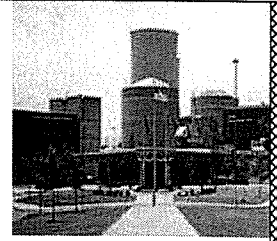
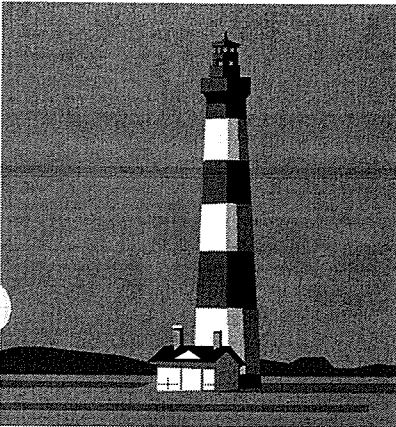


Sequoyah Nuclear Plant



SEPT 2010 NRC INITIAL LICENSE ADMINISTRATIVE JPMs



Facility: <u>Sequoyah Nuclear Station 1 & 2</u>		Date of Examination: <u>09/13/2010</u>
Examination Level: RO <input checked="" type="checkbox"/> SRO <input type="checkbox"/>		Operating Test Number: <u>2010302</u>
Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	M, C, P	2.1.5 Ability to use procedures related to shift staffing, such as minimum crew compliment, overtime limitations, etc (2.9/3.9) A.1.a Evaluate Overtime Restrictions (Both RO/SRO)
Conduct of Operations	M, C	2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc. (3.9/4.2) A.1.b Calculate Manual Makeup to the VCT
Equipment Control	D, R	2.2.12 Knowledge of surveillance procedures (3.7/4.1) A.2 Boric Acid Storage Tank Level Operability Determination
Radiation Control	N, R	2.3.4 Knowledge of radiation exposure limits under normal and emergency conditions. (3.2/3.7) A.3 Determine Potential Total Dose for Valve Alignment
Emergency Procedures/Plan		
NOTE: All items (5 total) are required for SROs. RO applicants require only 4 items unless they are retaking only the administrative topics, when all 5 are required.		
* Type Codes & Criteria: (C)ontrol room, (S)imulator, or Class(R)oom (D)irect from bank (≤ 3 for ROs; ≤ 4 for SROs & RO retakes) (N)ew or (M)odified from bank (≥ 1) (P)revious 2 exams (≤ 1 ; randomly selected)		

A.1.a Evaluate Overtime Restrictions – This JPM has the candidate review a list of times for scheduled work and determine if any of the scheduled times exceed the procedure requirements for allowable work. This JPM is a Modified Bank JPM that was used on Feb 2009 exam.

A.1.b Calculate Manual Makeup to the VCT – This JPM has the candidate determine the required amount of Boric Acid and Demineralized Water needed to perform a manual makeup to the VCT without changing RCS boron concentration. This is a Modified Bank JPM with (the VCT level change and initial RCS boron concentration have been changed) .

A.2 Boric Acid Storage Tank Level Operability Determination – This JPM has the candidate determine the operability of a Boric Acid Storage Tank before placing the Tank in service. This is a Bank JPM.

A.3 Determine Potential Total Dose for Valve Alignment – This JPM has the candidate determine their potential accumulated dose to perform a valve manipulation in the Unit 1 auxiliary building and determine if they will exceed their Administrative Dose Limit (ADL). This is a New JPM.

**SEQUOYAH NUCLEAR PLANT
September 2010 NRC Exam**

RO/SRO A.1.a

Evaluate Overtime Restrictions

**RO/SRO
JOB PERFORMANCE MEASURE**

Task: Evaluate Overtime Restrictions

Task #: (RO) 1190030301 ; (SRO) 3430050302 ; (SRO) 0001650302

Task Standard: The candidate identifies the need for an Overtime Limitation Exception Report on 2 occasions:
On 05/30/10 to allow exceeding 26 hours in a 48 hour period.
On 06/04/10 to allow exceeding 72 hours in 7 days.

Time Critical Task: YES: _____ NO: X

K/A Ratings: G 2.1.5 (2.9/3.9)

Method of Testing:

Simulated Performance: _____ **Actual Performance:** X

Evaluation Method:

Simulator _____ **In-Plant** _____ **Classroom** X

Main Control Room _____ **Mock-up** _____

Performer: _____
Trainee Name

Evaluator: _____ / _____
Name / Signature DATE

Performance Rating: SAT: _____ UNSAT: _____

Validation Time: 10 minutes **Total Time:** _____

Performance Time: **Start Time:** _____ **Finish Time:** _____

COMMENTS

SPECIAL INSTRUCTIONS TO EVALUATOR:

1. Critical steps are identified in step SAT/UNSAT column by bold print 'Critical Step'.
2. Any UNSAT requires comments
3. Ensure operator performs the following required actions for **SELF-CHECKING**;
 - a. Identifies the correct unit, train, component, etc.
 - b. Reviews the intended action and expected response.
 - c. Compares the actual response to the expected response.

Tools/Equipment/Procedures Needed:

SPP-1.5, Fatigue Management and Work Hour Limits, Section 3.2

References:

	Reference	Title	Rev No.
1.	SPP-1.5	Fatigue Management and Work Hour Limits	0008
2.	0-PI-OPS-000-027.0	Shift Manager Clerk Duty Station Shift Relief and Office Round Sheets.	0038

=====

READ TO OPERATOR

DIRECTIONS TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM. 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. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

You are a Licensed Operator that has worked the following schedule:

Date	Hours	Status	Notes
05/26/10	OFF		
05/27/10	OFF		
05/28/10	OFF		
05/29/10	0630 -2230	Normal Off Day	Worked on an Off day and stayed over 4 hours until relief arrived
05/30/10	1830-0700	Normal Work Day	30 minute turnover
05/31/10	1830-0645	Normal Work Day	15 minute turnover
06/01/10	1830-0715	Normal Work Day	45 minute turnover
06/02/10	1830-0730	Normal Work Day	60 minute turnover
06/03/10	OFF	Normal Off Day	
06/04/10	0630-1845	Normal Off Day	Called in to cover shift (15 minute turnover)
06/05/10	1300-1900	Normal Off Day	Called in to cover shift to relieve a sick operator. (30 minute turnover)
06/06/10	A/L	Normal Work Day	Took Annual Leave for the shift
06/07/10	0630-1845	Normal Work Day	15 minute turnover
06/08/10	0630-1930	Normal Work Day	60 minute turnover
06/09/10	OFF	Normal Off Day	

INITIATING CUES:

Determine the date(s), if any, that would exceed the Fatigue Rule and would require an Overtime Limitation Exception Report to be completed prior to you completing the identified working hours and the reason(s) for the report(s) being required.

Job Performance Checklist:

STEP / STANDARD	SAT / UNSAT
<p><u>STEP 1.:</u> Evaluate the hours worked against the requirements.</p> <p><u>STANDARD:</u> Candidate identifies an Overtime Limitation Exception Report is required prior to completing the 05/30/10 shift to allow exceeding 26 hours in a 48 hour period.</p> <p>Shaded portion is critical because the worker would exceed 26 hrs in a 48 hr period if allowed to complete the shift on 05/30/10.</p> <p><u>COMMENTS:</u></p>	<p>_____ SAT</p> <p>_____ UNSAT</p> <p>Critical Step (shaded portion)</p>
<p><u>STEP 2.:</u> Evaluate the hours worked against the requirements.</p> <p><u>STANDARD:</u> Candidate identifies an Overtime Limitation Exception Report is required prior to completing the 06/04/10 shift to allow exceeding 72 hours in a 7 day period.</p> <p>Shaded portion is critical because the worker would exceed 72 hrs in a 7 day period if allowed to complete the shift on 06/04/10.</p> <p><u>Cue:</u> This completes the JPM.</p> <p><u>COMMENTS:</u></p>	<p>_____ SAT</p> <p>_____ UNSAT</p> <p>Critical Step (shaded portion)</p> <p>Stop Time _____</p>

READ TO OPERATOR

DIRECTIONS TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM. 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. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

You are a Licensed Operator that has worked the following schedule:

Date	Hours	Status	Notes
05/26/10	OFF		
05/27/10	OFF		
05/28/10	OFF		
05/29/10	0630 -2230	Normal Off Day	Worked on an Off day and stayed over 4 hours until relief arrived
05/30/10	1830-0700	Normal Work Day	30 minute turnover
05/31/10	1830-0645	Normal Work Day	15 minute turnover
06/01/10	1830-0715	Normal Work Day	45 minute turnover
06/02/10	1830-0730	Normal Work Day	60 minute turnover
06/03/10	OFF	Normal Off Day	
06/04/10	0630-1845	Normal Off Day	Called in to cover shift (15 minute turnover)
06/05/10	1300-1900	Normal Off Day	Called in to cover shift to relieve a sick operator. (30 minute turnover)
06/06/10	A/L	Normal Work Day	Took Annual Leave for the shift
06/07/10	0630-1845	Normal Work Day	15 minute turnover
06/08/10	0630-1930	Normal Work Day	60 minute turnover
06/09/10	OFF	Normal Off Day	

INITIATING CUES:

Determine the date(s), if any, that would exceed the Fatigue Rule and would require an Overtime Limitation Exception Report to be completed prior to you completing the identified working hours and the reason(s) for the report(s) being required.



NPG Standard Programs and Processes

TITLE

Fatigue Management and Work Hour Limits

**SPP-1.5
Rev. 0008
Page 1 of 52**

Quality Related Yes No

Effective Date 06-28-2010

Responsible Peer Team/Working Group: Licensing

Approved by: R. M. Cook for R. M. Krich 6/18/2010
Corporate Functional Area Manager Date

3.2 Requirements

3.2.1 10 Code of Federal Regulations (CFR) 26 Overtime Limits

A. The following limits apply to covered individuals regardless of unit status:

1. No more than 16 work hours in any 24 hour period
2. No more than 26 work hours in any 48 hour period
3. No more than 72 work hours in any 7 day period
4. At least a 10 hour break between successive work periods, or an 8 hour break when a break of less than 10 hours is necessary to accommodate a crew's scheduled transition between work schedules or shifts.
5. At least a 34 hour break in any 9 calendar day period.

B. Online Requirements

1. During online operations and without issuance of a waiver, an individual's required average minimum days off shall adhere to the requirements listed in Table 1 below (averaged over the shift cycle):

Group	8 Hour Shift	10 Hour Shift	12 Hour Shift
Maintenance	1 day off/week	2 days off/week	2 days off/week
Operations, Radiation Protection, Chemistry	1 day off/week	2 days off/week	2.5 days off/week
Security	1 day off/week	2 days off/week	3 days off/week

2. For the purposes of calculating an average number of days off the duration of the shift cycle may not exceed six weeks nor be less than one week.
3. Online rules will be applied for a shift when any portion of the shift where the unit is defined to be online.

3.2.1 10 Code of Federal Regulations (CFR) 26 Overtime Limits (continued)

C. Outage Requirements

1. While working on an outage unit, and without issuance of a waiver, an individual's required days off shall adhere to the requirements listed in Table 2 below (not an average):

Table 2. Required Minimum of Days Off for Outages			
Group	8 Hour Shift Days Off	10 Hour Shift Days Off	12 Hour Shift Days Off
Maintenance	1 day off per week	1 day off per week	1 day off per week
Operations, Radiation Protection, Chemistry	3 days off in each successive (i.e., non-rolling) 15 day period	3 days off in each successive (i.e., non-rolling) 15 day period	3 days off in each successive (i.e., non-rolling) 15 day period
Security	4 days off in each successive (i.e., non-rolling) 15 day period	4 days off in each successive (i.e., non-rolling) 15 day period	4 days off in each successive (i.e., non-rolling) 15 day period

2. Table 2 applies to the first 60 days of a unit outage; after this 60 day period expires normal online work hour limits will apply.

The 60 day period may be extended seven days on an individual basis for each 7 day block during which they worked 48 hours or less. Multiple 7 day extensions may be given to an individual as long as the 7 day blocks are not overlapping.

