] **VERIFY** portions of 2-SI-4.2.B-60FT(II) which 2re applicable to 2-TS-64-73 functional testing are successfully completed.

AND

RECORD the time of completion below: [NCO 89-0216-0021

Time ______________

IM

Date ___________ ____

INITIALS

7.0 PROCEDURE STEPS (Continued)

NOTES:

(1) Section 7.52 substeps may be performed in any order.

(2) If a deficiency(s) is identified during performance of the independent verifications in the following step, the independent verifier shall stop and notify the Unit Supervisor immediately for further instructions prior to correcting the deficient condition(s).

7.52 **PERFORM** the following independent verifications to ensure CS LOOP II has been returned to its pretest configuration:

7.52.1 **VERIFY** CS Pump 2B is in a standby configuration **by** checking the green indicating light is illuminated white, and red indicating lights above 2-HS-75-33A, and the amber CS PUMP AUTO-INIT LQCKQUT LIGHT (PUMP 2B 2-IL-75-33), are extinguished on Panel 2-9-3. ____ IV

7.52.2 **VERIFY** CS Pump 2 8 is in a standby configuration by checking the green indicating light is illuminated white. and red indicating lights above 2-HS-75-42A, and the amber CS PUMP AUTO-INIT LOCKOUT LIGHT (PUMP 2D 2-IL-75-42), are extinguished on Panel 2-9-3. ____ IV

7.52.3 **VERIFY** CORE SPRAY SYS II TEST VALVE 2-FCV-75-50 is closed using valve position indicating lights above 2-HS-75-50A on Panel 2-9-3. ____ IV

7.52.4 **VERIFY** CORE SPRAY *SYS* II MIN FLOW VALVE 2-FCV-75-37 is open using valve position indicating lights above 2-HS-75-37A on Panel 2-9-3. ____ IV

Date ____________

INITIALS

7.0 PROCEDURE STEP'S (Continued)

NOTE:

Calculation **IV** consists of verifying arithmetic for accuracy and arithmetic inputs have been properly transferred between steps within this procedure. IV is NOT required to verify local pressure readings were correctly recorded.

7.52.5 **PERFORM** the following:

7.52.5.1 **VERIFY** calculation performed in step 7.14.11 is correct. ________ IV

7.52.5.2 **VERIFY** calculation performed in step 7.15.5 is correct. ________ IV

7.52.5.3 VERIFY calculation performed in step 7.32.5 is correct. ________ IV

7.52.5.4 **VERIFY** calculation performed in step 7.35.6 is correct. ________ IV

NOTE:

The following step is performed outside the main control room at the CS LOOP II vent station which is located on the north wall *of* Elevation 593 of the Reactor Building approximately 8 feet east of column lines R11-P.

7 52 6 VERIFY CS SYSTEM II TELL-TALE VENT SHUTOFF VALVE 2-SHV-75-72 is closed ________ IV

Date ________________

INITIALS

7.0 PROCEDURE STEPS (Continued)

NOTE:

The following step is performed by IMs in the NE Corner Room at Elevation 519'

7.52.7 **INDEPENDENTLY** VERIFY the following

7.52.7.1 **VERIFY OPEN** 2-PISV-075-0032 PANEL ISOL VALVE TO 2-PI-075-0032 ________ IV

7.52.7.2 **VERIFY** installed, test tee cap for 2-PI-075-0832 ________ IV

7.52.7.3 **VERIFY OPEN,** 2-PISV-075-0041. PANEL ISOL VALVE TO 2-PI-075-0041. ________ IV

7.52.7.4 VERIFY installed, test tee cap for 2-PI-075-0041 ________ IV

NOTE:

The following step is performed by IMs in the Unit 2 Auxiliary Instrument Room.

7.52.8 IF a DVOM was used to collect CS LOOP II flow rate data, **THEN**

VERIFY DVOM and associated test probe leads installed at Panel 2-9-19 on Flow Modifier 2-FM-75-49 have been removed and flow modifier has been returned to its inservice condition; Otherwise N/A. ________ IV

Bate ____ __________

INITIALS

7 0 PROCEDURE STEPS (Continued)

NOTES:

(1) Steps 7.52.9 through 7.52.15 are NOT independent verifications but are performed by the independent verifier to facilitate procedure performance efficiency.

(2) Steps 7.52.9 through 7.52.14 verify the closing spring for each CS and RHRSW pump operated by this procedure is properly charged upon termination of test activities.

(3) The UO should be contacted to determine which RHRSW pumps were operated. If there exists a question as to whether an RHRSW pump has been operated during performance of the procedure then the breaker charging spring status shall be verified to be conservative.

7.52.9 At 4160V Shutdown Board C, Compartment 7, in Electrical Board Room 2A.

VISUALLY VERIFY CORE SPRAY PUMP 2B BKR CHARGED amber light is illuminated and closing spring target indicates CHARGED [II-B-92-068] ______

7 52 10 At 4160V Shutdown Board D Compartment 8, in Electrical Board Room 26

VISUALLY VERIFY CORE SPRAY PUMP 2D BKR CHARGED amber light is illuminated and closing spring target indicates CHARGED [II-B-92-068] ______

7.52.11 **IF** RHRSW Pump A I was operated by this procedure. THEN

At 4160V Shutdown Board A, Compartment 10, in Electrical Board Room 1A.

VISUALLY VERIFY RHR SERVICE WATER PUMP A I BKR CHARGED amber light is illuminated and closing spring target indicates CHARGED; Othenvise N/A. [II-B-92-068] ______

Date ________________

INITIALS

7.0 PROCEDURE STEPS (Continued)

7.52.12 **IF** RHRSW Pump C1 was operated by this procedure, **THEN**

At 4160V Shutdown Board B, Compartment 10, in Electrical Board Room 1B.

VISUALLY VERIFY RHR SERVICE WATER PUMP C1 BKR CHARGED amber light is illuminated and closing spring target indicates CHARGED, Otherwise N/A [II-B-92-068] ________

7.52.13 **IF** RHRSW Pump B3 was operated by this procedure, **THEN**

At 4160V Shutdown Board C, Compartment 9, in Electrical Board Room 2A.

VISUALLY VERIFY RHR SERVICE WATER PUMP B3 BKR CHARGED amber light is illuminated and closing spring target indicates CHARGED; Otherwise N/A. [II-B-92-068] ________

7.52.14 **IF** RHRSW Pump D3 was operated by this procedure. **THEN**

At 4160V Shutdown Board D, Compartment 10, in Electrical Board Room 2B.

VISUALLY VERIFY RHR SERVICE WATER PUMP D3 BKR CHARGED amber light is illuminated and closing spring target indicates CHARGED; Otherwise N/A. [II-B-92-068] ________

7.52.15 **RESET** and **VERIFY** annunciators

- CORE SPRAY PUMP 2B RUNNING (0-XA-55-41C, window 37)
- CORE SPRAY PUMP 2D RUNNING (0-XA-55-41D, window 37)
- Any applicable RHR SERVICE WATER PUMP RUNNING at Central Diesel Information Center Panel 25-41 are reset ________

Bate ________________

INITIALS

7 0 PROCEDURE STEPS (Continued)

7.53 **COMPLETE** Attachment 1, Surveillance Procedure Review Form, up to Unit Supervisor review. ________

7.54 **IF** any valves required restroking (i.e. stroke times were recorded on Attachment 7), **THEN**

PROVIDE a copy of Attachment 7 to the Duty Maintenance Manager to deliver to the ASME IST Program owner, (Otherwise N/A.). [BFPER971386] ________

7.55 **NOTIFY** Unit 1, Unit 2, and Unit 3 Unit Operators this procedure is complete. ________

7.56 **NOTIFY** Unit Supervisor this procedure is complete

8.0 ILLUSTRATIONS/ATTACHMENTS

Attachment 1, Surveillance Procedure Review Form.

Attachment 2, ASME OM Code Inservice Testing Review Form.

Attachment 3, Flow Modifier 2-FM-75-49 Front View.

Attachment 4, Vibration Data Point Locations

Attachment 5, Annunciators Affected by Surveillance Procedure Performance

Attachment 6, Temporary ASME OM Code Pressure Indicators Installation and Removal

Attachment 7, ASME OM Code Restroke Time Record Form

Illustration 1, Process for Stroke Timing Valves Per the ASME OM Code

END OF TEXT

ATTACHMENT I
(Page 1 of 2)

SURVEILLANCE PROCEDURE REVIEW FORM

REASON FOR TEST: DATE/TIME STARTED 5/1/07 0630

_______Scheduled Surveillance DATE/TIME COMPLETED________

✓ System inoperable (Explain in Remarks) PLANT CONDITIONS MODE 1

✓ Maintenance (WO No. 03-6630-01)

_______Other (Explain in Remarks)

PRE-TEST REMARKS: Performed for PMT of 2-FCV-75-50 after maintenance [illegible] 5/1 1-7

PERFORMED BY:

Initials	Name (Print)		Name (Signature)
RL	Richard [illegible]	(Test Dir/Lead Perf)	[illegible]
RCL	Randall Kehoe	(Test Dir/Lead Perf)	[illegible]

Delays or Problems (If yes, explain in POST-TEST REMARKS)? ____Yes ____No

Acceptance Criteria Satisfied? ____Yes ____No

If the above answer is no, the Unit Supervisor shall determine if an LCO exists. LCO ____Yes ____No

UNIT SUPERVISOR________________ Date________

INDEPENDENT REVIEWER (SRO)________________ Date________

SCHEDULING COORDINATOR________________ Date________

POST-TEST REMARKS:________________

ATTACHMENT 1
(Page 2 of 2)

Surveillance Procedure Review Form
(Continuation Page)

PERFORMED BY:

Initials	Name (Print)	Name (Signature)

POST-TEST REMARKS (Continued):

ATTACHMENT 2
(Page 1 of 2)

ASME OM CODE INSERVICE TESTING REVIEW FORM

COMPONENT TESTED	STEP	ACCEPTABLE RANGE	NOT ACCEPTABLE	N/A OR NOT TESTED
CS Pump 2B				
Differential Pressure	step 7.15.5	☐	☐	☐
CS Pump 2D				
Differential Pressure	step 7.32.5			

ATTACHMENT 2
(Page 2 of 2)

ASME OM CODE INSERVICE TESTING REVIEW FORM

COMPONENT TESTED	STEP	FULLY ACCEPTABLE	NOT ACCEPTABLE	N/A OR NOT TESTED
2-FCV-95-50	step 7 25	☐	☐	☐
2-CKV-75-537B	step 7.13 and step 7.31	☐	☐	☐
2-CKV-75-537D	step 7.13 and step 7 31	☐	☐	☐
2-CKV-75-570B	step 7.14.12 and step 7 31		☐	☐
2-CKV-75-570D	step 7 13 and step 7 35.7			

NOTE:

Any valve with a stroke time exceeding it5 maximum value shall be declared inoperable. Any valve having a stroke time outside of its normal stroke time range shall be restroked immediately. If the restroke time is also outside the normal range, then the recorded stroke times shall be evaluated by engineering within 96 hours of the completion of this procedure for acceptability.