3. An operator who is on outage work-hour limitations should not provide relief to the operator at the controls or the senior operator in the control room for an operating unit, unless another operator who has been on non-outage work hours is not immediately available and the operator has had 2 days off in the preceding 7 day period. If the operator has not had 2 days off in the preceding 7 day period and no other operator who has had 2 days off is immediately available, the operator may provide short-term relief (up to 2 hours) to the operator at the controls or the senior operator in the control room for an operating unit or long-term relief (more than 2 hours) under a waiver of the Minimum Days Off (MDO) requirement that is applicable to the shift schedule (i.e., 8, 10, or 12 hour shifts) for personnel assigned to the operating unit.
4. Contractors/vendors are responsible for tracking and reporting their hours to their supervisors between outages. The NPG is not responsible for accounting for contractor/vendor work hours prior to supporting an outage unless the contractor/vendor is coming directly from another NPG site.
5. Minimum days off may be scheduled throughout the week, or 15 day period, without restraint (i.e., does not have to be one (1) day off every five (5) days).

3.2.1 10 Code of Federal Regulations (CFR) 26 Overtime Limits (continued)

6. A new shift cycle will be used at the completion of an outage. Post-outage transitions are in compliance if the unplanned outage schedule for the shift cycle would have provided for the required average days off.
7. If the interval between outages for a worker is less than nine (9) days, then the individual shall have a 34 hour break period and shall not exceed the following limits:
 - a. 16 work hours in any 24-hour period,
 - b. 26 work hours in any 48-hour period, and
 - c. 72 work hours in any 7-day period.
8. The option of keeping workers on normal online work hour restrictions is still available during outages.

3.2.2 Calculating Work Hours

- A. Work hour limits and the associated calculation and tracking of work hours apply to the individuals who perform covered work. This calculation includes both covered and non-covered work since the latter also contributes to fatigue.
- B. Work hour accounting practices may be different from record keeping for payroll purposes or gate times.
- C. Work hour records should show the number of hours worked each calendar day. Work period start and stop times should be recorded and documented in a consistent manner.
- D. Included In Work Hour Calculation
 1. All work, both covered and non-covered work, performed by the individual for the licensee (including hours worked while "working from home")
 2. Non-incidentals duties (i.e., incidentals duties exceeding the nominal, cumulative 30 minutes) performed off-site. The time between leaving the station and the non-incidentals duty are also included if that time is less than 10 hours in duration
 3. Any break time during the work period
 4. Time spent at lunch
 5. Call-ins (when an individual is called in to work) are considered an addition to the normal work schedule and, therefore, shall be included. The time between leaving the station and the call-in work period are also included if that time is less than 10 hours in duration
 6. Shift holdovers to cover for late arrivals of incoming shift members

3.2.2 Calculating Work Hours (continued)

7. Early arrivals of individuals for licensee required meetings, training, or pre-shift briefings for special evolutions (these activities are not considered shift turnover activities)
8. Holdovers for interviews needed for event investigations

E. Excluded From Work Hour Calculation

1. Either on-coming or off-going shift turnover
2. Only that portion of a break or rest period during which there is a reasonable opportunity and accommodation for restorative sleep (e.g., a nap of at least 30 minutes)
3. Incidental duties performed off-site (e.g., technical assistance by telephone from home), provided the cumulative duration of the work does not exceed a nominal 30 minutes during any single break period, may be excluded and are not considered a work period/work shift. Exceeding the nominal 30 minutes results in that time period counting as a workday.
4. Hours worked above the normal scheduled work hours for the purpose of participating in the actual conduct of an unannounced emergency preparedness exercise or drill may be excluded from the calculation of an individual's work hours. If an individual is on a day off, it is still considered a day off.
5. Paid hours during which the individual is not expected to perform work (e.g., vacation time, sick days, personal leave).
6. Waivers for changing to/from daylight savings time are not required

3.2.3 Work Hour Scheduling

- A. Work hours for covered individuals shall be scheduled with the objective of preventing impairment from fatigue due to duration, frequency, or sequencing of successive shifts.
- B. Consider the following factors when establishing schedules:
 1. Work hour limits defined in 10 CFR 26, Subpart I
 2. Consistent start/stop times for work periods
 3. Impact of backward shift rotation
 4. Training requirements
 5. Vacation scheduling

NPG Standard Programs and Processes	Fatigue Management and Work Hour Limits	SPP-1.5 Rev. 0008 Page 18 of 52
-------------------------------------	-----------------------------------------	---------------------------------------

3.2.3 Work Hour Scheduling (continued)

NOTE

To determine if an individual is eligible for a call-in work period use the guidelines from Steps 3.2.3C and 3.2.3D.

- C. Evaluation of periodic overtime for covered individuals' work hours should be performed with respect to an expected average of 54 hours per week.
- D. Hours worked should be evaluated to determine if any limit will be exceeded based on the work schedule by a backward look at the number of hours that have or will have been worked based on a time in the future (i.e., if the individual works at time T, how many hours will he/she have worked in the 24 hours/48 hours/7 days preceding time T relative to 10 CFR 26 work hour limits as well as minimum days off and break requirements).
- E. When entering an unplanned outage, unplanned security outage or increased threat condition, compliance with 10 CFR 26 work hour limits is met if the schedule for the shift cycle would have provided the required average days off.
- F. Deviations from 10 CFR 26 overtime limits may occur as the result of administrative errors or unforeseen circumstances. A PER shall be initiated, in accordance with SPP-3.1, Issue Identification and Screening Process, for each individual when this occurs.
- G. An individual is considered "reset" from deviation, whether under a waiver or otherwise, when they meet the hours worked and break requirements and have met the minimum days off requirement in the last seven (7) day period.
- H. Refer to eSOMS related guidance for use of the software to perform work hour calculating and scheduling.

3.2.4 Transitions

- A. Non-shift workers transitioning onto a shift (e.g., non-shift SRO standing proficiency watch) shall adhere to 10 CFR 26 overtime limits and have all work (including non-covered work performed before joining the shift) included in the calculation of their work hours. If the individual joins a shift after the start of a shift cycle, they shall meet the average days off requirement going forward and the minimum days off requirement for the shift from which they have transitioned.
- B. If an individual is transitioning to a different shift schedule for a long period of time, the applicable day off requirement is found by calculating the average duration of the shifts worked and to be worked during a period not more than six (6) weeks that encompasses the schedule transition. If the average shift duration is not more than nine (9) hours, then the minimum day off requirements for 8-hour shift schedules would apply. If the average shift duration is more than 9 hours but not more than 11 hours then the requirements for a 10-hour shift would apply, and so forth.

NPG Standard Programs and Processes	Fatigue Management and Work Hour Limits	SPP-1.5 Rev. 0008 Page 19 of 52
-------------------------------------	-----------------------------------------	---------------------------------------

3.2.4 Transitions (continued)

- C. If an individual is transitioning to a temporary shift schedule, then the individual's MDOs shall be evaluated against the average shift hours.
- D. When an individual works during two or more successive outages and the interval(s) between successive outages is less than nine (9) days, then the individual must have had a 34-hour break period and have not exceeded the following limits: 16 work hours in any 24-hour period, 26 work hours in any 48-hour period, and 72 work hours in any 7-day period.

3.2.5 Accounting

- A. Accounting practices may be different from record keeping for payroll purposes. Work periods should be rounded consistently. Also, it should be noted that gate or payroll times may not be an appropriate measure of 10 CFR 26 compliance since these measures may not be representative of work activities.
- B. Work hour records should show the number of hours worked each calendar day. Work period start and stop times should be recorded and documented in a consistent manner. Hours worked and start/stop times are per the scheduled shift times unless a change to work times is entered and approved.
- C. The cognizant supervisor will ensure any hours worked in excess of the schedule are included into eSOMS prior to the beginning of the individual's next shift.
- D. The periods of "24 hours," "48 hours," and "7 days" are considered rolling time periods. Rolling means the period is not re-zeroed or reset following a day off or after obtaining authorization to exceed 10 CFR 26 overtime limits. The "24 hours," "48 hours," and "7 days" periods do not restart after a day off; the period continues to roll.

3.2.6 Waivers

- A. Waivers shall be granted on an individual basis and only to address circumstances that could not have been reasonably controlled. Waivers may be granted if both of the following requirements are met:

1. The Site Vice President determines that the waiver is necessary to mitigate or prevent a condition adverse to safety, or to maintain site security, and approves the waiver.

The cognizant supervisor shall perform a face-to-face fatigue assessment on the individual and determine if there is reasonable assurance that the individual will be able to safely and competently perform assigned duties during the additional work period for which the waiver will be granted. The supervisor performing the assessment shall be trained in accordance with the requirements of 10 CFR 26.29 and 10 CFR 26.203(c) (ATIS 00074642) and shall be qualified to oversee the work (or related work) to be performed by the individual.

Fatigue assessments prior to the individual performing any work under the waiver shall be performed no more than four hours before the work activity.

3.2.6 Waivers (continued)

- B. If there is no supervisor on-site who is qualified to oversee the work, the assessment may be performed by another trained supervisor who is qualified to oversee the work being performed.
- C. The documented basis for a waiver must include the circumstances that necessitate the waiver, a statement of the scope of work, a time period for which the waiver is approved, and the bases for the approval of the waiver.
- D. Examples of instances that would most likely meet the threshold for a waiver:
 - 1. Risk significant SSC fails placing the unit in a shutdown LCO and maintenance crews must be held over or called in for repairs.
 - 2. At the onset of an unplanned outage an individual, who is part of the minimum shift complement, must be held over due to relief not available.
 - 3. Severe weather that requires hold-over of personnel.
- E. Cognizant supervisors must track waivers by initiating a PER. See Attachment 2 for waiver process.

3.2.7 Fatigue Assessments**A. Conditions Requiring a Fatigue Assessment****1. For Cause**

In response to an observed condition of impaired individual alertness creating a reasonable suspicion that an individual is not fit for duty.

- a. Conduct fatigue assessment in response to an observed condition of impaired individual alertness creating a reasonable suspicion that an individual is not fit to safely and competently perform their duties, except if the condition is observed during an individual's break period.
- b. A fatigue assessment need only be conducted when the observed condition is reasonably believed to be due to impaired alertness with no other behaviors or physical conditions creating a reasonable suspicion of possible substance abuse.
- c. The individual who observed the condition may not conduct the fatigue assessment.

2. Self-Declaration

In response to an individual's self-declaration to their supervisor that they are not fit to safely and competently perform their duties for any part of a work shift because of fatigue.

3.2.7 Fatigue Assessments (continued)

- a. It is the responsibility of each individual to communicate a clear self-declaration of fatigue to their cognizant supervisor. This may be verbally initiated, when necessary, by using the statement below. A casual statement to a supervisor or fellow employee that an individual is tired is not a self-declaration. Any individual covered by the FFD program can self-declare.

"By the requirements of 10 CFR 26, I believe I am too fatigued to perform the duties assigned to me and would like to make a self-declaration of fatigue."

Once an individual has made a verbal self-declaration, they must be removed, as soon as practicable, from duty and given the opportunity to complete Attachment 1, Section II. The individual must ensure their cognizant supervisor receives the form prior to the end of shift. The self-declaration is formal once an individual has submitted it to the supervisor; the cognizant supervisor completes Attachment 1, Sections III and IV.

- b. Self-declarations of fatigue should be encouraged to facilitate a healthy safety conscious work environment.
- c. If an individual is performing or being assessed for work under a waiver and makes a self-declaration of fatigue, the cognizant supervisor, as soon as practicable, shall stop the individual from performing any covered work and allow that individual to complete Attachment 1, Section II.
- d. A self-declaration fatigue assessment shall be performed as soon as possible.
- e. If the individual must continue performing the covered work until relieved, then the supervisor shall take immediate action to relieve the individual and provide oversight of the individual.
- f. Following the self-declaration or relief from performing the covered work, as applicable, the cognizant supervisor:
 - (1) May reassign the individual to duties other than covered work, but only if the results of a fatigue assessment indicate the individual is fit to safely and competently perform those other duties.
 - (2) May permit or require the individual to take a break of at least 10 hours before the individual returns to performing any covered work and ensure the individual is provided safe transportation home.
 - (3) Conduct a fatigue assessment in response to an individual's self-declaration to their supervisor that they are not fit to safely and competently perform their duties for any part of a work shift because of fatigue.

3.2.7 Fatigue Assessments (continued)

- g. If an individual disagrees with the results of a fatigue assessment, then the individual may request a second assessment by another trained FFD assessor. The individual may also pursue other management and oversight paths for resolution.

3. Post-event

In response to events requiring post-event drug and alcohol testing as specified in 10 CFR 26.31(c).

- a. Conduct a fatigue assessment in response to events requiring post-event drug and alcohol testing. (SPP-1.2)
- b. Necessary medical treatment shall not be delayed in order to conduct a fatigue assessment.
- c. The individual who conducts the fatigue assessment may not have:
 - (1) Performed or directed (on-site) the work activities during which the event occurred,
 - (2) Performed, within 24 hours before the event occurred, a fatigue assessment of the individuals who were performing or directing (on-site) the work activities during which the event occurred, and
 - (3) Evaluated or approved a waiver of the limits for any of the individuals who were performing or directing (on-site) the work activities during which the event occurred, if the event occurred while such individuals were performing work under that waiver.

4. Follow-up

To follow up after a "for cause" fatigue assessment or before an individual returns to work after a self-declaration resulting in a break of less than 10 hours.

- a. Conduct a follow-up fatigue assessment and determine the need to implement controls and conditions before permitting the individual to resume performing any duties if:
 - (1) A fatigue assessment was conducted for cause, or
 - (2) After a self-declaration, the individual returned to duty following a break of less than 10 hours.
 - (3) If no break occurs, only one assessment is required.

5. Waiver

As required by Section 3.2.6, Waivers.

NPG Standard Programs and Processes	Fatigue Management and Work Hour Limits	SPP-1.5 Rev. 0008 Page 23 of 52
-------------------------------------	-----------------------------------------	---------------------------------------

3.2.7 Fatigue Assessments (continued)

B. Fatigue assessments shall, at a minimum, address:

1. Acute and cumulative fatigue considering the individual's work history for at least the past 15 days. Fatigue may not be assessed solely based on the fact that an individual has not exceeded any work hour limits and has met all minimum day off requirements.
2. Potential degradations in alertness and performance due to circadian variations (particularly with respect to time of day when assessing for a waiver).
3. Potential degradations in alertness and performance to affect risk significant functions.
4. Whether any controls and conditions must be established under which the individual will be permitted to perform work.

C. Conducting a Fatigue Assessment

1. Only supervisors and FFD program personnel who are trained under 10 CFR 26.29 and 10 CFR 26.203(c) (ATIS Training Course 00074642) may conduct fatigue assessments. The fatigue assessor shall limit inquiries of the individual to information necessary to assess the required factors and review the individual's performance, if applicable.
2. Individuals shall provide complete and accurate information that may be required to address the required factors. The fatigue assessor shall limit inquiries of the individual to information necessary to assess the required factors and review the individual's performance, if applicable.
3. Fatigue assessments shall not conclude an individual is fit-for-duty solely based on the fact that the individual's work hours have not exceeded any of the work hour limits or that the individual has had the minimum breaks or minimum days off, as applicable.
4. Following a fatigue assessment, the cognizant supervisor shall determine and implement the controls and conditions, if any, which are necessary to permit the individual to resume the performance of their duties for the licensee, including the need for a break.
5. Individuals sent home in lieu or as a result of a fatigue assessment should be evaluated to determine if alternate transportation is appropriate. Personnel safety should be considered in any decision to send someone home due to fatigue.
6. Cognizant supervisors shall document the circumstances that necessitated the fatigue assessment and any controls and conditions that were implemented in Attachment 1. A copy of this record will be forwarded to and maintained by the Site Subject Matter Expert. Cognizant supervisors must track fatigue assessments by initiating a PER coded as "Fatigue Rule" and ensure the use of appropriate anonymity in the PER.

3.2.7 Fatigue Assessments (continued)

7. Each site shall maintain fatigue assessment records for covered individuals for three years. These records shall include:
 - a. The conditions under which each fatigue assessment was conducted (i.e., self-declaration, for cause, post-event, follow-up).
 - b. Documentation of whether or not the individual was working on outage activities at the time of the self-declaration or condition resulting in the fatigue assessment.
 - c. The category of duties the individual was performing, if the individual was performing covered work at the time of the self-declaration or condition resulting in the fatigue assessment.
 - d. The management actions, if any, resulting from each fatigue assessment.

3.3 Fatigue Management

- A. Fatigue management requirements, with the exception of work hour controls, are part of the FFD Program requirements and apply to all individuals (i.e., NPG employees and contractors/vendors) who:
 1. Have unescorted access to protected areas (even if their current location is not on-site), or
 2. Are required to physically report to the Technical Support Center or an Emergency Operations Facility, in accordance with site Emergency Plans and procedures (even if their current location is not on-site or they do not have unescorted access). Except for covered individuals in this group, the work hour limits discussed in Step 3.2 are not applied to all emergency response personnel.
- B. Personnel are required to be fit for duty. Getting sufficient rest is required to ensure a person is not subject to fatigue.
- C. Personnel who make choices that result in less than adequate sleep to remain alert and avoid fatigue are not meeting their obligations.

3.4 Conflict Resolution

NOTE

If the individual was determined to be fit-for-duty and disagrees with this finding, the cognizant supervisor needs to consider the impact of the individual working under distress. The cognizant supervisor should engage their management to discuss options and trending.

- A. All applicable individuals have the right to self-declare. Self-declarations of fatigue should be encouraged and respected.

SEQUOYAH NUCLEAR PLANT JOB PERFORMANCE MEASURE

RO Admin A.1.b

Calculate Manual Makeup to the VCT

RO/SRO
JOB PERFORMANCE MEASURE

Task: Calculate Manual Makeup to the Volume Control Tank

Task #: (RO) 0040250101

Task Standard: Manual make up to the VCT to bring level up 10% is determined to be 179±2 gallons water and 21±2 gallons of boric acid.

Alternate Path: YES: _____ NO: _____

Time Critical Task: YES: _____ NO: X

K/A Reference/Ratings: G2.1.25 (3.9/4.2)
004A4.13 (3.3/2.9)

Method of Testing:

Simulated Performance: _____ **Actual Performance:** X

Evaluation Method:

Simulator _____ **In-Plant** _____ **Classroom** X

Main Control Room _____ **Mock-up** _____

Performer: _____
Trainee Name

Evaluator: _____ / _____
Name / Signature DATE

Performance Rating: SAT: _____ UNSAT: _____

Validation Time: 20 minutes **Total Time:** _____

Performance Time: **Start Time:** _____ **Finish Time:** _____

COMMENTS

SPECIAL INSTRUCTIONS TO EVALUATOR:

1. Critical steps are identified in step SAT/UNSAT column by bold print 'Critical Step'.
2. Any UNSAT requires comments
3. Ensure operator performs the following required actions for **SELF-CHECKING**;
 - a. Identifies the correct unit, train, component, etc.
 - b. Reviews the intended action and expected response.
 - c. Compares the actual response to the expected response.

Tools/Equipment/Procedures Needed:

0-SO-62-7, section 6.5, Appendix C
TI-44, Appendix C Table 1, Table 2

References:

	Reference	Title	Rev No.
1.	0-SO-62-7	Boron Concentration Control	58
2.	TI-44	Boron Tables	0012

=====

READ TO OPERATOR

DIRECTIONS TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM. 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. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

1. Unit 1 is at 100% full power, steady state.
2. Reactor Coolant System boron concentration is 755 ppm.
3. BAT boron concentration is 6570 ppm.
4. B-10 depletion value is 48 ppm.
5. REACTF software is not available for boron calculations.

INITIATING CUES:

1. You are the Unit 1 OATC and are to perform a calculation for a manual blended makeup to the Chemical Volume Control System in accordance with 0-SO-62-7, "Boron Concentration Control," to increase Volume Control Tank level from 25% to 35%.
2. Notify the SRO when the calculation is completed.

Job Performance Checklist:

STEP / STANDARD	SAT / UNSAT
<p><u>STEP 1.:</u> Operator obtains appropriate copy of procedure and determines the appropriate section to perform.</p> <p><u>STANDARD:</u> Operator obtains copy of 0-SO-62-7, "Boron Concentration Control," and determines that section 6.5 is the appropriate section.</p> <p><u>COMMENTS:</u></p>	<p>_____ SAT</p> <p>_____ UNSAT</p>
<p>Evaluator Note: The following Cautions and Notes are from Sect 6.5 and precede step [1]</p>	
<p>CAUTION Returning the Boric Acid Blender to service after unplugging, cleaning, or maintenance on the Boric Acid System could introduce debris, sludge, air or chunks of solidified boron into the CCP suction resulting in pump damage. Extreme care must be exercised to properly flush the Boric Acid Blender system following an outage. [C.2]</p> <p>NOTE 1 This mode is the preferred method for normal makeup to the Volume Control Tank in Modes 1 and 2.</p> <p>NOTE 2 Sample may be obtained at normal RCS sample intervals provided the unit is at power and the unit response following the makeup is as expected.</p> <p>NOTE 3 Flow oscillations and/or erratic controller response may require manual operation of Boric Acid Flow Controller [FC-62-139] and Primary Water Flow Controller [FC-62-142] until stable conditions exist.</p> <p>NOTE 4 Minor variations in the adjustments for the primary water and boric acid valve controllers can result in small reactivity changes. Following makeup to the VCT, adjustments in control rod position may be required to maintain power and temperature.</p> <p>NOTE 5 In Modes 1 and 2, manual Makeup should be accomplished in batches of approximately 10% VCT volume (200 gallons) or less.</p>	
<p><u>STEP 2.:</u> [1] ENSURE unit is NOT in a Tech Spec or TRM action that prohibits positive reactivity additions.</p> <p><u>STANDARD:</u> Operator refers to turnover information and determines that there are no Tech Spec or TRM actions that would prohibit positive reactivity additions.</p> <p><u>COMMENTS:</u></p>	<p>_____ SAT</p> <p>_____ UNSAT</p>
<p>Evaluators Note: The following Notes proceed step [2]</p>	
<p>Note 1 If available, it is preferred that the REACT software be used to perform the following two steps.</p> <p>Note 2 REACTF data sheets are to be signed by the preparer and reviewer.</p>	

Job Performance Checklist:

STEP / STANDARD	SAT / UNSAT
<p><u>STEP 3.:</u> [2] CALCULATE Boric Acid and Primary Water Integrator settings for manual makeup by: (N/A method not chosen)</p> <p> [a] PERFORM calculation using REACTF software.</p> <p> OR</p> <p> [b] PERFORM calculation using Appendix C.</p> <p><u>STANDARD:</u> Operator N/As step [2] [a] based on turnover information and goes to Appendix C.</p> <p><u>COMMENTS:</u></p>	<p>_____ SAT</p> <p>_____ UNSAT</p>
<p>NOTE: Following steps are contained in 0-SO-62-7, Appendix C CALCULATION OF BORIC ACID AND PRIMARY WATER INTEGRATOR SETTING FOR MANUAL MAKEUP TO VCT (RCS).</p>	
<p><u>STEP 4.:</u> [1] OBTAIN Current RCS Boric Acid Concentration</p> <p><u>STANDARD:</u> Operator obtains current RCS boron concentration by referring to initial conditions. RCS Boric Acid Concentration 755 ppm</p> <p><u>COMMENTS:</u></p>	<p>_____ SAT</p> <p>_____ UNSAT</p>
<p><u>STEP 5.:</u> [2] OBTAIN Current BAT Boric Acid Concentration</p> <p><u>STANDARD:</u> Operator obtains current BAT boron concentration by referring to initial conditions. BAT Boric acid concentration 6570 ppm.</p> <p><u>COMMENTS:</u></p>	<p>_____ SAT</p> <p>_____ UNSAT</p>

Job Performance Checklist:

STEP / STANDARD	SAT / UNSAT
<p>STEP 6.: [3] OBTAIN B-10 depletion value from Rx Eng Information page _____ PPM</p> <p>Cue: <i>If operator contacts Rx Engineering, current boron depletion value is 48 ppm.</i></p> <p>STANDARD: Operator obtains current boron depletion value by referring to initial conditions. Boron depletion 48 ppm.</p> <p>COMMENTS:</p>	<p>_____ SAT</p> <p>_____ UNSAT</p>
<p>NOTE Result in Step [4] should be rounded to the second decimal place.</p>	
<p>STEP 7.: [4] CALCULATE BAT Boric Acid Concentration Ratio (BACR):</p> <p>NOTE: 6820 ppm ÷ 6570 ppm (from step 2) = 1.038 (value rounded to second decimal = 1.04)</p> <p>STANDARD: Operator determines that BACR is 1.038, applies the preceding note and rounds the value to 1.04</p> <p>This step is critical for the operator to apply the preceding note to be able to correctly calculate the BACR.</p> <p>COMMENTS:</p>	<p>___ SAT</p> <p>___ UNSAT</p> <p>Critical Step</p>
<p>STEP 8.: [5] CALCULATE B-10 corrected boron concentration:</p> $\frac{\text{STEP [1]}}{\text{STEP [3]}} - \frac{\text{STEP [3]}}{\text{STEP [3]}} = \text{B-10 corrected boron}$ <p>NOTE: 755 ppm – 48 ppm (from step 2) = 707 ppm</p> <p>STANDARD: Operator calculates corrected B-10 concentration</p> <p>COMMENTS:</p>	<p>_____ SAT</p> <p>_____ UNSAT</p>

Job Performance Checklist:

STEP / STANDARD	SAT / UNSAT
<p>STEP 9.: [6] DETERMINE Corrected Boric Acid Flow Rate and Controller Setting using appropriate table from TI-44 Appendix C.</p> <p> [a] RECORD Corrected Boric Acid Flow Rate from TI-44 Appendix C <u>Table 1</u>.</p> <p>NOTE: TI-44, Appendix C, Table 1, BA flow for 700 ppm is 7.92 gpm; BA flow rate for 710 ppm is 8.05 gpm. Interpolating for 707 ppm, (8.05-7.92=0.13; 0.13x7/10=0.091; 7.92+0.091=8.011) BA flow rate is <u>8.01 gpm</u></p> <p><u>STANDARD:</u> Operator calculates the Corrected Boric Acid Flow rate from TI-44 Appendix C Table 1. 8.01 gpm</p> <p>Critical step to interpolate the corrected boron flow to ensure correct boron concentration to set the integrator to ensure no unplanned reactivity changes occur.</p> <p><u>COMMENTS:</u></p>	<p>___ SAT</p> <p>___ UNSAT</p> <p>Critical Step</p>
<p>STEP 10.: [b] RECORD Correct Boric Acid Controller Setting from TI-44 Appendix C <u>Table 2</u></p> <p>NOTE: TI-44, Appendix C, Table 2, BA controller setting for 700ppm is 15.8%; BA controller setting for 710 ppm is 16.1%. Interpolating for 1102 ppm, (16.1-15.8=0.3; 0.3x7/10=0.21; 15.8+0.21=16.01%) BA controller setting for is <u>16.0% (16.01%)</u>.</p> <p><u>STANDARD:</u> Operator calculates the Corrected Boric Acid Controller Setting from TI-44 Appendix C Table 2 for a value of 16.0% (16.01%)</p> <p>Critical step to interpolate the corrected boric acid controller setting to ensure correct boron concentration to set the integrator to ensure no unplanned reactivity changes occur.</p> <p><u>COMMENTS:</u></p>	<p>___ SAT</p> <p>___ UNSAT</p> <p>Critical Step</p>

Job Performance Checklist:

STEP / STANDARD	SAT / UNSAT
<p><u>STEP 11.:</u> [7] Calculate Boric Acid Controller Setting</p> <p>[Corrected BA Controller Setting] x [BACR] = [Boric Acid Controller Setting]</p> <p>NOTE: 16.0% (16.01%) x 1.04 (1.038) = 16.64% (16.618%)</p> <p><u>STANDARD:</u> Operator calculates the Boric Acid Controller Setting.</p> <p>Critical step to ensure that the boric acid controller is set to the proper value to ensure that an unanticipated reactivity addition does not occur.</p> <p><u>COMMENT:</u></p>	<p>_____ SAT</p> <p>_____ UNSAT</p> <p>Critical Step</p>
<p><u>STEP 12.:</u> [8] Desired VCT level - Actual VCT level X 20 gal = _____</p> <p>NOTE: (35% - 25%) X 20 gal = 200 gal</p> <p><u>STANDARD:</u> Operator calculates total volume of water to raise VCT level 10% to be 200 gallons.</p> <p><u>COMMENT:</u></p>	<p>_____ SAT</p> <p>_____ UNSAT</p>
<p><u>STEP 13.:</u> [9] [Corrected BA Flow Rate] x [BACR] + 70 gpm = [Total Flow Rate]</p> <p>NOTE: 8.01 gpm x 1.04 + 70 gpm = 78.33 gpm (~ 78.3 gpm)</p> <p><u>STANDARD:</u> Operator determines total flow rate of ~ 78.3 gpm ± 1% (77.52-79.08)</p> <p>Critical step to accurately determine the required total flow rate through the blender to fill the VCT with the correct blended flow.</p> <p><u>COMMENT:</u></p>	<p>_____ SAT</p> <p>_____ UNSAT</p> <p>Critical Step</p>

Job Performance Checklist:

STEP / STANDARD	SAT / UNSAT
<p>STEP 14.: [10] [Corrected BA Flow Rate] X BACR] ÷ Total Flow Rate X Total Volume Change = BORIC ACID INTEGRATOR SETTING</p> <p>NOTE: ((8.01 x 1.04) / (78.33)) x 200 = 21.27 gal. Note tolerance in previous step's calculation. ((8.01 x 1.04) / (77.52)) x 200 = 21.490 gal. (high) ((8.01 x 1.04) / (79.08)) x 200 = 21.068 gal. (low)</p> <p>STANDARD: Operator establishes correct integrator setting as below [8.01 gpm X 1.04 (BACR) ÷ 78.33gpm] X 200 gal = 21.27 gal</p> <p>Critical step to determine the correct value for the boric acid flow integrator to ensure correct amount of boric acid is added.</p> <p>COMMENTS:</p>	<p>_____ SAT</p> <p>_____ UNSAT</p> <p>Critical Step</p>
<p>Evaluator Note: The total of this step and the previous stem should be ~ the same as the total volume needed to raise level 10% in the VCT. (21.27 gals +</p>	
<p>STEP 15.: 70 GPM (PW) ÷ Total Flow Rate X Total Volume = PRIMARY WATER INTEGRATOR SETTING</p> <p>NOTE: (70 gpm / 78.3 gpm) x 200 gals = 178.80 (178.7994) gals</p> <p>STANDARD: Operator establishes correct integrator setting as below [70 gpm ÷ 78.3 gpm] X 200 gal = ~ 178.8 gal (178.7994)</p> <p>Critical step to ensure that primary water integrator is set correctly to stop the blended flow when desired amount of water is added to VCT.</p> <p>COMMENTS:</p> <p>Cue: This completes the JPM.</p>	<p>___ SAT</p> <p>___ UNSAT</p> <p>Critical Step</p> <p>Stop Time_____</p>

Key

SQN 1,2	BORON CONCENTRATION CONTROL	0-SO-62-7 Rev. 58 Page 33 of 201
------------	-----------------------------	----------------------------------------

Unit 1

Date today

6.5 Manual Makeup Control (preferred method for VCT makeup in Modes 1 and 2)

CAUTION Returning the Boric Acid Blender to service after unplugging, cleaning, or maintenance on the Boric Acid System could introduce debris, sludge, air or chunks of solidified boron into the CCP suction resulting in pump damage. Extreme care must be exercised to properly flush the Boric Acid Blender system following an outage. [C.2]

NOTE 1 This mode is the preferred method for normal makeup to the Volume Control Tank in Modes 1 and 2.

NOTE 2 Sample may be obtained at normal RCS sample intervals provided the unit is at power and the unit response following the makeup is as expected.

NOTE 3 Flow oscillations and/or erratic controller response may require manual operation of Boric Acid Flow Controller [FC-62-139] and Primary Water Flow Controller [FC-62-142] until stable conditions exist.

NOTE 4 Minor variations in the adjustments for the primary water and boric acid valve controllers can result in small reactivity changes. Following makeup to the VCT, adjustments in control rod position may be required to maintain power and temperature.

NOTE 5 In Modes 1 and 2, manual Makeup should be accomplished in batches of approximately 10% VCT volume (200 gallons) or less.

~~[1]~~ ENSURE unit is NOT in a Tech Spec or TRM action that prohibits positive reactivity additions. [C.1]

AS

SQN 1,2	BORON CONCENTRATION CONTROL	0-SO-62-7 Rev. 58 Page 34 of 201
------------	-----------------------------	----------------------------------------

Unit 1

Date today

6.5 Manual Makeup Control (preferred method for VCT makeup in Modes 1 and 2)
(Continued)

N/A **NOTE 1** If available, it is preferred that the REACT software be used to perform the following two steps.

N/A **NOTE 2** REACTF data sheets are to be signed by the preparer and reviewer.

[2] CALCULATE Boric Acid and Primary Water Integrator settings for manual makeup by: (N/A method not chosen)

[a] PERFORM calculation using REACTF software. N/A

OR

[b] PERFORM calculation using Appendix C. _____

NOTE 1 Verification is expected to be completely independent. Use of the REACT software will require that one of the signoffs on each printout be N/A'd.

NOTE 2 REACTF data sheets are to be signed by the preparer and reviewer.

NOTE 3 Due to eyeball interpolation the verified calculation may slightly differ from the initial calculation. The following signoff indicates that any differences in the two results have been discussed and are close enough to be considered validated.

[3] VERIFY Boric Acid and Primary Water Integrator settings for manual makeup by: (N/A method not chosen)

[a] PERFORM calculation using REACTF software. _____

SRO

OR

[b] PERFORM calculation using Appendix H _____

SRO

[4] ENSURE makeup system aligned for **AUTO** operation in accordance with Section 5.1. _____

APPENDIX C
Page 1 of 2

CALCULATION OF BORIC ACID AND PRIMARY WATER INTEGRATOR SETTING FOR MANUAL MAKEUP TO VCT (RCS)

[1] **OBTAIN** Current RCS Boric Acid Concentration 755 PPM

[2] **OBTAIN** Current BAT Boric Acid Concentration 6570 PPM

[3] **OBTAIN** B-10 depletion value from Rx Eng Information page 48 PPM

NOTE Result in Step [4] should be rounded to second decimal place.

[4] **CALCULATE** BAT Boric Acid Concentration Ratio (BACR): $6820 \text{ ppm} \div \text{Step [2] ppm} = \underline{1.04} \text{ (1.03\%)}$

[5] **CALCULATE** B-10 corrected boron concentration: $\frac{755}{\text{STEP [1]}} - \frac{48}{\text{STEP [3]}} = \underline{707}$ B-10 corrected boron

[6] **DETERMINE** Corrected Boric Acid Flow Rate and Controller Setting using appropriate Table from TI-44 Appendix C.

[a] **RECORD** Corrected Boric Acid Flow Rate from TI-44 Appendix C Table 1. 8.01 GPM

[b] **RECORD** Corrected Boric Acid Controller Setting from TI-44 Appendix C Table 2. 16.0 %

[7] $\left(\frac{16.0}{\text{Corrected Boric Acid Controller Setting (Step [6][b])}} \right) \times \left(\frac{1.04}{\text{BACR from step 4}} \right) = \underline{16.64} \text{ (Boric Acid Controller Setting)}$

CONTINUED ON NEXT PAGE

SQN
1,2

BORON CONCENTRATION CONTROL

0-SO-62-7
Rev. 58
Page 162 of 201

APPENDIX C
Page 2 of 2

$$[8] \quad \left[\frac{35 \%}{\text{(Desired VCT level)}} - \frac{25 \%}{\text{(Actual VCT level)}} \right] \times 20 \text{ GAL/percent} = \frac{200 \text{ GALS}}{\text{Total Volume}}$$

$$[9] \quad \left(\frac{8.01 \text{ GPM}}{\text{Corrected Boric Acid Flow Rate (Step [6][a])}} \right) \times \left(\frac{1.04}{\text{(BACR from step 4)}} \right) + 70 \text{ GPM (Primary H}^2\text{O)} = \frac{78.33 \text{ GPM}}{\text{Total Flow Rate}}$$

$$[10] \quad \left(\frac{8.01 \text{ GPM}}{\text{Corrected Boric Acid Flow Rate (Step [6][a])}} \right) \times \left(\frac{1.04}{\text{(BACR from step 4)}} \right) \div \frac{78.33 \text{ GPM}}{\text{(Obtained from step [9])}} \times \frac{200 \text{ GALS}}{\text{(Obtained from step [8])}} =$$

21.27 GALS
BORIC ACID INTEGRATOR
SETTING

$$[11] \quad 70 \text{ GPM (Primary H}^2\text{O)} \div \frac{78.3 \text{ GPM}}{\text{Total Flow Rate (Obtained from step [9])}} \times \frac{200 \text{ GALS}}{\text{Total Volume (Obtained from step [8])}} = \frac{178.8 \text{ GALS}}{\text{PRIMARY WATER INTEGRATOR SETTING}}$$

Calculation check: Step [10] results + Step [11] results should ~ = Step [8] results

READ TO OPERATOR

DIRECTIONS TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM. 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. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

1. Unit 1 is at 100% full power, steady state.
2. Reactor Coolant System boron concentration is 755 ppm.
3. BAT boron concentration is 6570 ppm.
4. B-10 depletion value is 48 ppm.
5. REACTF software is not available for boron calculations.