Date Received: ____________________

ASME OM CODE REVIEWER (Components) _ DATE ________

ASME OM Code Data entered in SI(s) 2-SI-3 1 1 and 2-SI-3 2 1

ANII REVIEWER - ________________________ DATE ________

REMARKS: ____

____

____

____

____

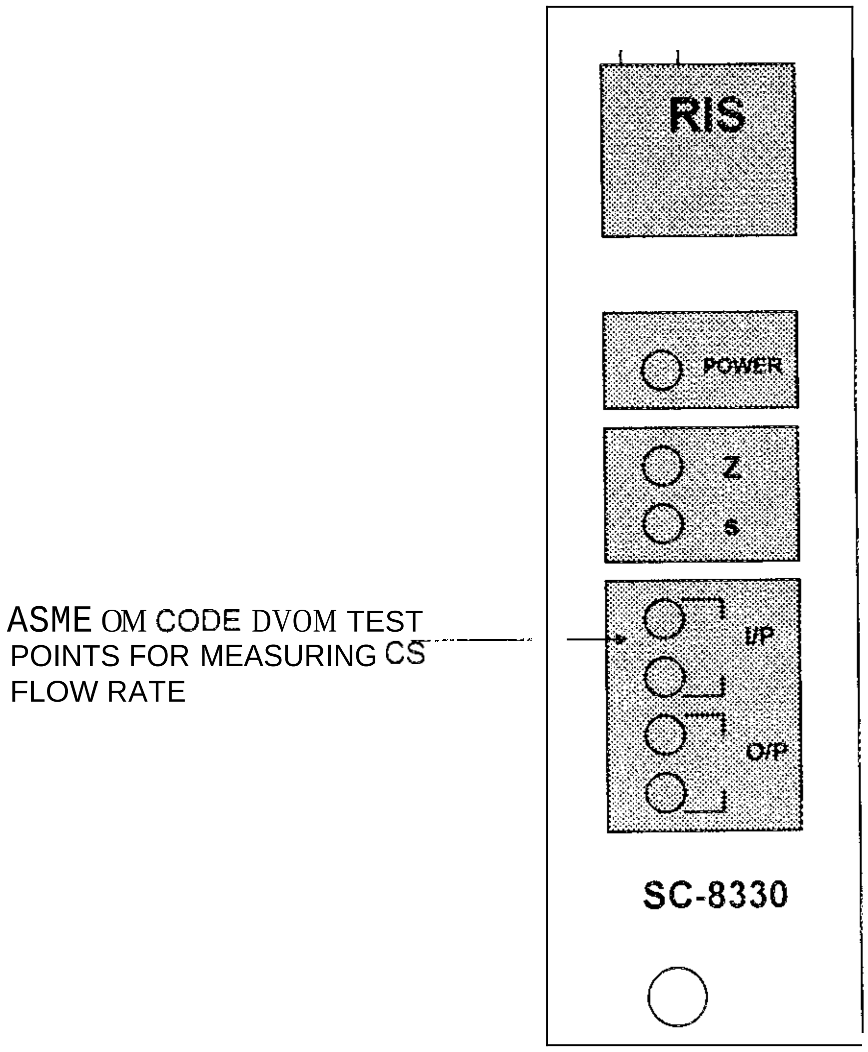
ATTACHMENT 3
(Page 1 of 1)

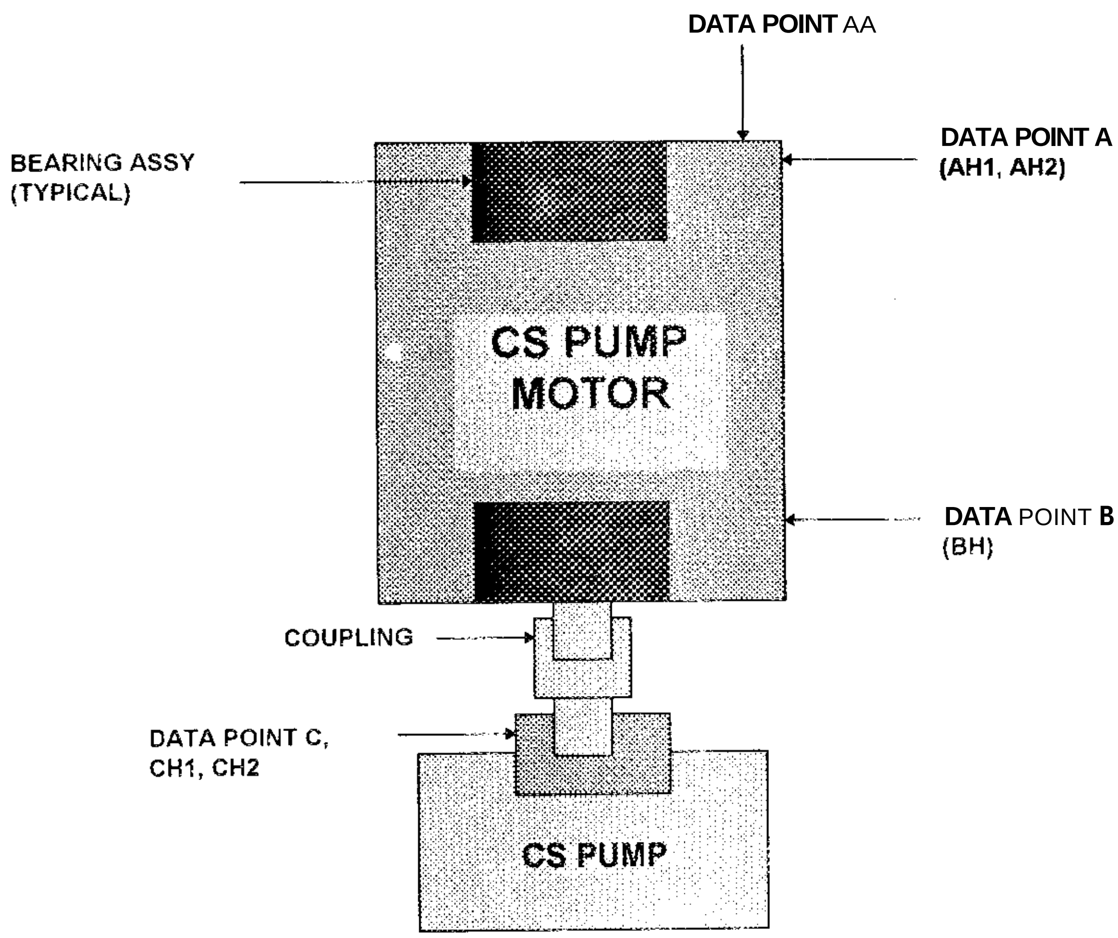
FLOW MODIFIER 2-FM-75-49 FRONT VIEW

NOTES:

(1) Connect (+) DVOM probe lead to I/P (+) flow modifier test jack and (-) DVOM test probe lead to I/P (-) flow modifier test jack per the diagram shown below.

(2) Set DVOM to 2 V DC voltage range and observe voltage readings for any signs of anomaly. Voltage readings should be 248 mV plus or minus process noise voltage variations which should NOT exceed +/- 1 mV when *a* no flow condition is present.




DATA POINT AA
BEARING ASSY
(TYPICAL)
DATA POINT A
(AH1, AH2)
CS PUMP
MOTOR
DATA POINT B
(BH)
COUPLING
DATA POINT C,
CH1, CH2
CS PUMP

ATTACHMENT 5
(Page 1 of 1)

ANNUNCIATORS AFFECTED BY SURVEILLANCE PROCEDURE PERFORMANCE

UNID	DESCRIPTION	LOCATION
2-9-3	RHR OR CS PUMPS RUNNING ABS BLOWDOWN PERMISSIVE	2-XA-55-3C Window 10
2-9-3	CORE SPRAY SYS II PUMP B START	2-XA-55-3F Window 4
2-9-3	CORE SPRAY SYS II PUMP D START	2-XA-55-3F Window 2
2-9-3	CORE SPRAY SYS II DISCH PRESS HIGH	2-XA-55-3F Window 30
0-9-23-7	RHR SERVICE WATER PUMP A1 RUNNING	0-XA-55-23A Window 34
0-9-23-7	RHR SERVICE WATER PUMP C1 RUNNING	0-XA-55-23B Window 34
0-9-23-8	RHR SERVICE WATER PUMP B3 RUNNING	0-XA-55-23C Window 34
0-9-23-8	CORE SPRAY PUMP 2B RUNNING	0-XA-55-23C Window 37
0-9-23-8	RHR SERVICE WATER PUMP D3 RUNNING	0-XA-55-23D Window 34
0-9-23-8	CORE SPRAY PUMP 2D RUNNING	0-XA-55-23D Window 37
0-25-41	RHR SERVICE WATER PUMP A1 RUNNING	0-XA-55-41A Window 34
0-25-41	RHR SERVICE WATER PUMP C1 RUNNING	0-XA-55-41B Window 34
0-25-41	CORE SPRAY PUMP 2B RUNNING	0-XA-55-41B Window37
0-25-41	RHR SERVICE WATER PUMP B3 RUNNING	0-XA-55-41C Window 34
0-25-41	RHR SERVICE WATER PUMP D3 RUNNING	0-XA-55-41D Window 34
0-25-41	CORE SPRAY PUMP 2D RUNNING	0-XA-55-41D Window 37

This attachment provides the UO with a listing of main control room and local alarms that will be affected by performance of this procedure, This attachment is for information only.

ATTACHMENT 6
(Page 1 of 2)

TEMPORARY **ASME** OM CODE PRESSURE INDICATORS INSTALLATION AND REMOVAL

Bate 5/8/03

INITIALS

A. PERFORM the following steps to install the temporary pressure gauges described by step 5.2.2 at Instrument Rack 2-LPNL-25-0068:

A.1 CLOSE 2-PISV-075-0032 PANEL ISOL VALVE TO 2-PI-75-32 ______

A.2 **REMOVE** 2-PI-075-0032 test tee cap below gauge ______

A.3 CONNECT temporary pressure gauge at 2-Pi-075-0032 test tee ______

A.4 **OPEN** 2-PISV-095-0032 PANEL ISOL VALVE TQ 2-PI-75-32 ana VENT temporary gauge connection as required ______

A.5 CLOSE 2-PISV-075-0041 PANEL ISOL VALVE TO 2-PI-75-41 ______

A.6 REMOVE 2-PI-095-0041 test tee cap located below gauge ______

A.7 CONNECT temporary pressure gauge at 2-PI-075-0841 test tee ______

A.8 OPEN 2-PIS-075-0041 PANEL ISOL VALVE TO 2-PI-75-47 and VENT temporary gauge connection as required ______

ATTACHMENT 6
(Page 2 of 2)

TEMPORARY ASME OM CODE PRESSURE INDICATORS INSTALLATION AND REMOVAL

Date ________________

INITIALS

B. **PERFORM** the following steps to remove the temporary pressure gauges installed at Instrument Rack 2-LPNL-25-0060:

B.1 **CLOSE** 2-PISV-0754032 PANEL ISOL VALVE TO 2-PI-75-32 ________

B.2 DISCONNECT temporary pressure gauge at 2-PI-075-0032 test tee ________

B.3 **INSTALL** 2-PI-075-0032 test lee cap ________

B.4 OPEN 2-PISV-075-0032 PANEL ISOL VALVE TO 2-PI-75-32 ________

B.5 **CLOSE** 2-PISV-095-0041 PANEL ISOL VALVE TO 2-PI-75-41 ________

B.6 DISCONNECT temporary pressure gauge at 2-PI-075-0041 test tee ________

B.7 INSTALL 2-PI-075-0041 test tee cap ________

B.8 OPEN 2-PISV-075-0041 PANEL ISOL VALVE TO 2-PI-75-47 ________

ATTACHMENT 7
(Page 1 of 1)