INITIATING CUES:

1. You are the Unit 1 OATC and are to perform a calculation for a manual blended makeup to the Chemical Volume Control System in accordance with 0-SO-62-7, "Boron Concentration Control," to increase Volume Control Tank level from 25% to 35%.
2. Notify the SRO when the calculation is completed.



Sequoyah Nuclear Plant

Unit 0

Technical Instruction

TI-44

BORON TABLES

Revision 0012

Quality Related

Level of Use: Information Use

Effective Date: 03-02-2007

Responsible Organization: RXE, Reactor Engineering

Prepared By: Jerry Espy

Approved By: Charles Griffin

SQN Unit 0	BORON TABLES	TI-44 Rev. 0012 Page 2 of 25
----------------------	---------------------	------------------------------------

Current Revision Description

This procedure was converted from Word 95 to Word 2002 (XP) using revision 0011.
Deleted table 11 for 580 deg F for PER 113152 action 1.

Table of Contents

1.0	INTRODUCTION	5
1.1	Purpose	5
1.2	Scope.....	5
1.3	Frequency and Conditions	5
1.4	Background.....	6
1.4.1	General.....	6
1.4.2	Method.....	6
2.0	REFERENCES	7
2.1	Performance References	7
2.2	Developmental References.....	7
3.0	PRECAUTIONS AND LIMITATIONS	7
4.0	PREREQUISITE ACTIONS	7
5.0	ACCEPTANCE CRITERIA.....	7
6.0	PERFORMANCE.....	8
6.1	Boron Tables	8
6.2	Pressurizer Level Factor	9
6.3	Examples	10
6.3.1	Boration	10
6.3.2	Dilution.....	10
6.3.3	Pressurizer Level Correction.....	11
7.0	POST PERFORMANCE ACTIVITIES	11
8.0	RECORDS.....	11
Appendix A:	SEQUOYAH PRIMARY SYSTEM VOLUME	12
Appendix B:	DERIVATION OF EQUATION.....	13
Appendix C:	REACTOR COOLANT BORON CONCENTRATION VS. AUTOMATIC MAKEUP FLOW RATE PRIMARY WATER FLOW OF 70 GPM	16

SQN Unit 0	BORON TABLES	TI-44 Rev. 0012 Page 4 of 25
-----------------------	---------------------	---------------------------------------------

Table of Contents (continued)

ATTACHMENTS

Attachment 1:	Table 1
Attachment 2	Table 2
Attachment 3	Table 3
Attachment 4	Table 4
Attachment 5	Table 5
Attachment 6	Table 6
Attachment 7	Table 7
Attachment 8	Table 8
Attachment 9	Table 9
Attachment 10	Table 10

1.0 INTRODUCTION

1.1 Purpose

This Instruction provides steps for the use of the boron tables to ensure the desired boron concentration in the RCS can be achieved.

1.2 Scope

The boron tables consists of the following:

<u>TABLE</u>	<u>RCS</u> <u>PRESSURE</u>	<u>RCS</u> <u>TEMP</u>	<u>PZR</u> <u>LEVEL</u>
1	0.0	70	24.7
2	0.0	100	24.7
3	200	200	24.7
4	400	300	24.7
5	2235	400	24.7
6	2235	500	24.7
7	2235	547	24.7
8	2235	555	33.8
9	2235	565	45.1
10	2235	578.2	60.0

1.3 Frequency and Conditions

This Instruction is applicable in all Modes.

1.4 Background

1.4.1 General

During reactor operations it is necessary to change the boron concentration in the reactor coolant system (RCS) either by adding boric acid (boration) or adding primary water (dilution). The boron tables are designed to show, given the initial boron concentration, the volume of boric acid or primary water to inject into the RCS to produce a desired boron concentration.

Boron concentration in the RCS is adjusted either by diluting the primary fluid with makeup water or borating with a 4% boric acid solution. These dilution or boration volumes may be determined through the use of tables.

The boron tables are generated as output of the BORON code. These tables give the required addition volumes versus the current RCS boron concentration and the desired concentration change. Some interpolation may be necessary in using the tables.

1.4.2 Method

The addition volumes are calculated in accordance with the equation below. The derivation of the equation is given in Appendix B:

$$V_n = V_p * f * \ln \left[\frac{C_n - C_p}{C_n - (C_p + \Delta C)} \right] \quad \text{where: } V_p = \text{RCS volume (gallon)}$$

$V_n =$ Additional volumes (gallon) ($n =$ Boration or Dilution)

$C_p =$ Initial Primary System boron concentration (ppm)

$C_n =$ Boron concentration of addition volume (ppm)
(Either boric acid solution or makeup water)

$\Delta C =$ Change in primary system boron concentration (ppm)

$f =$ Ratio of primary to addition volume densities (frac)
(ρ_p / ρ_M)

The primary system volume is determined through the use of a best estimate system model, shown in Appendix A. In this model, the primary system volume agrees with the FSAR value for Hot Full Power (HFP) conditions. A constant pressurizer level of 24.7% span is assumed at and below Hot Zero Power (HZP) condition. Thermal expansion effects are included in this model.

SQN Unit 0	BORON TABLES	TI-44 Rev. 0012 Page 7 of 25
-----------------------	---------------------	---------------------------------------------

2.0 REFERENCES

2.1 Performance References

- A. 0-SO-62-7, Boron Concentration Control.

2.2 Developmental References

- A. "Nuclear Fuel Management", by Graves.

3.0 PRECAUTIONS AND LIMITATIONS

- A. Automatic make up uses a primary water flow setting of 70 gpm. Appendix C boric acid flow rates are based on 70 gpm primary water flow. These setting are based on the boron concentration of 6820 ppm. There may be small deviations in makeup if the BAT tanks are at a lower concentration. Linear interpolation can be used for boron concentration between table values.
- B. The REACTINF/REACTF computer programs is to be used for concentration adjustments greater than 2500 PPM.
- C. Small makeup deviations should be expected due to integrator inaccuracies, small errors in design mispredictions, BAT tank concentration difference than tables, calculation inaccuracies.

4.0 PREREQUISITE ACTIONS

None

5.0 ACCEPTANCE CRITERIA

None

6.0 PERFORMANCE

NOTE

The REACTINF/REACTF computer programs is to be used for concentration adjustments greater than 2500 PPM and may be used in lieu of the tables for other boron concentration adjustments.

6.1 Boron Tables

NOTE

The boron tables (Table 1 thru Table 10) give the addition volumes as a function of the current RCS boron concentration and the desired change in boron concentration.

- [1] **FIND** the table corresponding to the current RCS temperature (T_{ave}) and pressure.
- [2] **FIND** the current RCS boron concentration in the left-hand column. (The given concentrations cover the range of 0 to 2500 ppm.)

NOTE

Some interpolation from the tables may be necessary.

- [3] **FIND** the addition volume by going across to the correct column. (The volumes are given for concentration changes from 1 to 100 ppm. The boric acid solution volume is found in the BORATE subcolumn and the makeup water volume is found in the DILUTE subcolumn.)

NOTE

Appendix C is to be used for guidance in setting the automatic flow controller.

6.2 Pressurizer Level Factor

NOTE

The tables have been constructed for the specified system volume model. However, if an unusual pressurizer level is encountered, the values given by the tables can be corrected for pressurizer level. The factor to be used is calculated as:

$$\text{FACTOR} = \frac{77851.5 + 134.65 \times (\text{PLEVEL}) + 5.236 \times (T)}{(V_p)}$$

Where: PLEVEL = Desired Pressurizer Level in Percent (%)

T = System Temperature Tave (°F)

V_p = System Volume (gallons)

V_p can be found in Appendix A at temperature T.

- [1] **IF** the initial boron concentration and desired concentration change are known, **THEN**
- [1.1] **FIND** the boration or dilution volume (V_B or V_D) for the given conditions using the tables.
- [1.2] **MULTIPLY** the addition volume V_B or V_D by the **FACTOR** calculated above:
- $$V_B^* = V_B \text{ FACTOR}$$
- $$\text{and } V_D^* = V_D \text{ FACTOR}$$
- where V_B^* or V_D^* is the corrected addition volume
- [2] **IF** the initial boron concentration and the boration or dilution volume (V_B or V_D) are known, **THEN**
- [2.1] **DIVIDE** the addition volume V_B or V_D by the **FACTOR** calculated above:
- $$V_B^{**} = V_B / \text{FACTOR}$$
- $$\text{and } V_D^{**} = V_D / \text{FACTOR}$$
- [2.2] **USE** the tables to find the desired concentration change for the given initial boron concentration and the corrected addition volume V_B^{**} or V_D^{**}

6.3 Examples

NOTE

The following examples use both methods discussed previously.

6.3.1 Boration

- [1] **DETERMINE** the required boric acid solution volume to increase the primary boron concentration from 1200 ppm to 1230 ppm at HFP conditions (578.2°F):

Boron tables:

$$\Delta C = 1230 - 1200 = 30$$

At 1200 ppm boron:

$$\text{for } \Delta C = 20 \text{ ppm } V_B = 226.1 \text{ gallons}$$

At 1220 ppm boron:

$$\text{for } \Delta C = 10 \text{ ppm } V_B = 113.4 \text{ gallons}$$

$$V_B = 226.1 + 113.4 = 339.5 \text{ gallons boric acid solution}$$

6.3.2 Dilution

- [1] **DETERMINE** the required makeup water volume to dilute the primary boron concentration from 900 to 850 ppm when the RCS is at 500°F:

Boron tables:

$$\Delta C = 900 - 850 = 50 \text{ ppm}$$

At 900 ppm boron:

$$\text{for } \Delta C = 50 \text{ ppm } V_D = 3833.0$$

$$V_D = \underline{3833.0} \text{ gallons makeup water}$$

SQN Unit 0	BORON TABLES	TI-44 Rev. 0012 Page 11 of 25
-----------------------------	---------------------	----------------------------------------------------------

6.3.3 Pressurizer Level Correction

- [1] **DETERMINE** the required boric acid solution volume to increase the primary boron concentration from 1000 to 1100 ppm at 200°F with a pressurizer level of 85% span:

Boron tables:

$$\Delta C = 1100 - 1000 = 100 \text{ ppm}$$

At 1000 ppm boron:

$$\text{for } \Delta C = 100 \text{ ppm } V_B = 1363.4 \text{ gallons}$$

$$V_B = 1363.4 \text{ gallons}$$

$$\text{FACTOR} = \frac{77851.5 + 134.65 \times (85.) + 5.236 \times (200)}{82654.0}$$

$$= 1.093$$

$$V_B^* = 1363.4 \times 1.093 = 1490.2 \text{ gallons boric acid solution}$$

7.0 POST PERFORMANCE ACTIVITIES

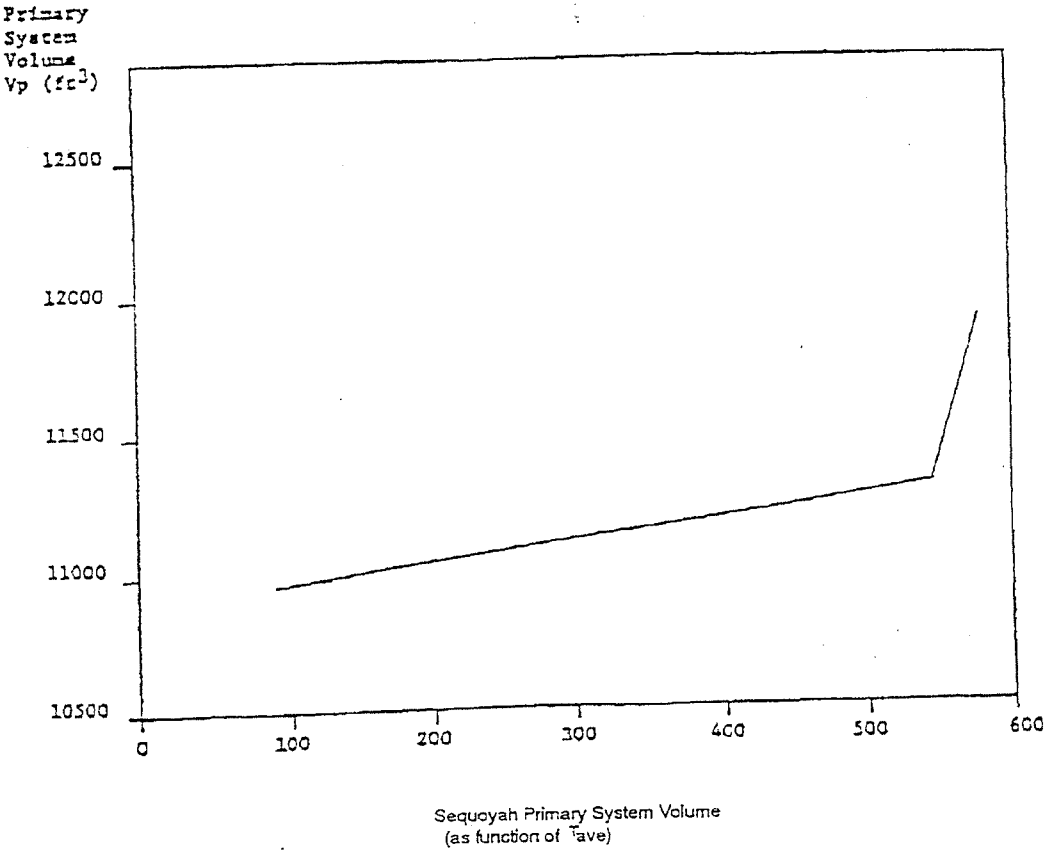
None

8.0 RECORDS

None

Appendix A
(Page 1 of 1)

SEQUOYAH PRIMARY SYSTEM VOLUME
(as function of T_{ave})



SQN Unit 0	BORON TABLES	TI-44 Rev. 0012 Page 13 of 25
-----------------------------	---------------------	----------------------------------------------------------

Appendix B
(Page 1 of 3)

DERIVATION OF EQUATION

The equation in Section 1.4.2 is derived by considering a mass balance on the boron in the RCS. The boron mass in the RCS is given by equation B-1 below:

Equation B-1

$$m = \rho_p V_p C_p \quad \text{where: } m = \text{mass of boron in RCS (10}^{-6} \text{ lbm)}$$

- $\rho_p =$ RCS fluid density (lbm/ft³)
- $V_p =$ RCS fluid volume (ft³)
- $C_p =$ RCS boron concentration (ppm)

The first derivative gives the rate of change of the RCS boron mass, as shown in equation B-2 below:

Equation B-2

$$\frac{dm}{dt} = \rho_p V_p \frac{dC_p}{dt}$$

Similarly, the rate of change of boron in the RCS is found in equation B-3, a mass balance, where boron is added by makeup and removed by letdown:

Equation B-3

$$\frac{dm}{dt} = \frac{dm_m}{dt} - \frac{dm_l}{dt}$$

where: $\frac{dm_m}{dt}$ = rate of boron addition due to makeup flow

$\frac{dm_l}{dt}$ = rate of boron removal due to letdown flow

Appendix B
(Page 2 of 3)

The makeup and letdown boron changes are defined as Equation B-4 and B-5, respectively:

Equation B-4

$$\frac{dm_m}{dt} = \rho_m \dot{V}_m C_m$$

Equation B-5

$$\frac{dm_l}{dt} = \rho_l \dot{V}_l C_l$$

where: ρ_m, ρ_l = fluid density of makeup or letdown (lbm/ft³)
 \dot{V}_m, \dot{V}_l = volumetric flowrate of makeup or letdown (ft³/min)
 C_m, C_l = boron concentration of makeup or letdown (ppm)

Substituting B-2, B-4, and B-5 into B-3, we obtain

Equation B-6:

Equation B-6

$$\rho_p V_p \frac{dC_p}{dt} = \rho_m V_m C_m - \rho_l V_l C_l$$

In order to proceed, Equation B-6 must be simplified by making the following assumptions:

- [1] The mass flow rate entering the RCS is equal to the mass flow rate leaving ($\rho_m V_m = \rho_l V_l$), and
- [2] Fluid properties in the letdown stream are identical to those found in the RCS ($\rho_l = \rho_p$ and $C_l = C_p$).

SQN Unit 0	BORON TABLES	TI-44 Rev. 0012 Page 15 of 25
---------------	--------------	-------------------------------------

**Appendix B
(Page 3 of 3)**

This yields equation B-7:

Equation B-7

$$\rho_p V_p \frac{dC_p}{dt} = \rho_m \dot{V}_m (C_m - C_p)$$

With the exception of the density terms, this equation is similar to Equation 5-16 found in Nuclear Fuel Management by Graves. The densities above are found since the boron concentration, C, is in terms of a mass fraction (ppm) in lieu of the more common units of weight per unit volume.

The final result is found by rearranging and integrating the resulting equation:

$$\dot{V}_m dt = \frac{\rho_p}{\rho_m} V_p \frac{dC_p}{C_m - C_p} \text{ therefore}$$

$$\int_{t_0}^{t_0 + \Delta t} \dot{V}_m dt = \frac{\rho_p}{\rho_m} V_p \int_{C_0}^{C_0 + \Delta C} \frac{dC_p}{C_m - C_p}$$

and

$$\dot{V}_m(t) \Big|_{t_0}^{(t_0 + \Delta t)} = - \frac{\rho_p}{\rho_m} V_p \ln(C_m - C_p) \Big|_{C_0}^{C_0 + \Delta C} \text{ or}$$

$$V_m = \dot{V}_m(t_0 + \Delta t - t_0) = - V_p \frac{\rho_p}{\rho_m} [\ln(C_m - (C_0 + \Delta C)) - \ln(C_m - C_0)]$$

yielding

$$V_m = V_p \frac{\rho_p}{\rho_m} \ln \left[\frac{C_m - C_0}{C_m - (C_0 + \Delta C)} \right]$$

where: $C_0 =$ initial primary system boron concentration (ppm), which is the same as the equation in Section 1.4.2.

Appendix C
(Page 1 of 10)

REACTOR COOLANT BORON CONCENTRATION VS. AUTOMATIC MAKEUP FLOW
RATE PRIMARY WATER FLOW OF 70 GPM

TABLE 1

BORON CONC.	BORIC ACID FLOWRATE (GPM)	BORON CONC.	BORIC ACID FLOWRATE (GPM)
50	.51	380.0	4.09
100	1.03	390.0	4.20
125	1.30	400.0	4.31
150	1.56	410.0	4.43
200.0	2.09	420.0	4.54
210.0	2.20	430.0	4.66
220.0	2.31	440.0	4.78
230.0	2.42	450.0	4.89
240.0	2.53	460.0	5.01
250.0	2.64	470.0	5.13
260.0	2.74	480.0	5.24
270.0	2.85	490.0	5.36
280.0	2.96	500.0	5.48
290.0	3.08	510.0	5.60
300.0	3.19	520.0	5.72
310.0	3.30	530.0	5.83
320.0	3.41	540.0	5.95
330.0	3.52	550.0	6.07
340.0	3.63	560.0	6.19
350.0	3.75	570.0	6.32
360.0	3.86	580.0	6.44
370.0	3.97	590.0	6.56

Appendix C
(Page 2 of 10)

REACTOR COOLANT BORON CONCENTRATION VS. AUTOMATIC MAKEUP FLOW
RATE PRIMARY WATER FLOW OF 70 GPM

TABLE 1

BORON CONC.	BORIC ACID FLOWRATE (GPM)	BORON CONC.	BORIC ACID FLOWRATE (GPM)
600.0	6.68	800.0	9.20
610.0	6.80	810.0	9.33
620.0	6.92	820.0	9.46
630.0	7.05	830.0	9.59
640.0	7.17	840.0	9.73
650.0	7.29	850.0	9.86
660.0	7.42	860.0	9.99
670.0	7.54	870.0	10.12
680.0	7.67	880.0	10.26
690.0	7.79	890.0	10.39
700.0	7.92	900.0	10.53
710.0	8.05	910.0	10.66
720.0	8.17	920.0	10.80
730.0	8.30	930.0	10.93
740.0	8.43	940.0	11.07
750.0	8.56	950.0	11.21
760.0	8.68	960.0	11.34
770.0	8.81	970.0	11.48
780.0	8.94	980.0	11.62
790.0	9.07	990.0	11.76

Appendix C
(Page 3 of 10)

**REACTOR COOLANT BORON CONCENTRATION VS. AUTOMATIC MAKEUP FLOW
RATE PRIMARY WATER FLOW OF 70 GPM**

TABLE 1

BORON CONC.	BORIC ACID FLOWRATE (GPM)	BORON CONC.	BORIC ACID FLOWRATE (GPM)
1000.0	11.90	1200.0	14.78
1010.0	12.04	1210.0	14.93
1020.0	12.18	1220.0	15.08
1030.0	12.32	1230.0	15.24
1040.0	12.46	1240.0	15.39
1050.0	12.60	1250.0	15.54
1060.0	12.74	1260.0	15.69
1070.0	12.88	1270.0	15.84
1080.0	13.03	1280.0	15.99
1090.0	13.17	1290.0	16.15
1100.0	13.32	1300.0	16.31
1110.0	13.46	1310.0	16.46
1120.0	13.61	1320.0	16.62
1130.0	13.75	1330.0	16.77
1140.0	13.90	1340.0	16.93
1150.0	14.04	1350.0	17.09
1160.0	14.19	1360.0	17.25
1170.0	14.34	1370.0	17.41
1180.0	14.49	1380.0	17.56
1190.0	14.64	1390.0	17.72

Appendix C
(Page 4 of 10)

**REACTOR COOLANT BORON CONCENTRATION VS. AUTOMATIC MAKEUP FLOW
RATE PRIMARY WATER FLOW OF 70 GPM**

TABLE 1

BORON CONC.	BORIC ACID FLOWRATE (GPM)	BORON CONC.	BORIC ACID FLOWRATE (GPM)
1400.0	17.88	1600.0	21.22
1410.0	18.05	1610.0	21.40
1420.0	18.21	1620.0	21.57
1430.0	18.37	1630.0	21.75
1440.0	18.53	1640.0	21.92
1450.0	18.70	1650.0	22.10
1460.0	18.86	1660.0	22.27
1470.0	19.02	1670.0	22.45
1480.0	19.19	1680.0	22.63
1490.0	19.36	1690.0	22.81
1500.0	19.52	1700.0	22.99
1510.0	19.69	1710.0	23.17
1520.0	19.86	1720.0	23.35
1530.0	20.03	1730.0	23.53
1540.0	20.20	1740.0	23.72
1550.0	20.36	1750.0	23.90
1560.0	20.54	1760.0	24.08
1570.0	20.71	1770.0	24.27
1580.0	20.88	1780.0	24.45
1590.0	21.05	1790.0	24.64

Appendix C
(Page 5 of 10)