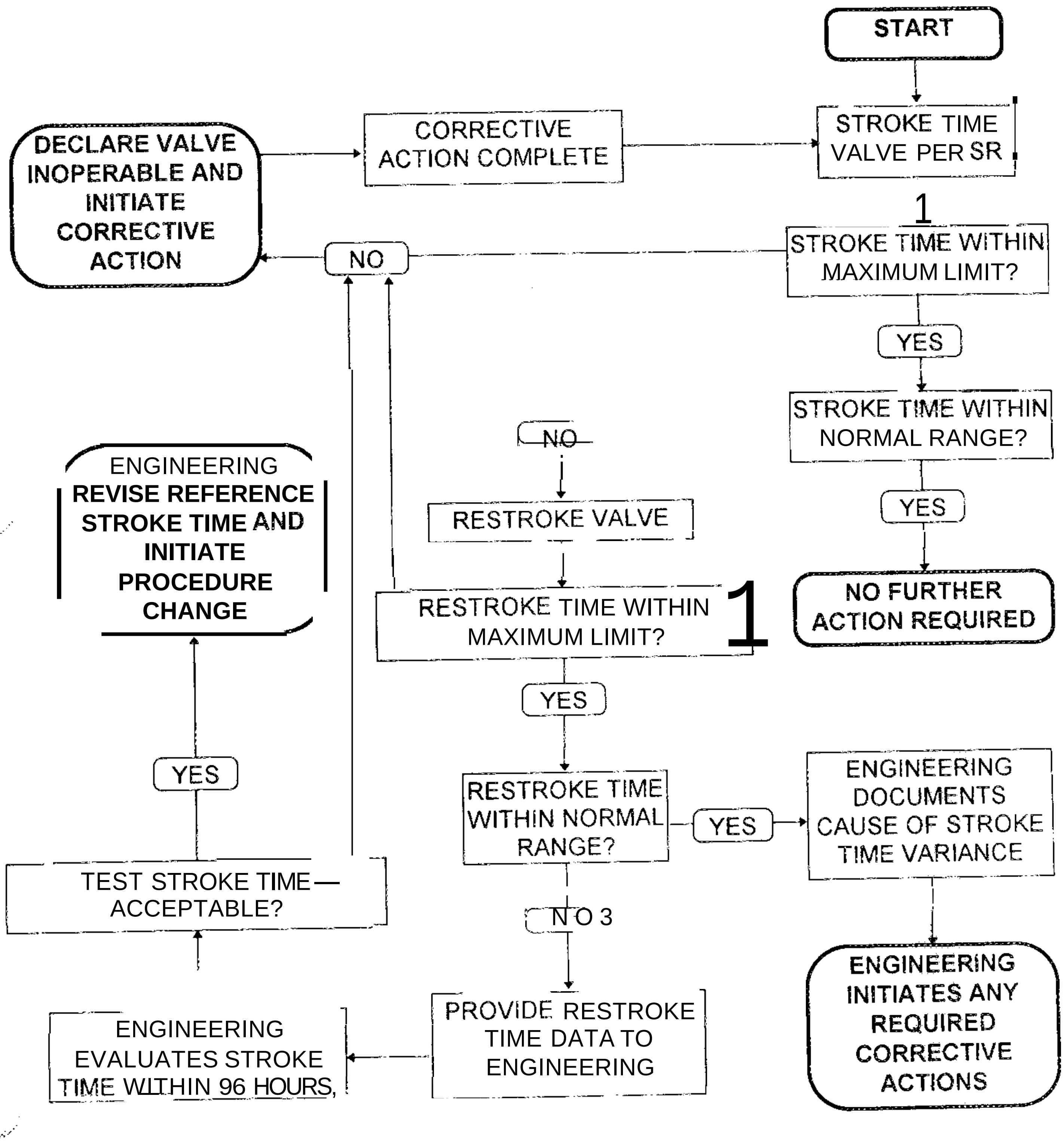
ASME OM CODE RESTROKE TIME RECORD FORM

VALVE UNID	NORMAL STROKE TIME (SEC)	MEASURED STROKE TIME (SEC)	MAXIMUM STROKE TIME (SEC)
2-FCV-75-5-) (CLOSE)	23.0 to 30.		
2-FCV-95-50 (CLOSE')	23.0 to 30.0		30.0'

ILLUSTRATION 1

(Page 1 of 1)

PROCESS FOR STROKE TIMING VALVES PER THE ASME OM CODE



BROWNS FERRY NUCLEAR PLANT
JOB PERFORMANCE MEASURE

REVISION *LOG*

Revision Number	Effective Date	Pages Affected	Description of Revision
0	9/9/02	*ALL*	NEW

BROWNS FERRY NUCLEAR PLANT
JOB PERFORMANCE MEASURE

OPERATOR:____

RO ____ SRO ____ BATE:___________

JPM NUMBER: A3.1R
TASK NUMBER: ADMIN
TASK TITLE: N/A
WA NUMBER: 2.3.10 K/A RATING: RO 2.9 SRO: 3.3

****

TASK STANDARD: REVIEW A RADIOLOGICAL SURVEY MAP TO DETERMINE IF A TASK CAN BE COMPLETED WITHOUT EXCEEDING AN INDIVIDUAL'S TVA EXPOSURE LIMITS.

LOCATION OF PERFORMANCE: SIMULATOR _ PLANT _ CONTROL ROOM

REFERENCES/PROCEDURES NEEDED: N/A

VALIDATION TIME: CONTROL ROOM:______LOCAL:

MAX. TIME ALLOWED: ____ (Completed for Time Critical JPMs only)

PERFORMANCE TIME: ____ CONTROL ROOM __ LOCAL

COMMENTS: ____

____

____

Additional comment sheets attached? YES __ NO

RESULTS: SATISFACTORY ___ UNSATISFACTORY

EXAMINER SIGNATURE:_________________________________ BATE.

EXAMINER

EXAMINER'S KEY (Page 1 of 2)

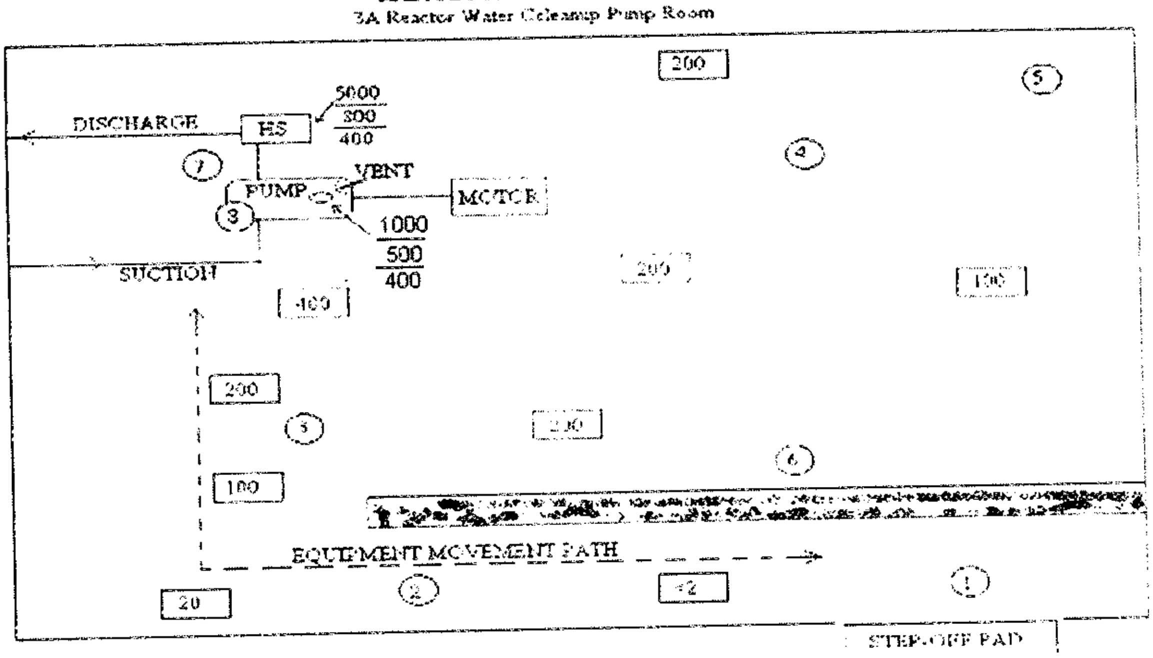
INITIAL CONDITIONS: You are the Supervisor of a Browns Ferry employee who has obtained an accumulative yearly radiation dose of 800 mrem.

INITIATING CUES: Given the attached radiological. survey map of 3A Reactor Water Cleanup Pump room, DETERMINE if your worker can complete the assigned task in the room without exceeding his/her TVA administrative yearly dose limit.

The job requires replacing the cleanup pump vent. The worker will have to transport welding leads and a cutting torch along the designated walkway (marked as EQUIPMENT MOVEMENT PATH), cut the existing vent valve off, weld in the new vent valve, and transport the equipment back to the C-zone step off pad. Transporting the required equipment will require approximately 3 minutes, each way. Once the pump area on the survey map is reached (marked as Pump Vent), it will require 30 minutes to replace the pump vent. The attached Rad?ological Survey Map contains the information you must interpret to successfully complete this JPM.

EXAMINER'S KEY

RADIOLOGICAL SURVEY MAP
3A Reactor Water Cleanup Pump Room



Smear #	Dpm/100 cm^2
1	<1000
2	<1000
3	5,000
4	<1000
[illegible]	<1000
[illegible]	<1000
7	<1000
8	10,000

LEGEND

CT
30 cm
GA

[GA] - General Area

[HS] - Hot Spot

(#) - Smear

RESPONSE:
The candidate already has 800 mrem for the year. Working in 400 mr/hr for thirty minutes would give the candidate an additional 200 mr for a total of 1000 mr for the year, or the TVA administrative limit. Add to that the time required for transporting tools and equipment and the worker would exceed the administrative limit which would mean that the worker would NOT be able to complete the task without exceeding his/her administrative limit.

EXAMINER'S KEY

****

These questions may be asked to clarify or expand on the required knowledge items at the discretion of the examiner.

****

What is the Yearly TVA Administrative Dose Limit?

1000 Mr

****

According to information on the survey map, Would the worker be required to wear protective clothing in this area?

Yes, this is a C-Zone

****

Where would the worker find the specified protective clothing requirements?

On the RWP

****

According to information on the survey map and radiological posting requirements, Is this area a Radiation —— or High Radiation Area?

High Radiation Area

****

When the worker is working on the Pump Vent, what is the general area dose rate?

408 mr/hr

****

Where in the room would the worker stay to avoid unnecessary exposure while waiting in the room. IE, Where is the Low Dose Waiting Area

Near the Step Off Pad.

****

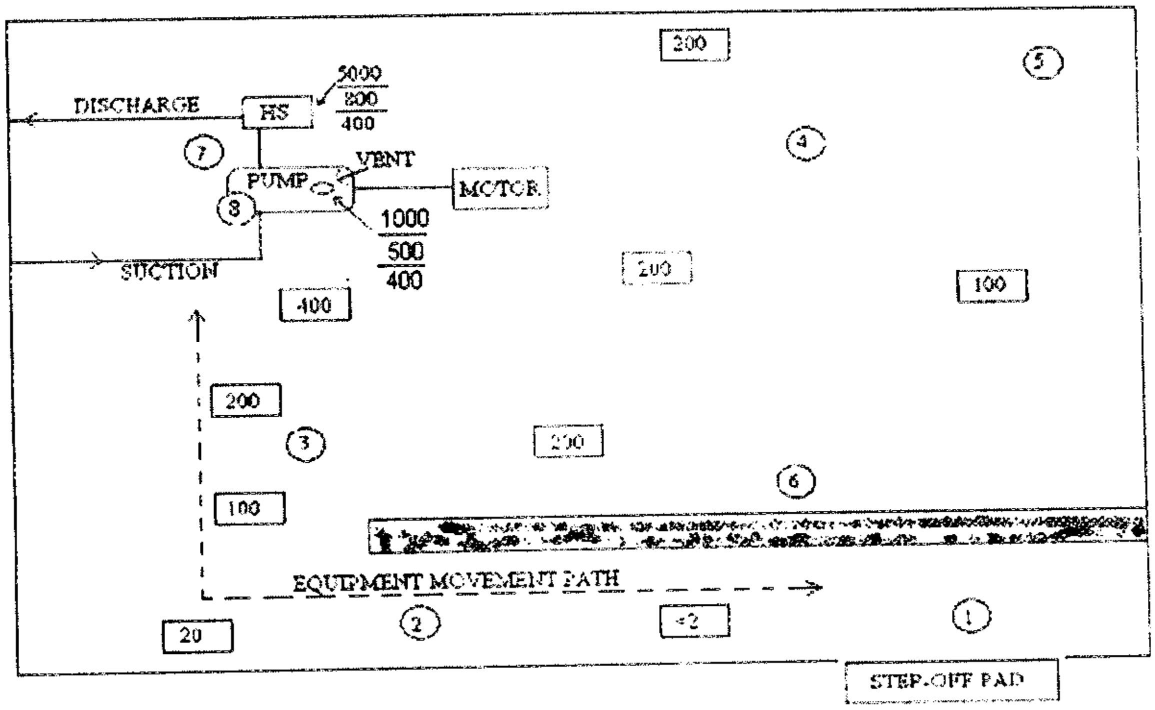
BROWNS FERRY NUCLEAR PLANT
JOB PERFORMANCE MEASURE

INITIAL CONDITIONS: You are a Supervisor of a Browns Ferry employee who has obtained an accumulative yearly radiation dose of 800 mrem.

INITIATING CUES: Given the attached radiological survey map of 3A Reactor Water Cleanup Pump room, DETERMINE if the worker can complete the assigned task in the room without exceeding his/her TVA administrative yearly dose limit..

The job requires replacing the cleanup pump vent. The worker will nave to transport welding leads and a cutting torch along the designated walkway (marked as EQUIPMENT MOVEMENT PATH), cut the existing vent valve off, weld in the new vent valva, and transport the equipment back to the C-zone step off pad. Transporting the required equipment will require approximately 8 minutes, each way. Once the pump area on the survey map is reached (marked as Pump Vent), it will require 30 minutes to replace the pump vent. The attached Radiological Survey Map contains the information you must interpret to successfully complete this JPM.

RADIOLOGICAL SURVEY MAP
3A Reactor Water Cleanup Pump Room



mear #	Dpm/100 cm^2
1	<1000
2	<1000
3	5.000
4	<1000
5	<1000
6	<1000
7	<1000
8	10,000

LEGEND

CT
30 cm
GA

GA - General Area

HS - Hot Spot

\# - Smear

A. RADCON (or **RSO**) shall investigate TLD and secondary dosimeter reading discrepancies.

B. Individuals shall inform RADCON (or RSO) if they lose their TLD or their secondary dosimeter is lost, damaged, exhibits an unexpected response or is off-scale.

C. NVLAP accredited TLD results are normally used as the official record **of** radiation exposure.

3.4.1.6 Administrative **Dose Levels***

Occupational radiation dose limits at TVAN facilities are consistent with the limits given in 10 CFR 20.

Occupational radiation dose limits at TVAN facilities are consistent with the limits given in 10 CFR 20.

Administrative dose levels (ADLs) to be used *as* guidelines for maintaining doses below regulatory limits have been established within TVAN and shall be observed for routine work. This program is not applicable to minors or declared pregnant women. The TVAN Administrative Dose Level Program is summarized in Table 1 below:

TABLE 1

ADMINISTRATIVE **DOSE LEVEL PROGRAM**

Dose Equivalent (REM)	Requirement	Authorization to exceed (signatures)
Up to **0.5 TEDE** (or 1.5 LDE or 5.0 SDE) at TVA	Statement of current year dose and previous years dose signed by individual	Not applicable
Up tu **1.0** TERE (or 3 0 LDE or 10 SDE) all sources	NRC FORM-4 or equivalent to document current year and prsvious years dose equivalent	Not applicable
To exceed **1.0 TEDE** (or 3.0 LDE or 10 SDE) all sources	Same as above	Site Radiologicaland Chemistry Control Manager/RSO
TO exceed **5.0 TEDE**[3] **all** sources	Form-4 information must be verified and a Planned Special Exposure initiated	Sie Radiologicaland Chemistry Control Manager/RSO, Plant Manager', and Site VP[2] or SED as appropriate
To exceed 1N[4] **all sources**	Form4 must be verified	Site Radiologicaland Chemistry Contrui Manager/RSO, Plant Manager[1], and Site VP[2] or SED as appropriate.

1. At non-nuclear plant sites, this will **be** the RSO's immediate supervisor.
2. At non-nuclear plant sites, this will be the applicable TVA VP.
3. Authorizations for a planned special exposure will only be considered in an exceptional situation when alternatives that might avoid the dose estimated to result from the planned special exposure are unavailable or impractical.
4. Total effective dose equivalent should not exceed 1N rem, where N equals the individual's age in years at last birthday. without the authorization signatures delineated.

BROWNS FERRY NUCLEAR PLANT
JOB PERFORMANCE MEASURE

JPM NUMBER: A4.1R

TITLE: CLASSIFY THE EVENT PER THE REP (LOSS OF SHUTDOWN COOLING)

TASK NUMBER: S-000-EM-21

SUBMITTED BY: ______________________ DATE: _____

VALIDATED BY: ______________________ DATE: _____

APPROVED: ______________________ DATE: _____
TRAINING

PLANT CONCURRENCE: ______________________ DATE: _____
OPERATIONS

- Examination **JPMs** Require Operations Training Manager or Designee Approval 2nd Plant Concurrence

BROWNS FERRY NUCLEAR PLANT
JOB PERFORMANCE MEASURE

REVISION LOG

Revision Number	Effective Bate	Pages Affected	Description of Revision
0			New

BRQWNS FERRY NUCLEAR. PLANT
JOB PEKFORMANCE MEASURE

OPERATOR:__________________________________

RO ____ SRO X DATE:______________

JPM NUMBER: A4.1R

TASK NUMBER: S-000-EM-21 (SRO ONLY)

TASK TITLE: CLASSIFY THE EVENT PER THE REP (LOSS OF SHUTDOWN COOLING)

K/A NUMBER: 2.4.38 WA RATING: RO 2.2 SRO: 4.0

****

TASK STANDARD: C lassify the event as an ALERT and. perform associated SRO actions.

LOCATION OF PERFORMANCE: SIMULATOR X PLANT _ CONTROL ROOM _

REFERENCES/PROCEDURES NEEDED: EPIP 1, REV ~~29~~ 30 RK 4-18-03; EPIP 3. REV ~~27~~ 29 RK 4-18-03

VALIDATION TIME: CONTROL ROOM: 25:00 LOCAL: N/A

MAX. TIME ALLOWED: ____ (Completed for Time Critical JPMs only)

PERFORMANCE TIME: _____CONTKOL ROOM __ LOCAL N/A

COMMENTS:____

Additional comment sheets attached? YES __ NO

RESULTS: SATISFACTORY ___ UNSATISFACTORY _____

SIGNATURE: ________________________ DATE: ___________________
EXAMINER

BROWNS FERRY NUCLEAR PLANT
JOB PERFORMANCE MEASURE

CANDIDATE HANDOUT

INITIAL CONDITIONS: You are the SHIFT MANAGER. Following a 400 day run, Unit 2 was in Mode 4 with the first pass of head bolt de-tensioning complete and was placed in Shutdown Cooling at 0800. At 1030 shutdown cooling isolated due to a failed pressure switch. 2-AOI-74-1 has been implemented. Unit 2 conditions are *as* foilows:

Reactor Power	Shutdown, All rods inserted
Reactor Level	+80 inches on Shutdown Vessel Flood Range
Reactor Pressure	3 psig
Moderator Temperature	215 Degrees Fahrenheit
DW Pressure	0 psig
DW Temperature	105 degrees F
DW Radiation	RR-90-256 Normal
DW Leakage Rate	None
Torus Temperature	88 degrees F
Torus Level	-3 inches
Torus Pressure	0 psig

NOTE: No release in progress
Wind speed is 5 rnph from the SW

INITIATING CUES: Determine the EVENT CLASSIFICATION and PERFORM THE REQUIRED actions until the emergency centers are staffed and you are relieved.

BROWNS FERRY NUCLEAR PLANT
JOB PERFORMANCE MEASURE

EXAMINER'S COPY

INITIAL CONDITIONS: You are the SHIFT MANAGER. Following a 400 day run, Unit 2 is in Mode **4** with the first pass of head bolt de-tensioning complete and was placed in Shutdown Cooling at 0800. At 1030 shutdown cooling isolated due to a failed pressure switch. 2- AOI-74-1 has been implemented. Unit 2 conditions are as follows:

Reactor Power	Shutdown, All rods inserted
Reactor Level	+80 inches OR Shutdown Vessel Flood Range
Reactor Pressure	3 psig
Moderator Temperature	215 Degrees Fahrenheit
DW Pressure	0 psig
DW Temperature	105 degrees F
DW Radiation	RR-90-256 Normal
DW Leakage Rate	None
Torus Temperature	88 degrees **F**
Torus Level	-3 inches
Torus Pressure	0 psig

NOTE: No release in progress
Wind speed is 5 mph from the SW

INITIATING CUES: Determine the EVENT CLASSIFICATION and PERFORM THE REQUIRED actions until the emergency centers are staffed and you are relieved.

START TIME:_______

****

Performance Step : Critical X Not Critical —

REFERS TO EPIP 1 to classify emergency event

Standard:

Candidate **REFERS TO** EPIP-1 Section II, 1.5A, LOSS OF DECAY HEAT REMOVAL and deciares an ALERT based on reactor moderator temperature CANNOT be maintained below 212° F whenever Technical Specifications require Mode **4** conditions or during operation in Mode 5.