**REACTOR COOLANT BORON CONCENTRATION VS. AUTOMATIC MAKEUP FLOW
RATE PRIMARY WATER FLOW OF 70 GPM**

TABLE 1

BORON CONC.	BORIC ACID FLOWRATE (GPM)	BORON CONC.	BORIC ACID FLOWRATE (GPM)
1800.0	24.83	2060.0	29.97
1810.0	25.01	2070.0	30.17
1820.0	25.20	2080.0	30.38
1830.0	25.39	2090.0	30.59
1840.0	25.58	2100.0	30.81
1850.0	25.77	2150.0	31.89
1860.0	25.96	2200.0	32.97
1870.0	26.16	2250.0	34.10
1880.0	26.35	2300.0	35.23
1890.0	26.54	2350.0	36.41
1900.0	26.74	2400.0	37.60
1910.0	26.93	2450.0	38.84
1920.0	27.13	2500.0	40.07
1930.0	27.33	2550.0	41.37
1940.0	27.53	2600.0	42.66
1950.0	27.72	2650.0	44.02
1960.0	27.92	2700.0	45.37
1970.0	28.12	2750.0	46.80
1980.0	28.33	2800.0	48.23
1990.0	28.53	2860.0	50.00
2000.0	28.73		
2010.0	28.93		
2020.0	29.14		
2030.0	29.34		
2040.0	29.55		
2050.0	29.76		

Appendix C
(Page 6 of 10)

**REACTOR COOLANT BORON CONCENTRATION VS. AUTOMATIC MAKEUP FLOW
RATE PRIMARY WATER FLOW OF 70 GPM**

TABLE 2

BORON CONC.	BORIC ACID CONTROLLER SETTING (%)	BORON CONC.	BORIC ACID CONTROLLER SETTING (%)
50	1.0	380.0	8.2
100	2.1	390.0	8.4
125	2.6	400.0	8.6
150	3.1	410.0	8.9
200.0	4.2	420.0	9.1
210.0	4.4	430.0	9.3
220.0	4.6	440.0	9.6
230.0	4.8	450.0	9.8
240.0	5.1	460.0	10.0
250.0	5.3	470.0	10.3
260.0	5.5	480.0	10.5
270.0	5.7	490.0	10.7
280.0	5.9	500.0	11.0
290.0	6.2	510.0	11.2
300.0	6.4	520.0	11.4
310.0	6.6	530.0	11.7
320.0	6.8	540.0	11.9
330.0	7.0	550.0	12.1
340.0	7.3	560.0	12.4
350.0	7.5	570.0	12.6
360.0	7.7	580.0	12.9
370.0	7.9	590.0	13.1

Appendix C
(Page 7 of 10)

**REACTOR COOLANT BORON CONCENTRATION VS. AUTOMATIC MAKEUP FLOW
RATE PRIMARY WATER FLOW OF 70 GPM**

TABLE 2

BORON CONC.	BORIC ACID CONTROLLER SETTING (%)	BORON CONC.	BORIC ACID CONTROLLER SETTING (%)
600.0	13.4	800.0	18.4
610.0	13.6	810.0	18.7
620.0	13.8	820.0	18.9
630.0	14.1	830.0	19.2
640.0	14.3	840.0	19.5
650.0	14.6	850.0	19.7
660.0	14.8	860.0	20.0
670.0	15.1	870.0	20.3
680.0	15.3	880.0	20.5
690.0	15.6	890.0	20.8
700.0	15.8	900.0	21.1
710.0	16.1	910.0	21.3
720.0	16.3	920.0	21.6
730.0	16.6	930.0	21.9
740.0	16.9	940.0	22.1
750.0	17.1	950.0	22.4
760.0	17.4	960.0	22.7
770.0	17.6	970.0	23.0
780.0	17.9	980.0	23.2
790.0	18.1	990.0	23.5

Appendix C
(Page 8 of 10)

REACTOR COOLANT BORON CONCENTRATION VS. AUTOMATIC MAKEUP FLOW
RATE PRIMARY WATER FLOW OF 70 GPM

TABLE 2

BORON CONC.	BORIC ACID CONTROLLER SETTING (%)	BORON CONC.	BORIC ACID CONTROLLER SETTING (%)
1000.0	23.8	1200.0	29.6
1010.0	24.1	1210.0	29.9
1020.0	24.4	1220.0	30.2
1030.0	24.6	1230.0	30.5
1040.0	24.9	1240.0	30.8
1050.0	25.2	1250.0	31.1
1060.0	25.5	1260.0	31.4
1070.0	25.8	1270.0	31.7
1080.0	26.1	1280.0	32.0
1090.0	26.3	1290.0	32.3
1100.0	26.6	1300.0	32.6
1110.0	26.9	1310.0	32.9
1120.0	27.2	1320.0	33.2
1130.0	27.5	1330.0	33.5
1140.0	27.8	1340.0	33.9
1150.0	28.1	1350.0	34.2
1160.0	28.4	1360.0	34.5
1170.0	28.7	1370.0	34.8
1180.0	29.0	1380.0	35.1
1190.0	29.3	1390.0	35.4

Appendix C
(Page 9 of 10)

**REACTOR COOLANT BORON CONCENTRATION VS. AUTOMATIC MAKEUP FLOW
RATE PRIMARY WATER FLOW OF 70 GPM**

TABLE 2

BORON CONC.	BORIC ACID CONTROLLER SETTING (%)	BORON CONC.	BORIC ACID CONTROLLER SETTING (%)
1400.0	35.8	1600.0	42.4
1410.0	36.1	1610.0	42.8
1420.0	36.4	1620.0	43.1
1430.0	36.7	1630.0	43.5
1440.0	37.1	1640.0	43.8
1450.0	37.4	1650.0	44.2
1460.0	37.7	1660.0	44.5
1470.0	38.0	1670.0	44.9
1480.0	38.4	1680.0	45.3
1490.0	38.7	1690.0	45.6
1500.0	39.0	1700.0	46.0
1510.0	39.4	1710.0	46.3
1520.0	39.7	1720.0	46.7
1530.0	40.1	1730.0	47.1
1540.0	40.4	1740.0	47.4
1550.0	40.7	1750.0	47.8
1560.0	41.1	1760.0	48.2
1570.0	41.4	1770.0	48.5
1580.0	41.8	1780.0	48.9
1590.0	42.1	1790.0	49.3

Appendix C
(Page 10 of 10)

**REACTOR COOLANT BORON CONCENTRATION VS. AUTOMATIC MAKEUP FLOW
RATE PRIMARY WATER FLOW OF 70 GPM**

TABLE 2

BORON CONC.	BORIC ACID CONTROLLER SETTING (%)	BORON CONC.	BORIC ACID CONTROLLER SETTING (%)
1800.0	49.7	2060.0	59.9
1810.0	50.0	2070.0	60.3
1820.0	50.4	2080.0	60.8
1830.0	50.8	2090.0	61.2
1840.0	51.2	2100.0	61.6
1850.0	51.5	2150.0	63.8
1860.0	51.9	2200.0	65.9
1870.0	52.3	2250.0	68.2
1880.0	52.7	2300.0	70.5
1890.0	53.1	2350.0	72.8
1900.0	53.5	2400.0	75.2
1910.0	53.9	2450.0	77.7
1920.0	54.3	2500.0	80.1
1930.0	54.7	2550.0	82.7
1940.0	55.1	2600.0	85.3
1950.0	55.4	2650.0	88.0
1960.0	55.8	2700.0	90.8
1970.0	56.2	2750.0	93.6
1980.0	56.7	2800.0	96.5
1990.0	57.1	2860.0	100.0
2000.0	57.5		
2010.0	57.9		
2020.0	58.3		
2030.0	58.7		
2040.0	59.1		
2050.0	59.5		

SEQUOYAH NUCLEAR PLANT JOB PERFORMANCE MEASURE

RO A-2

Boric Acid Storage Tank Level Operability Determination

RO/SRO
JOB PERFORMANCE MEASURE

Task: Determine the Operability of a BAT before use.

Task #: 0040090101 (RO)

Task Standard: Determine the operability of BAST C prior to placing tank in service.

Time Critical Task: YES: _____ NO: X

K/A Reference/Ratings: 2.1.25 (3.9/4.2)
2.2.12 (3.7/4.1)

Method of Testing:

Simulated Performance: _____ Actual Performance: X

Evaluation Method:

Simulator _____ In-Plant _____ Classroom X

Main Control Room _____ Mock-up _____

Performer: _____
Trainee Name

Evaluator: _____ / _____
Name / Signature DATE

Performance Rating: SAT: _____ UNSAT: _____

Validation Time: 15 minutes Total Time: _____

Performance Time: Start Time: _____ Finish Time: _____

COMMENTS

SPECIAL INSTRUCTIONS TO EVALUATOR:

1. Critical steps are identified in step SAT/UNSAT column by bold print 'Critical Step'.
2. Any **UNSAT** requires comments
3. Ensure operator performs the following required actions for **SELF-CHECKING**;
 - a. Identifies the correct unit, train, component, etc.
 - b. Reviews the intended action and expected response.
 - c. Compares the actual response to the expected response.

Tools/Equipment/Procedures Needed:

TRM (TR 3.1.2.5 and 3.1.2.6)
1-SI-OPS-000-003.W
OPSLINKS

References:

	Reference	Title	Rev No.
1.	TRM	Technical Requirements Manual	36
2.	1-SI-OPS-000-003.W	Weekly Shift Log	43
3.	OPSLINKS	Common/Chemistry Information	N/A

=====

READ TO OPERATOR

DIRECTIONS TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM. 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. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

1. Unit 1 is at 100% power
2. Preparations are in progress to place a clearance on BAT A for Maintenance.
3. BAT C level is indicating 9,600 gallons on 0-LI-62-242.

INITIATING CUES:

1. You are the Unit 1 OATC and the US has directed you to determine operability status for BAT C level in accordance with 1-SI-OPS-000-003.W, prior to aligning BAT C to Unit 1.
2. Plot minimum level for operability of BAT C and notify the US of results when determination of operability has been completed.

Job Performance Checklist:

STEP / STANDARD	SAT / UNSAT
<p><u>STEP 1.:</u> Operator goes to 1-SI-OPS-000-003.W to review BAT C Level operability requirements.</p> <p><u>STANDARD:</u> Operator reviews 1-SI-OPS-000-003.W, Appendix A, SR requirements for BAT C level operability.</p> <p><u>COMMENTS:</u></p>	<p>___ SAT</p> <p>___ UNSAT</p> <hr/> <p>Start Time</p>
<p><u>NOTE:</u> Operator may call Chem Lab to verify the latest boron concentration for Unit 1 RWST and BAT C.</p>	
<p><u>STEP 2.:</u> Operator obtains a copy of the Operations Information sheet containing Chemistry data from Computer: 'Opslinks/Common/Chemistry Information' for the current day to determine the boron concentration of BAT C and the Unit 1 RWST or call Chemistry lab to verify U1 RWST and BAT C Boric Acid concentrations.</p> <p>CUE <i>When Operator describes how the chemistry data would be obtained, give them Attachment 1, Operations Information (Chemistry data sheet)</i></p> <p><u>STANDARD:</u> Operator determines the current Boric Acid concentration for BAT C and Unit 1 RWST is 6475 ppm.</p>	<p>_____ SAT</p> <p>_____ UNSAT</p>
<p><u>STEP 3.:</u> Operator utilizes the TRM and reviews TR 3.1.2.6.a.1 requirement for BAT C level.</p> <p>NOTE: Figure 3.1.2.6 has three separate graphs, one for each BAT, operator must select graph for BAT C.</p> <p><u>STANDARD:</u> Operator states the BAT storage system is required to contain a volume of borated water in accordance with Figure 3.1.2.6.</p> <p><u>COMMENT:</u></p>	<p>_____ SAT</p> <p>_____ UNSAT</p>

Job Performance Checklist:

STEP / STANDARD	SAT / UNSAT
<p><u>STEP 4.</u> Operator goes to TRM Figure 3.1.2.6.</p> <p>STANDARD: Operator selects TRM FIGURE 3.1.2.6 for BAT C only, the figure required to be used is found on page 3/ 4 1-10b.</p> <p>This is critical to select the correct figure for BAT C.</p> <p><u>COMMENT:</u></p>	<p>___ SAT</p> <p>___ UNSAT</p> <p>CRITICAL STEP</p>
<p><u>NOTE:</u> Steps to determine Region of Acceptable Operation on FIGURE 3.1.2.6 based on RWST Boric Acid Concentration or BAT Boric Acid concentration may be performed in either order (JPM step 5 and 6).</p>	
<p><u>STEP 5.</u> Operator selects the appropriate line on FIGURE 3.1.2.6 for BAT C, Boric Acid Concentration, as determined from the data on the Operations Information page, Attachment 1, to determine region of acceptable operation.</p> <p>STANDARD: Using Attachment 1, Operations Information, the operator determines the correct Boric Acid Tank Concentration is 6475 ppm. Conservatively the operator can use the 6450 line for determination of Region of Acceptable Operation.</p> <p>Critical step for candidate select correct line on Fig 3.1.2.6 to determine BAT operable.</p> <p><u>COMMENT:</u></p>	<p>___ SAT</p> <p>___ UNSAT</p> <p>CRITICAL STEP</p>
<p><u>STEP 6.</u> Operator selects the appropriate line on FIGURE 3.1.2.6 based on U-1 RWST Boric Acid Concentration as determined from data on the Operations Information page, Attachment 1, to determine the region of acceptable operation.</p> <p>STANDARD: Using the chemistry page the operator determines the correct RWST Concentration is 2580 ppm. Conservatively the candidate selects the line for the RWST labeled 2550 ppm.</p> <p>Critical step to determine the correct RWST boron concentration line.</p> <p><u>COMMENTS:</u></p>	<p>___ SAT</p> <p>___ UNSAT</p> <p>Critical Step</p>