SAT__UNSAT__N/A__ COMMENTS:____

____

****

Performance Step : Critical X Not Critical —

IMPLEMENTS EPIP-3 ALERT

Standard:

SHIFT MANAGER/SED **RECOGNIZES/IMPLEMENTS** an ALERT per EPIP-3.

SAT__UNSAT__N/A__ COMMENTS:____

____

3.0 INSTRUCTIONS

****

Performance Step : Critical ___ Not Critical X

3.1 **If** all Emergency centers **ARE STAFFED Then** notify the following that an **ALERT** Emergency Classification has been issued and EPIP 3 **is** being implemented, and continue in this procedure at Step 3.4. **If** all Emergency Centers **ARE NOT STAFED, Then** N/A this step and continue in this procedure.

CECC, TSC, OSC, CONTROL ROOMS, PLANT PA ANNOUNCEMENT

CUE: EMERGENCY CENTERS ARE NOT STAFFED.

Standard:

NA STEP 3.1 & continue in procedure

SAT__UNSAT__N/A__ COMMENTS:________________________________

____

****

3.2 Notification of the Operations Duty Specialist (ODS) & Ernergency Responders

Note: The ODS should be notified within *5* minutes after the emergency event is declared.

****

Performance Step : Critical X Not Critical —

3.2.1 **Complete** Attachment **A** (Initial Notification Form).

Standard:

ATTACHMENT A **is** completed as shown in Examiner's Copy, Page 7 of 10. **(INFORMATION GIVEN IN INITIAL CONDITIONS & INITIATING CUES EXCEPT EAL DESIGNATOR) NOTE: THIS IS GENERIC INFORMATION FOR DESCRIPTION OF EVENT--ALL THIS EXACT INFORMATION IS NOT REQUIRED FOR ACCEPTANCE UNDER BRIEF DESCRIPTION OF EVENT.**

SAT__UNSAT__N/A__ COMMENTS:____

____

Performance Step : Critical X Not Critical —

3.2.2 Activating Emergency Response Organization (ERO)

3.2.2.1 **if** ongoing/anticipated on-site security events may present a danger to the emergency responders, **Then** consult with Nuclear Security.

3.2.2.2 **If** ongoinglanticipated events present a danger to emergency responders, **Then** direct the Unit 1 Unit Operator to make notifications per Attachment B and select "Staging Area" as the option for the Emergency Paging System.

3.2.2.3 **If** there are no ongoing/anticipated danger to emergency responders, **Then** direct the Unit I Unit Operator to make notifications per Attachment B and select as applicable, "Brill" or "Emergency" as the option for the Emergency Paging System.

Standard:

DIRECTS Unit 1 Operator to make notifications per Attachment B (3.2.2.3).

SAT — U NSAT __ N/A __ COMMENTS: ______________________________

****
*

Performance Step : Critical X_ Not Critical —

3.2.3 **Notify** the ODS and **Provide** the information from Attachment A.

Note: Utilize the direct ring-down OBS phone when making this notification or as applicable dial direct.

OBS Telephone Numbers
5-1-751-1700, 2495

If the ODS cannot be reached within 10 minutes, **Then** contact the State of Alabama directiy by requesting the Rad Health Duty Officer at:

Day Shift 8 a.m.-5 p.m.	Holidays-Weekends-Offshifts
9-1-334-206-5391	9-1-334-242-4378

Standard:

Attempts Notification of the OBS.

CUE: The ODS Cannot be contacted by phone due to communications problems, no estimate on repair time.

Notifies the State of Alabama directly by one of the number listed above.

SAT__UNSAT__N/A__ COMMENTS:______________________________________

____

****

****

Performance Step : Critical — Not Critical X

3.2.4 **Fax** a copy of Attachment A to the ODS for confirmation of information or if the state is contacted directly.

ODS **Fax**	AL Rad Health Fax
5-751-8620	9-1-334-206-5387

CUE: FAXING TO THE STATE WILL BE SIMULATED.

Standard:

SIMULATES faxing **a** copy of Attachment A to the state.

SAT — UNSAT __ N/A __ COMMENTS: ______________________________

____

3.2.5 **Receive** confirmation call from the ODS (to verify notification of the State of Alabama)(NA this step, if the state **was** contacted directly).

CUE: [THREE MINUTES AFTER THE PERFORMER FAXES ATTACHMENT A]

STATE RAD HEALTH OFFICER CALLS AND CONFIRMS RECEIPT OF FAX.

3.3 NOTIFICATION OF SITE PERSONNEL

****

Performance Step : Critical______ Not Critical_X

3.3.1 **Make** the following P.A. announcement:

THIS IS (NAME), SHIFT MANAGER.
AN ALERT HAS BEEN DECLARED ON UNIT _2_. I HAVE ASSUMED THE DUTIES OF SITE EMERGENCY DIRECTOR. REPORT TO YOUR ASSIGNED EMERGENCY RESPONSE FACILITY **AT** THIS TIME!

Standard:

MAKES P. A. announcement as above.

SAT__UNSAT__N/A__ COMMENTS:____

____

****

CAUTION: Do not initiate Assembly and Accountability if:

1. A severe weather condition exists or projected'on-site, such *as* a Tornado.
2. An on-site security risk condition exists that may present a danger to site personnel during the assembly/accountability process.(Consult with Nuclear Security)

3.4 **ACCOUNTABILITY**

****

Performance Step: Critical______ Not Critical_X_____

3.4.1 **If** the emergency situation warrants an Assembly, Accountability, **Then** implement EPIP-8, Appendix C, concurrently with this procedure. (N/A STEP IF NOT APPLICABLE)

3.4.2 **If** the emergency situation does not warrant an Assembly, Accountability at this time, **Continue** to assess the situation, implementing EPIP-8 when necessary.

Standard:

ADDRESSES Accountability and at candidate's discretion may or may not implement the accountability section EPIP-8.

SAT_UNSAT_N/A_ COMMENTS:____

EXAMINERS NOTE: If candidate chooses NOT to initiate Assembly and Accountability then continue at Step 3.5, OFFSITE DOSE ASSESSMENT, page 20 (of this JPM).

SHIFT MANAGER/SITE EMERGENCY DIRECTOR -ASSEMBLY AND ACCOUNTABILITY ACTIONS

The following appendix shall be utilized by the Shift Manager/Site Emergency Director (SM/SED) or designee for the purpose of conducting site assembly and accountability actions.

****

Performance Step: Critical______ Not Critical_X_____

1. The SM/SED has determined that conditions require the activation of the assembly and accountability siren system and process.

Standard:

ENTERS initials and time.

SAT_UNSAT_N/A_ COMMENTS.

****
****

Performance Step: Critical______ Not Critical_X_

2. **NOTIFY...** Nuclear Security (NS) at extension 3238 or 2219 that:

   A. The assembly and accountability sirens will be activated immediately

   AND

   B. NS should implement EPIP-8, Appendix D.

Standard:

NOTIFIES NS assembly and accountability sirens will be activated and **DIRECTS** NS to implement EPIP 8, Appendix D.

SAT_UNSAT_N/A_ COMMENTS:

****

****

Performance Step: Critical______ Not Critical_X_____

3. **NOTIFY...** Radiological Control (RADCON) at extension 7865 that:

A. The assembly and accountability sirens will be activated immediately.

AND

B. RADCON should implement EPIP-8, Appendix E.

Standard:

NOTIFIES RADCON assembly and accountability sirens will be activated and **DIRECTS** RADCON to implement EPIP 8, Appendix E..

SAT__UNSAT__N/A__ COMMENTS:____________________________________

____

****

Performance Step: Critical______ Not Critical_X_____

4. **MAKE... a** public address announcement similar to:

"Attention all plant personnel, the site assembly and accountability process has been initiated. All personnel report immediately *to* your assigned assembly areas."

(REPEAT)

Standard:

MAKES P.A. announcement twice.

SAT__UNSAT__N/A__ COMMENTS:____________________________________

____

****

****

Performance Step: Critical_____ Not Critical_X_____

5. ACTIVATE... the assembly and accountability sirens.

Standard:

ACTIVATES the assembly and accountability sirens by DEPRESSING red button on 0-CNTL-244-6398 on Shift Manager's desk. (Critical) ENTERS initials and time. (Not Critical)

SAT_UNSAT_N/A_ COMMENTS:_________________________________

____

****

****

Performance Step: Critical_____ Not Critical_X_____

6. WHEN...the Assembly and Accountability Sirens have completed the 3-minute cycle and silenced.

MAKE... a PA announcement similar to:

"Attention all plant personnel, the site assembiy and accountability process has been initiated. All personnel report immediately to your assigned assembly areas."

(REPEAT)

Standard:

MAKES P.A. announcement twice.

SAT_UNSAT_N/A_ COMMENTS:_________________________________

____

****

NOTE

If at any time during the assembly and accountability process RADCON determines that radiation guidelines for an assembly area(s) have been exceeded, request NS to re-locate affected personnel to another assembly area or evacuate affected personnel off-site.

****

Performance Step: Critical_____ Not Critical **X**

7. **NOTIFY...** Central Emergency Control Center (CECC) Director either by the direct ring-down telephone in the TSC or at extension 751-1614.

OR

If the CECC Director can not be reached, notify the Operations Duty Specialist (ODS) at extension 751-1700 that:

A. The assembly and accountability sirens have been activated.

AND

B. BFN EPIP-8 is currently being implemented for assembly and accountability.

Standard:

NOTIFIES the Operations Duty Specialist (ODS) at extension 751-1700 that:

A. The assembly and accountability sirens have been activated.

AND

B. BFN €PIP-8 is currently being implemented for assembly and accountability.

SAT_UNSAT_N/A_ COMMENTS:________________________________

____

****

CUE: CECC HAS NOT YET BEEN STAFFED C [illegible] i ti [illegible] bl [illegible] ith the ODS have been corrected.

CUE: NUCLEAR SECURITY NOTIFIES YOU THAT ASSEMBLY AND ACCOUNTABILITY ARE SUCCESSFULLY COMPLETE.

****

Performance Step: Critical______ Not Critical_X_____

8. WHEN... Notified by NS that the assembly and accountability process has been completed.

THEN.... **MAKE** a public address announcement similar to:

"Attention all plant personnel, the site assembly and accountability process has been completed. All personnel remain in your assigned assembly areas."

(REPEAT)

Standard

MAKES P A. announcement twice

SAT_UNSAT_N/A_ COMMENTS:______________________________________

____

****

Performance Step: Critical______ Not Critical_X_____

9. **VERIFY** with the SM/SED, that conditions at this time require an order to evacuate all non-emergency response personnel from the Owner Controlled Area.

Standard:

ENTERS N/A

SAT_UNSAT_N/A_ COMMENTS:______________________________________

____

****

Performance Step: Critical______ Not Critical_X_____

10. IF... Conditions at this time, DO require an order to evacuate all non-emergency response personnel from the Owner Controlled Area.

THEN... Initiate Appendix F of this procedure (EPIP-8).

Standard:

Evacuation not required for an ALERT, candidate does not require evacuation of non-emergency response personnel.

SAT__UNSAT__N/A__ COMMENTS:____________________________________

11. IF... Conditions at this time, DO NOT require an order to evacuate all non-emergency response personnel from the Owner Controlled Area.

THEN..Exit this procedure. Re-enter this procedure at Appendix F when it has been determined by the SM/SED that conditions require an order to evacuate all non-emergency response personnel.

****

CUE: THE CECC IS STILL NOT STAFFED. THE COMMUNICATIONS PROBLEM WITH THE ODS HAVE BEEN CORRECTED.

****
**

3.5 OFFSITE DOSE ASSESSMENT

Performance Step : Critical — Not Critical X

3.5.1 Evaluate the need for offsite dose assessment
(N/A STEP IF NOT APPLICABLE)

CUE: CECC IS NOT OPERATIONAL.

3.5.1.1 When offsite dose assessment is required obtain the information from the CECC when operational.

3.5.1.2 If the CECC **is** not operational, contact the TSC, when staffed or the RADCON Shift Supervisor and request the implementation of EPIP 14, for dose assessment.

Standard:

SHIFT MANAGER/SED addresses the OFFSITE DOSE ASSESSMENT, may request activation of RABCON VANS in accordance with EPIP-14.

SAT__UNSAT__N/A__ COMMENTS:____

____

****

Performance Step : Critical X Not Critical —

3.6 NOTIFICATION OF THE NRC

3.6.1 **Notify** the **NRC** immediately or within 1 hour and if requested by the **NRC** maintain an open and continuous communications channel.

3.6.2

Note: **Utilize** the Emergency Notification System **(ENS)** when making this notification. Dial the first number listed on the sticker affixed to the **ENS** telephone, by dialing 9-1-"The Ten Digit Number listed on the ENS telephones". **IF** the number is **busy, THEN** select in order, the alternate numbers until a connection is achieved. No access codes required.

Standard:

NOTIFIES NRC on simulator red phone.

SAT— UNSAT__ N/A__ COMMENTS:______________________________

EXAMINER'S CUE: NRC does not require an open line at this time.

3.7 PERIODIC EVALUATION OF THE EVENT

CUE: THE EMERGENCY CENTERS *ARE* STAFFED AND THE PLANT MANAGER (SITE EMERGENCY DIRECTOR) IS HERE TO RELIEVE YOU.

END OF TASK

STOP TIME:___________

MSL/OFFGAS RADIATION	LOSS OF DECAY HEAT REMOVAL	
DESCRIPTION	DESCRIPTION	
1.4-U Valid MAIN **STEAM** LINE **RADIATION** HIGH-HIGH alarm, RA-90-135C **OR** Valid OG PRETREATMENT RADIATION HIGH alarm, RA-90-157A. OPERATING CONDITION: - Mode 1 - Mode 2 - Mode 3		**UNUSUAL EVENT**
	1.5A **Reactor** moderator temperature CANNOT be maintained below **212" I:** whenever Technical Specifications require Mode 4 conditions or during operations in Mode 5. OPERATING CONDITION: - Mode 4 - Mode 5	**ALERT**
	1.5-S Suppression Pool temperature, level and RPV pressure CANNOT **be** maintained in the safe area of Curve 1.5-S. OPERATING CONDITION: - Mode 1 - Mode 2 - Mode 3	**SITE EMERGENCY**
		GENERAL EMERGENCY

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT

EMERGENCY PLAN IMPLEMENTING PROCEDURE

EPIP-3

ALERT

REVISION 29

PREPARED BY T W CORNELIUS PHONE: 2038

RESPONSIBLE ORGANIZATION EMERGENCY PREPAREDNESS

APPROVED BY JEFF LEWIS DATE: 04/02/2003

EFFECTIVE DATE 04/07/2001

LEVEL OF USE: REFERENCE USE

QUALITY-RELATED

REVISION LOG

Procedure Number: EPIP-3 Revision Number: 29

Pages Affected: 3,5,6,8,11

Description of Change:

IC-30 This change is being conducted to incorporate the management of NRC Commitment changes as prescribed in the correspondence from site licensing RIMS R08000217713, and to human factor the notification and fullow-up notification forms.

Page 2 - change to step 3.2.1 involves human factoring the Notification Form Title.

Page 6 - changes involves removing the "NRC Commitment Brackets to step requiring the review of PORC and the human factoring of applicable steps.

Page **7** - change involves human factoring attachment title and modifying information to ensure consistency with NRC guidance.

Page 8 - change involves adding information regarding the support of the Unit 1 Operator in staffing the ERO.

Page 9 - Updated information for the Unit Operator to **use** during the ERO staffing process.

Page 10 - change involved adding a clarify statement concerning the appropriate use of the Follow-up Notification Form.

IC-3 1 EPIP-3, revision 26 is being issued to incorporate changes regarding assembly and accountability actions. All actions to initiate the accountability and evacuation processes are now located in EPIP-8. The revision additionally standardizes telephone numbers, and PORC reviews. This revision also adds clarification for the **actions** taken by the Unit 1 Unit Operator during their stifling of the ERO process.

Page 3 - added a statement to the caution information regarding security threat. Clarified steps 3.4.1 and 3.4.2 to implement EPIP-8 regarding actions to be taken for **assembly/accountability** and evacuation.

Page 6 - standardize Alert procedure closure information.

Page **8,9** - Clarify actions taken by the Unit 1 Unit Operator during the notification attachment.

IC-32 EPIP-3, revision ***21*** is being conducted to incorporate changes regarding actions to be taken when dangerous conditions exist on site that mould require ***the*** assembly of the ERO at the staging area. Additionally page 3 and 5 were revised to update telephone information regarding the Office of Radiation Control.

Page 2 - change instruct the SED when to direct the Unit 1 Unit Operator to assembly the ERO at the staging area.

Page **4** - revision adds clarification to the caution note regarding on-site security conditions for assembly/accountability.

Page 8 - revision adds option for staging area

IC-13 EPIP-3, revision is being conduct lo change the procedure reference for Dose Assessment from EPIP-14 to EPIP-13. Page 4 of **this** procedure is be revised.

IC-34 EPIP-3, rev. 29 is being condnct to standardize record retention (page 6) and revise the notification forms to include NRC Terminology from RIS 2002-16 for normal and abnormal releases (page 8 & 11). Additionally the revision will provide a place to document the time and EAL Designation when centers are staffed (page 2). Attachment section was renumbered (page ***5)***.

1.0 PURPOSE

1.1 Provide for timely notification of appropriate individuals or organizations when the Shift Manager/Site Emergency Director (SED) has determined by EPIP-I that an incident has occurred which is classified as an ALERT.

1.2 Provide for periodic evaluation of the current situation by the Shift Manager/SED to determine whether the ALERT should be terminated, continued, or upgraded to a more serious classification.

2.0 SCOPE

This procedure applies to emergency events that are classified as Alert by EPIP-1, Emergency Classification Procedure.

3.0 INSTRUCTIONS

Date: ___/ ___/ ___

3.1 **If all** Emergency Centers **ARE STAFFED, Notify** the following that a **"ALERT** Emergency Classification was declared at **Time:** ________, **EAL Designator** ________, and EPIP 3 is being implemented".

Then continue in this procedure at Step **3.4**

CECC	☐	Control Rooms	☐	INITIALS	TIME
TSC	☐	Plant PA Announcement	☐		
OSC	☐	This is NAME, Site Emergency Director. an Alert has been declared at BFN. we are currently implementing EPIP-3. Standby fur further update.			

If all Emergency Centers **ARE NOT STAFFED,**

Then N/A this step and continue in this procedure.

4.2 **Notification of the Operations Duty Specialist (ODS) & Emergency Responders**

Note: The ODS **should** be notified within 5 minutes after the emergency event is declared.

3.2.1 **Complete** Attachment A (Initial Notification Form). INITIALS TIME

3.2.2 **Activating Emergency Response Organization (ERO)**

3.2.2.1 **If** ongoinglanticipated on-site security events may present a danger to the emergency responders, INITIALS TIME

Then consult with Nuclear Security.

3.2.2.2 **If** ongoinglanticipated events present a danger to emergency responders, **Then** direct the Unit 1 Unit Operator to make notifications per Attachment B and select "Staging Area" **as** the option for the Emergency Paging System. INITIALS TIME

3.2.2.3 **If** there are no ongoing/anticipated danger to emergency responders, **Then** direct the Unit 1 Unit Operator to make notifications per Attachment B and select as applicable, "Drill" or "Emergency" as the option for the Emergency Paging System. INITIALS TIME

3.0 INSTRUCTIONS (CONTINUED)

3.2.3 **Notify** the ODS and **Provide** the information from Attachment A.

INITIALS TIME

Note: Utilize the direct ring-down ODS phone when making this notification or a5 applicable dial direct.

ODS Telephone Numbers - 5-751-1700, or 2495

If the ODS cannot be reached within 10 minutes, **Then** contact the State of Alabama directly by requesting the Office of Radiation Control at:

Day Shift 8 a.m. - 5 p.m. (Central)		Holidays-Weekends-Off-Shifts
Primary:	9-1-334-206-5391	Montgomery State Trooper Post
Backup:	9-1-800-582-1866	9-1-334-242-4378

3.2.4 **Fax** a copy of Attachment A to the ODS for confirmation of information or state if the state was contacted directly

INITIALS TIME

ODS Fax	Office of Radiation Control Fax
5-751-8620	9-1-334-206-5387

3.2.5 **Receive** confirmation call from the ODS (to verify notification of the State of Alabama)(NA this step, if the state was contacted directly).

INITIALS TIME

3.3 NOTIFICATION OF SITE PERSONNEL

3.3.1 **Make** the following plant P.A. announcement:

INITIALS TIME

THIS **IS** (*NAME*), SHIFT MANAGER. A ALERT **HAS** BEEN DECLARED ON UNIT ___ . I HAVE ASSUMED THE: DUTIES OF SITE EMERGENCY DIRECTOR. REPORT TO YOUR ASSIGNED EMERGENCY-RESPONSE FACILITY AT THIS TIME.

3.0 INSTRUCTIONS (CONTINUED)

CAUTION: Do not initiate Assembly and Accountability if:
1. **A** severe weather condition exist/projected on-site, such as a Tornado.
2. ***An*** on-site security **risk** condition exists that may present a danger to site personnel during the assembly/accountability process (Consult with Nuclear Security).

3.4 ACCOUNTABILITY

3.4.1 **If** the emergency situation warrants an Assembly, Accountability, **Then** implement **EPIP-8,** Appendix C, concurrently with this procedure. (N/A STEP IF NOT APPLICABLE)

_______ INITIALS _______ TIME

3.4.2 **If** the emergency situation does not warrant an Assembly, Accountability at this time, **Continue** to assess the situation, implementing EPIP-8 when necessary.

3.5 OFFSITE DOSE ASSESSMENT

3.5.1 Evaluate the **need** for offsite dose assessment. (N/A STEP IF NOT APPLICABLE)

_______ INITIALS _______ TIME

3.5.1.1 When offsite dose assessment is required obtain the information from the CECC when operational.

3.5.1.2 If the CECC is not operational, contact the TSC, when staffed or the RADCON Shift Supervisor and request the implementation of EPIP **13,** for dose assessment.

3.0 *INSTRUCTIONS (CONTINUED)*

3.6 <u>NOTIFICATION OF THE NRC</u>

3.6.1 **<u>Notify</u> the** NRC immediately or within 1 hour and if requested by the NRC maintain ***an*** open and continuous communications channel.

INITIALS ______ TIME ______

Note: **<u>Utilize</u>** the Emergency Notification System (ENS) when making this notification. Dial the first number listed on the sticker affixed to the ENS telephone, by dialing 9-1- "<u>The Ten Digit Number Listed on the ENS Teleohones</u>". **<u>If</u>** the number is busy, **<u>Then</u>** select in order, the alternate numbers until a connection is achieved. No access codes are required.

3.7 <u>PERIODIC EVALUATION OF THE EVENT</u>

3.7.1 Continue to **<u>Evaluate</u>** the event using EPIP-I as conditions warrant,

3.7.2 **<u>If</u>** plant conditions warrant the need for follow up information, **<u>Complete</u>** the Follow Up Notification Form, Attachment C.

Note: Conditions that warrant this evaluation are as a minimum when other EAL conditions exist indicating the current emergency classification or significant changes in plant conditions have occurred.

3.7.3 **<u>If</u>** the CECC is not staffed, **<u>Then</u>** notify the **ODS** and provide follow up information from the completed Attachment C form. Utilize the direct ring-down ODS phone when making this notification or as applicable dial direct.

ODS **-** 5-751-2495, **1700**

Note: **<u>If</u>** the ODS cannot be reached, **<u>Then</u>** contact the State of Alabama directly by requesting the Office of Radiation Control at:

<u>Day Shift **8** a.m. **-** 5 p.m. (Central Time)</u>	<u>Holidavs-Weekends-Off-shifts</u>
Primary: 9-1-334-206-539I	Montgomery State Trooper Post
Backup: 9-1-800-582-1866	9-1-334-242-4378

3.7.4 **<u>If</u>** the conditions warrant upgrading to a higher classification, ***<u>Then</u>*** initiate the appropriate EPIP.

3.8 INSTRUCTIONS (CONTINUED)

3.7.5 **If** the conditions warrants termination of the classifications, **Then** enter EPIP-16, Termination and Recovery Procedure.

3.7.6 **After** the evaluation has been completed, **.ifstaffed, Notify** the following of the status:

- CECC
- NRC (ENS)
- TSC
- OSC
- CONTROL ROOMS
- PLANT PA ANNOUNCEMENT

3.7.7 **Re-enter** this procedural section as conditions warrant at step 3.7.1 or until directed to exit this procedure by steps 3.7.4 or 3.7.5.

3.8 CLOSURE OF THE ALERT

3.8.1 Upon termination of the Notification of Alert, the Shift Manager shall send the completed EPIP-3 and all attachments to Emergency Preparedness (El').	INITIALS	TIME
3.8.2 Upon receipt of completed EPIP-3 and all attachments, Emergency Preparedness shall forward documents for the purpose of documentation storage.	INITIALS	TIME

4.8 RECORD RETENTION

4.1 RECORDS OF CLASSIFIED EMERGENCIES

The materials generated in support of key actions during an actual emergency classified as NOUE or higher are considered Lifetime retention Non-QA records. Materials shall be forwarded to the EP Manager who shall submit any records deemed necessary to demonstrate performance to the Corporate EP Manager for storage.

4.2 DRILL AM) EXERCISE RECORDS

The materials deemed necessary to demonstrate performance of key actions during drills are considered Non-QA records. These records shall be forwarded to the EP Manager who shall retain records deemed necessary to demonstrate six-year plan performance for six years. The EP Manager shall retain other records in this category for three years.

5.0 ATTACHMENTS

Attachment A - Initial Notification Form Alert

Attachment B - **Unit 1,** Unit Operator Notifications

Attachment *C* - Follow **Up** Information Form Alert

ATTACHMENT A (Page 1 of 1)
INITIAL NOTIFICATION FORM
ALERT

1. ☐ This is a Drill ☐ This is an Actual Event - Repeat - This is an Actual Event

2. This is ____________, **Browns Ferry** has declared *a* **ALERT**
affecting: ☐ Unit 1 ☐ Unit 2 ☐ Unit 3 ☐ Common

3. EAL Designator(s): ______________________

4. Brief Description of the Event:

____

____

____

____

5. **Radiological Conditions:** (Check one under both Airborne and Liquid column.)

Airborne Releases Offsite
☐ Minor releases within federally approved limits'
☐ Releases above federally approved limits'
☐ Release information not known
('Tech Specs)

Liquid Releases Offsite
☐ Minor releases within federally approved limits'
☐ Releases above federally approved limits'
☐ Release information not known
('Tech Specs)

6. **Event Declared:** Time: ______________ Date: ______________

7. Provide Protective Action Recommendation: ☐ None

8. Please repeat the information you have received to ensure accuracy.

9. Time and Bate this information *was* provided ____________/____________

Action: When completed, telecopy this information.

ATTACHMENT B (Page 1 of 2)

UNIT 1, UNIT OPERATOR NOTIFICATIONS

Date: ____/____/____

NOTES: **(1)** The Emergency Paging System (EPS) consists of a dedicated touch screen CRT. Activation of any screen feature requires the user place their fingertip within the boundary of the select button and leave it there for at least 1 second. The CRT Screen will normally display a large rectangle that indicates that the paging system is available but currently inactive.

(2) **If** the EPS fails to operate, contact the SM/SED immediately. Request that the ODS be contacted to initiate the system from his location. If the system fails to operate from the ODS area, then utilize the Weekly Duty List and Call-Out List to manually staff the Emergency Responders, implementing this attachment at step E.

1. Activation of the Emergency Paging System (EPS).

A. **PRESS** the EPS CRT Screen once to activate the paging options. ____________ INITIALS TIME

B. **PRESS** the appropriate option as instructed by the SED ____________ INITIALS TIME
- PAGERTEST
- DRILL
- EMERGENCY
- STAGING AREA
- ABORT

C. **PRESS** the START Button to initiate the option or **ABORT** to deny the option request. ____________ INITIALS TIME

D. **MONITOR** the Paging System Terminal Display ____________ INITIALS TIME

1. **IF**... **A** "NO" response is observed
OR
The position being paged has not responded within approximately 20 minutes

THEN... Utilize the Weekly Duty List and attempt to contact the position representative with available information. (No Fitness for Duty Question Required)

2. **IF...** The individual cannot be reached utilizing the Weekly Duty List

THEN... Utilize the Call-Out List and attempt to contact an alternate position representative. (Fitness for Duty Question Required)

ATTACHMENT B (Page 2 of 2)
UNIT 1, UNIT OPERATOR NOTIFICATIONS

Date: / /

E. Manual Call-Out (N/A step if EPS operates normally) — INITIALS TIME

1. Utilize the current Weelkv Dutv List and contact positions as listed.

2. If a position can not be reached from the current Weekly Duty list, then refer to the Call-out List as applicable to fill all vacant positions.

F. **CONTINUE** until all positions have been filled. — **INITIALS TIME**

2. Notify the Unit Supervisors on shift. — **INITIALS TIME**

3. Notify Nuclear Security Shift Supervisor and state *"AN* ALERT HAS BEEN DECLARED" and direct to activate EPIP-11, Security and Access Control. — **INITIALS TIME**

- Plant Extension 3150 or 2219

4. Notify the Chemistry Lab Supervisor and state *"AN* ALERT HAS BEEN DECLARED" and direct to implement 2/3-TI-331, Post Accident Sampling Procedure and CI-900 series, Analysis Procedures. — **INITIALS TIME**

- Plant Extension 2364 or 2368

5. Notify the RADCON Shift Supervisor and state *"AN* ALERT HAS BEEN DECLARED" and direct **to** activate EPIP-14, Radiological Control Procedure — **INITIALS TIME**

- Plant Extension **7865** or 3104

6. Notify the *"On-Call"* NRC Resident and state **"AN** ALERT HAS BEEN DECLARED." per BFN-EPIP-03 — **INITIALS TIME**

- Plant Extension 2572 [Secretary] or from weekly duty list

ATTACHMENT C (Page 1 of 1)
FOLLOW-UP INFORMATION FORM
ALERT

☐ THIS IS A REAL EVENT ☐ THIS IS A DRILL

Note: This form is for conducting Follow-up Information only.
This is ______________________.________ __ at Browns Ferry.
Name
There has been a Alert declared at Browns Ferry affecting:
☐ Unit 1 Unit 2 ☐ Unit 3 ☐ Common

The Reactor is ☐ Shutdown At Power

Plant Conditions are ☐ Stable ☐ Deteriorating

"Follow-Up" Information (e.g., Key Events, Status Changes)
____
____
____

Current Radiological Conditions are: (Check one under both Airborne and Liquid column.)

Airborne Release Offsite
☐ Minor releases within federally approved limits'
☐ Releases above federally approved limits'
☐ Release Information not known
('Tech Specs)

Liquid Releases Offsite
☐ Minor releases within federally approved limits'
☐ Releases above federally approved limits[1]
☐ Release Information not known
('Tech Specs)

Additional Rad information: (e.g., release duration)
____
____
____

☐ There is no Protective Action Recommendation at this time.

Please repeat the information you have received to ensure accuracy.

The time for this follow up is: Time: ____________ Date: ____________

SIGNATURE: ______________________

LAST PAGE

REVISION LOG
Page 2 of 2

Revision Number	Effective Date	Pages Affected	Description of Revision
8	5/2/01	2, 37	Update Form "NRC Event Notification Worksheet" to reflect changes to NRC Form 361. (Minor/editorial changes.)
9	6/29/01	2-4, 6, 8-10, 38, 39	Added guidance on key information to be communicated to the NRC Resident Inspectors following a significant operating event.
10	2/6/02	3-9, 11-27, 32, 34	Revised procedure to clearly identify organizational responsibilities for making reportability determinations. immediate notifications, and follow-up written reports. Also added section lo Appendix A, pertaining to the optional verbal notification that is allowed under 10 CFR 50.73(a)(2)(iv)(A).
11	11/19/02	3, 4, 6-8, 14, 18-20, 23-51	Revised procedure to add reporting requirements (App. C) for Independent Spent Fuel Storage Facilities prescribed in 10 CFR 72. Also added flexibility to Appendix A, Section 3.5 pertaining to written reporting requirements. Also added Appendix J to provide guidance pertaining to serious Accident Internal Notifications. Added 30-day verbal NRC notification that is required by 10 CFR 20.2201(a)(ii) to address PER 02-000344-000.

TABLE OF CONTENTS

1.0 PURPOSE

This Standard Program and Process (SPP) specifies the requirements for various reports to the Nuclear Regulatory Commission (NRC) and other regulatory agencies to ensure compliance with the reporting requirements.

2.0 SCOPE

This SPP identifies the reporting requirements specified in the following:

A. Title 10, Code of Federal Regulations
B. Technical Specifications
C. Final Safety Analysis Reports
D. Correspondence to various regulatory agencies.

?he appendixes provide guidelines for reporting of conditions and/or events

3.0 INSTRUCTIONS

3.1 Periodic Reports

Each site licensing organization shall ensure that a matrix of their site's periodic reporting requirements is maintained.

Each report matrix should contain the report topic or title, regulatory basis for the report, time requirements for submitting the report, and the groupjs) responsible for preparing, coordinating review, obtaining approval and issuing the report.

A. Report Preparation

1. The organization or individual responsible for preparing the report shall ensure that the report is initiated in a timely manner (for proper review and approval), addresses all necessary items, is technically accurate. is reviewed and coordinated, and meets the requirements for submittal. If submitted to NRC, the report must be processed in accordance with Business Practice (BP) 213, "Managing TVA's Interface with NRC."

2. For reports to other regulatory agencies. the responsible organization shall ensure applicable reporting guidelines are satisfied.

3. Reports with commitments shall meet the requirements of SPP-3.3, "NRC Commitment Management."

B. Review and Approval of Reports

The organization or individual responsible for review shall ensure that the report is consistent with TVA policy and shall resolve comments. The organization or individual responsible for approval of the report shall perform a final review.

C. Distribution of Reports

The organization responsible for preparing the report shall ensure that adequate copies are made in a timely manner to support the transmittal of the report to the regulatory agency. Internal copies of the report shall be distributed in accordance with organizational procedures or instructions on preparation of the report.

D. Complete and accurate information must be provided to NRC at all times. Information can be in violation of this requirement even if it is not in writing, supplied under oath, or supplied without knowledge of its falsity. Information can be considered incomplete or inaccurate due to any one of the following reasons:

1. An affirmative statement which is false
2. An omission.
3. Inadequate review.
4. Failure to review
5. Careless disregard or deliberateness
6. Negligence not amounting to careless disregard
7. Inadvertent clerical or similar error involving information which, ha6 it been available to NRC and accurate at the time the information should have been submitted, wauld probably have resulted in regulatory action or NRC seekiriy additional information.
8. Failure to correc: material information tha: has significant implication for public health and safety which was correct when submitted but becomes incorrect due to subsequen! changes or events.

E. Identification of incorrect statements or misrepresentations having significant implication for public health and safety or common defense and security made in previously submitted information, including reports, must be reported to NRC's Regional Office within two working days of identifying the information. Phis requirement is not applicable to informatiort which is already required to be provided to NRC by other reporting or updating requirements. Licensing is responsible for making the determination of reportability and notifying NRC in accordance with 10 CFR 50.9.

3.2 Event or Condition Reporting

A. The specific event or condition reporting requirements applicable tc different licenses or permits TVA possesses are identified in the following Appendices:

1. Appendix A. "Repotting of Events or Conditions Affecting Licensed Nuclear Power Plants" contains the criteria for reporting of events or conditions affecting licensed nuclear power plants. Operations is responsible for making the reportability determinations for 50.72 and 50.73 repoits. Operatioris is responsible for making the immediate

notification to NRC in accordance with 10 CFR 50.72. Licensing is responsible for developing (with input from affected organizations) and submitting the written reports (or optional telephone reports) required by 10 CFR 50.73.

NOTE Reporting requirements for personnel exposure required by 10 CFR Part 20 are contained in RCDP-4, "Personnel lnprocessing and Dosimetry Administrative Processes."

Appendix A also contains criteria for reporting "radiation injuries" in accordance with 10 CFR 140.6. Site RADCON is responsible for reporting "radiation injuries" to Licensing. Licensing is responsible for developing and submitting the written report to NRC (with input frorn Site RADCON).

2. Appendix B, "Reporting of Events or Conditions Affecting Activities involving Byproducts, Source, or Special Nuclear Material Licenses" contains the criteria for reporting of events or conditions affecting activities involving byproduct. source or special nuclear material licenses. Site Licensing and Site RadCon are responsible for making the reportability determinations for 10 CFR Part 20, 30, 40, or 70 events associated with their site. Corporate Licensing and Corporate RadChem are responsible for making the reportability determinations for 10 CFR Part 20, 30. 40, or 70 events associated with ail other TVA licensed activities. Licensing is responsible for making the immediate notification and developiog (with input from affected organizations) and submitting written reports to NRC in accordance with 10 CFR Part 20. 30, 40, or 70 requirements.

3. Appendix C, "Reporting of Events or Conditions Affecting Independent Spent Fuel Storage Installations (ISFSI)" contains the criteria for reporting events or ccriditions affecting ISFSI. TVA, as the general licensee of the ISFSI, is required by 10 CFR 72.216 to make initial and written reports in accordance with 10 CFR 72.74 and 10 CFR 72.75. Operatioris is responsible for making the reportability determinations for 10 CFR 72.74 and 10 CFR 72.75 reports. Operations is responsible for making the immediate notification to NRC in accordance with 10 CFR 72.74. Operations is responsible for making the immediate, 4-hour. and 24-hour notifications in accordance with 10 CFR 72.75. Licensing is responsible for developing (with input from affected organizations) and submitting the written reports required by 10 CFR 72.75.

4. Appendix D, "Site Event Notificatioo Matrix" contains the internal management notification requirements for plant events. If designated TVA Manager is unavailable for notification due to temporary assignment. (e.g., INPO loanee) notification should be made to designee or next higher manage;. Operations and the Plant Manager (or Duty Plant Manager) are responsible for making these internal management notifications.

5. Appendix E, "Other Regulatory Reporting" contains the criteria for reporting of events or conditions to Federal and State regulatory agencies other than the NRC. Operations is responsible for making the reportability determinations and notifications for these non-NRC Federal and State regulatory agency reporting requirements.

NOTE Additional reporting guidance for defects is contained in SPP-3.1, "Corrective Action Program."

6. Appendix F, "Evaluatien and Reporting of Defects and Failures to Comply Associated with Substantial Safety Hazards Per 10 CFR 50.55(e) Reporting Requirements" contains the criteria tor reporting of deficiencies for nuclear plants with a construction permit in accordance with 10 CFR 55.55(e). Licensing is responsible for making the final reportability determination and written report to NRC in accordance with 10 CFR 50.55(e).

NOTE Additional reporting guidance for deficiencies is contained in SPP-3.1

7. Appendix G, "Determination of Reportability Under 10 CFR Part 21" contains the criteria for reporting defects in basic components in accordance with 10 CFR Part 21. Licensing is responsible for making the final reportability determination and written report to NRC in accordance with 10 CFR Part 21.

8. Appendix H, "Reporting of Decommissioning Funding" contains the criteria for notifying NRC when permanently shutting down the operation of a reactor, as required by 10 CFR 50.54(bb). Licensing is responsible for making the written notification to NRC in accordance with 10 CFR 50.54(bb).

9. Appendix I, "Communication with the NRC Following a Significant Operational Event" contains guidance on communications that needs to be established with the NRC within 24 to 36 hours following a significant operational event that could result in an incident investigation by the NRC. Site Licensing coordinates this communication with NRC.

10. Appendix J. "Internal Notification of Events Requiring Serious Accident Investigations" provides internal management notification requirements for serious accidents, as prescribed in TVA-SPP-18.010, "Conduct Serious Accident Investigation."

11. Reporting requirements for fitness for duty events required by 10 CFR Part 26 are contained in SPP-1.2 "Fitness For Duty." Responsibilities for reportability determinations ard irnrnediate notification requirements are assigned to Site Nuclear Security and Corporate Nuclear Security. Licensing *is* responsible for making the written reports required by 10 CFR Part 26.

12. Reporting requiremerits for events or conditions affecting the physical protection *of* the licensed nuclear plant specified in 10 CFR 73.71 are contained in SPP-1.3 'Plant Access and Security." Responsibilities for

reportability determiriations and immediate notification requirements are assigned to Site Nuclear Security and Corporate Nuclear Security. In accordance with SPP-1.3, **if NRC** notification is required (e.g., one or twenty-four hour phone call), the Site Security Manager will request the Plant Shifl Manager to call the **NRC** Operations Center. Licensing is responsible for making the written reports required by 10 **CFR** Part **73.71.**

B. Report Preparation

1. The organization or individual responsible for preparing the report shall ensure that the report is initiated in a timely manner (for proper review and approval), addresses all necessary items, is technically accurate, is reviewed and coordinated, and meets the requirements for submittal. If subrnitted to NRC, the report must be processed in accordance with Business Practice (BP) 213, "Managing TVA's Interface with **NRC.**"

2. For reports to other regulatory agencies, the responsible organization shall ensure applicable reporting guidelines are satisfied.

3. **Reports** with cornrnitrnents shall meet the requirements of SPP-3.3, "NRC Commitment Management."

C. Review and Approval of Reports

The organization or individual responsible for review shall ensure that the report is consistent with TVA policy and shall resolve comments. The organization or individual responsible for approval of the report shall perform a final review.

5. Distribution of Reports

The organization responsible for preparing the report shall ensure that adequate copies are made in a tirnely manner to support the transmittal of the repott to the regulatory agency. Internal ccpies of the report shall **be** distributed in accordance with organizational procedures or instructions on preparation of the report.

E. Complete and accurate information must be provided to NRC at all times. Information can be in violation of this requirernerrt even if it is not in writing. supplied under oath, or supplied without knowledge of its falsity. Information can be considered incomplete or inaccurate dce to any one of the following reasons:

1. An affirmative statement which is false.

2. An omission

3. Inadequate review

4. Failure to review

5. Careless disregard or deliberateness.

5. Negligence not amounting to careless disregard

7. Inadvertent clerical or similar error involving information which, had it been available to NRC and accurate at the time the information should have been submitted, would probably have resulted in regulatory action or NRC seeking additional information.

a. Failure to correct material information that has significant implication tor public health and safety which was correct when submitted but becomes incorrect due to subsequent changes or events.

4.0 RECORDS

Records of the reports and their transmittals shall be maintained in RIMS as non-QA records unless separate procedures require the reports to be maintained as QA records.

5.0 DEFINITIONS

Actuation - The minimum number cf tripped channels required to complete the logic of a function. (Example: 2/4 logic - at least two (2) channels must trip to be considered an actuation

Administrative Control Program (ACP) - An approved, proceduralized method of documenting adverse conditions and implementing corrective action.

Basic Component - When applied to nuclear power reactors, means a plant structure, system component, or part thereof necessary to ensure:

1. The integrity of the reactor coolant pressure boundary.
2. The capability lo shut down the reactor and maintain it in a safe shutdown condition or
3. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in 10 CFR 100.11

In all cases, "Basic Component" includes safety-related design, analysis, inspection, testing, fabrication. replacement parts, or consulting services that are associated with the component hardware whether these services are performed by the component supplier or other (10 CFR 21.3 and 10 CFR 50.2).

Completion Of Any Nuclear Plant Shutdown - This is when reactor is taken subcritical.

Construction - The analysis, design, manufacture, fabrication, quality assurance, placement, erection, installation, modification, inspection, or testing of a facility or activity and consulting services related to the facility or activity that are important to safety.

Control Of The Items - TVA is subject to Part 21 requirements regarding procured materia'. components, or parts after TVA has taken control of the items. This occurs after conducting the required receipt inspection. Evaluation of defects found during receipt inspection is the responsibility of the supplier if the items are returned to the supplier. If TVA accepts ownership of the item, any defect must be evaluated and. if applicable, reported under Part 21