Job Performance Checklist:

STEP / STANDARD	SAT / UNSAT
<p><u>Evaluator Note:</u> If Operator addresses the statement at the bottom of graph concerning the indicated values including the unusable volume and the instrument error, they should explain the contained water volume limits include allowance for water not available and is discussed in the TRM bases.</p>	
<p><u>STEP 7.</u> Operator determines minimum BAT C level in gallons by locating the intersection of the line for the RWST and BAT Boron concentrations and verifying the actual level in the tank is less than the minimum level indicated on TRM page 3/ 4 1-10b Boric Acid Tank Levels.</p> <p><u>STANDARD:</u> Operator determines the minimum BAT level in gallons required for operability is 9850 gallons (+50/-100 gal.) by picking the point the boric acid concentration lines for the RWST and the BAT intersect, and verifying the actual number of gallons is in the Region of Unacceptable Operation.</p> <p>Critical for candidate to correctly determine the BAT C is acceptable</p> <p><u>COMMENT:</u></p>	<p>_____ SAT</p> <p>_____ UNSAT</p> <p>Critical Step</p>
<p><u>STEP 8.</u> Operator notifies the US that the level in the C BAT, 9,600 gallons does not meet the operability requirements for level in accordance with TR 3.1.2.6.a</p> <p><u>STANDARD:</u> SRO is notified that level is BAT is C is not adequate to meet the operability requirements in accordance with the TRM.</p> <p><u>COMMENTS:</u></p> <p><u>Cue:</u> This completes the JPM.</p>	<p>_____ SAT</p> <p>_____ UNSAT</p> <p>_____</p> <p>Stop Time</p>

READ TO OPERATOR

DIRECTIONS TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM. 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. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

1. Unit 1 is at 100% power
2. Preparations are in progress to place a clearance on BAT A for Maintenance.
3. BAT C level is indicating is 9,600 gallons on 0-LI-62-242.

INITIATING CUES:

1. You are the Unit 1 OATC and the US has directed you to determine operability status for BAT C level in accordance with 1-SI-OPS-000-003.W, prior to aligning BAT C to Unit 1.
2. Plot minimum level for operability of BAT C and notify the US of results when determination of operability has been completed.

Attachment 1
Operations Information (Chemistry)
 (Page 1 of 2)

Sample Point	Units	Boron	Date / Time	Goal	Limit
U1 RCS	ppm	1130	11/3/09 1130	Variable	Variable
U2 RCS	ppm	2476	11/4/09 0856	Variable	Variable
U1 RWST	ppm	2580	11/01/09 0330	2550 – 2650	2500 – 2700
U2 RWST	ppm	2608	10/30/09 1114	2550 – 2650	2500 – 2700
BAT A	ppm	6869	11/02/09 0320	Variable	Variable
BAT B	ppm	6932	11/05/09 0202	Variable	Variable
BAT C	ppm	6475	11/02/09 0320	Variable	Variable
U1 CLA #1	ppm	2549	11/02/09 0330	2470-2630	2400-2700
U1 CLA #2	ppm	2558	11/02/09 0336	2470-2630	2400-2700
U1 CLA #3	ppm	2567	11/02/09 0320	2470-2630	2400-2700
U1 CLA #4	ppm	2555	11/02/09 0341	2470-2630	2400-2700
U2 CLA #1	ppm	2539	9/28/09 0947	2470-2630	2400-2700
U2 CLA #2	ppm	2455	10/14/09 1455	2470-2630	2400-2700
U2 CLA #3	ppm	2542	9/28/09 1115	2470-2630	2400-2700
U2 CLA #4	ppm	2534	9/28/09 1023	2470-2630	2400-2700
Spent Fuel Pool	ppm	2285	11/04/09 0215	≥ 2050	≥ 2000
Lithium Results			Date / Time	Goal	Midpoint
U1 RCS Lithium	ppm	4.02	11/3/09 1130	3.84-4.14	3.99
U2 RCS Lithium	ppm	0.09	10/26/09 1615		

Attachment 1
Operations Information (Chemistry)
(Page 2 of 2)

Primary to Secondary Leakrate Information (Total CPM RM-90-99/119)					
Indicator	Units	U1	Date / Time	U2	Date/Time
SI 50 S/G Leakage?	Yes/No	No	11/05/09 0005	No	11/02/09 0001
SI 137.5 CVE Leakrate	gpd	<0.1	11/03/09 0500	<5.0	10/6/09 0310
5 gpd leak equivalent	cpm	71	11/03/09 0500	135	10/03/09 0130
30 gpd leak equivalent	cpm	224	11/03/09 0500	612	10/03/09 0130
50 gpd leak equivalent	cpm	347	11/03/09 0500	993	10/03/09 0130
75 gpd leak equivalent	cpm	500	11/03/09 0500	1469	10/03/09 0130
CVE Air Inleakage	cfm	26	11/03/09 0500	8.4	10/03/09 0130
Bkgd on 99 /119	cpm	40	11/03/09 0500	40	10/03/09 0130
Correlation Factor 99/119	cpm/gpd	6.13	11/03/09 0500	19.06	10/03/09 0130
Steady State conditions are necessary for an accurate determination of leak rate using the CVE Rad Monitor					

READ TO OPERATOR

DIRECTIONS TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM. 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. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

1. Unit 1 is at 100% power
2. Preparations are in progress to place a clearance on BAT A for Maintenance.
3. BAT C level is indicating is 9,600 gallons on 0-LI-62-242.

INITIATING CUES:

1. You are the Unit 1 OATC and the US has directed you to determine operability status for BAT C level in accordance with 1-SI-OPS-000-003.W, prior to aligning BAT C to Unit 1.
2. Plot minimum level for operability of BAT C and notify the US of results when determination of operability has been completed.

**Attachment 1
Operations Information (Chemistry)
(Page 1 of 2)**

Sample Point	Units	Boron	Date / Time	Goal	Limit
U1 RCS	ppm	1130	09/13/10 1130	Variable	Variable
U2 RCS	ppm	2476	09/14/10 0856	Variable	Variable
U1 RWST	ppm	2580	09/13/10 0330	2550 – 2650	2500 – 2700
U2 RWST	ppm	2608	09/13/10 1114	2550 – 2650	2500 – 2700
BAT A	ppm	6869	09/12/10 0320	Variable	Variable
BAT B	ppm	6932	09/15/10 0202	Variable	Variable
BAT C	ppm	6475	09/12/10 0320	Variable	Variable
U1 CLA #1	ppm	2549	09/15/10 0330	2470-2630	2400-2700
U1 CLA #2	ppm	2558	09/15/10 0336	2470-2630	2400-2700
U1 CLA #3	ppm	2567	09/15/10 0320	2470-2630	2400-2700
U1 CLA #4	ppm	2555	09/15/10 0341	2470-2630	2400-2700
U2 CLA #1	ppm	2539	09/18/10 0947	2470-2630	2400-2700
U2 CLA #2	ppm	2455	09/14/10 1455	2470-2630	2400-2700
U2 CLA #3	ppm	2542	09/18/10 1115	2470-2630	2400-2700
U2 CLA #4	ppm	2534	09/18/10 1023	2470-2630	2400-2700
Spent Fuel Pool	ppm	2285	09/10/10 0215	≥ 2050	≥ 2000
Lithium Results			Date / Time	Goal	Midpoint
U1 RCS Lithium	ppm	4.02	09/13/10 1130	3.84-4.14	3.99
U2 RCS Lithium	ppm	0.09	09/12/10 1615		

**Attachment 1
Operations Information (Chemistry)
(Page 2 of 2)**

Primary to Secondary Leakrate Information (Total CPM RM-90-99/119)					
Indicator	Units	U1	Date / Time	U2	Date/Time
SI 50 S/G Leakage?	Yes/No	No	09/15/10 0500	No	11/02/09 0001
SI 137.5 CVE Leakrate	gpd	<0.1	09/13/10 0500	<5.0	10/6/09 0310
5 gpd leak equivalent	cpm	71	09/13/10 0500	135	10/03/09 0130
30 gpd leak equivalent	cpm	224	09/13/10 0500	612	10/03/09 0130
50 gpd leak equivalent	cpm	347	09/13/10 0500	993	10/03/09 0130
75 gpd leak equivalent	cpm	500	09/13/10 0500	1469	10/03/09 0130
CVE Air Inleakage	cfm	26	09/13/10 0500	8.4	10/03/09 0130
Bkgd on 99 /119	cpm	40	09/13/10 0500	40	10/03/09 0130
Correlation Factor 99/119	cpm/gpd	6.13	09/13/10 0500	19.06	10/03/09 0130
Steady State conditions are necessary for an accurate determination of leak rate using the CVE Rad Monitor					

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

TR 3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

- a. A boric acid storage system with:
 1. A minimum contained borated water volume of 6400 gallons,
 2. Between 6120 and 6990 ppm of boron, and
 3. A minimum solution temperature of 63°F.

- b. The refueling water storage tank with:
 1. A minimum contained borated water volume of 55,000 gallons,
 2. A minimum boron concentration of 2500 ppm, and
 3. A minimum solution temperature of 60°F.

APPLICABILITY: MODES 4, 5 and 6.

ACTION:

MODE 4 - With no borated water source OPERABLE, suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of Technical Specification LCO 3.1.1.1.

MODE 5 - With no borated water source OPERABLE, suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of Technical Specification LCO 3.1.1.2.

MODE 6 - With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS and suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet Technical Specification LCO 3.9.1.

SURVEILLANCE REQUIREMENTS

TR 4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

- a. For the boric acid storage system, when it is the source of borated water by:
 1. Verifying the boron concentration at least once per 7 days,
 2. Verifying the borated water volume at least once per 7 days, and

REACTIVITY CONTROL SYSTEMS

MONITORING REQUIREMENTS (Continued)

3. Verifying the boric acid storage tank solution temperature is greater than or equal to 63°F at least once per 7 days by verifying the area temperature to be greater than or equal to 63°F, or
 4. When the boric acid tank area temperature is less than 63°F and the boric acid storage system being used as the source of borated water, within 6 hours and every 24 hours thereafter, verify the boric acid tank solution temperature to be greater than or equal to 63°F until the boric acid tank area temperature has returned to greater than or equal to 63°F.
- b. For the refueling water storage tank by:
1. Verifying the boron concentration at least once per 7 days,
 2. Verifying the borated water volume at least once per 7 days, and
 3. Verifying the solution temperature at least once per 24 hours while in Mode 4 or while in Modes 5 or 6 when it is the source of borated water.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

TR 3.1.2.6 As a minimum, the following borated water source(s) shall be OPERABLE as required by TR 3.1.2.2:

- a. A boric acid storage system with:
 - 1. A contained volume of borated water in accordance with Figure 3.1.2.6,
 - 2. A boron concentration in accordance with Figure 3.1.2.6, and
 - 3. A minimum solution temperature of 63°F.
- b. The refueling water storage tank with:
 - 1. A contained borated water volume of between 370,000 and 375,000 gallons,
 - 2. Between 2500 and 2700 ppm of boron,
 - 3. A minimum solution temperature of 60°F, and
 - 4. A maximum solution temperature of 105°F.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

- a. With the boric acid storage system inoperable and being used as one of the above required borated water sources, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% delta k/k at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 30 hours.
- b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

REACTIVITY CONTROL SYSTEMS

VEILLANCE REQUIREMENTS

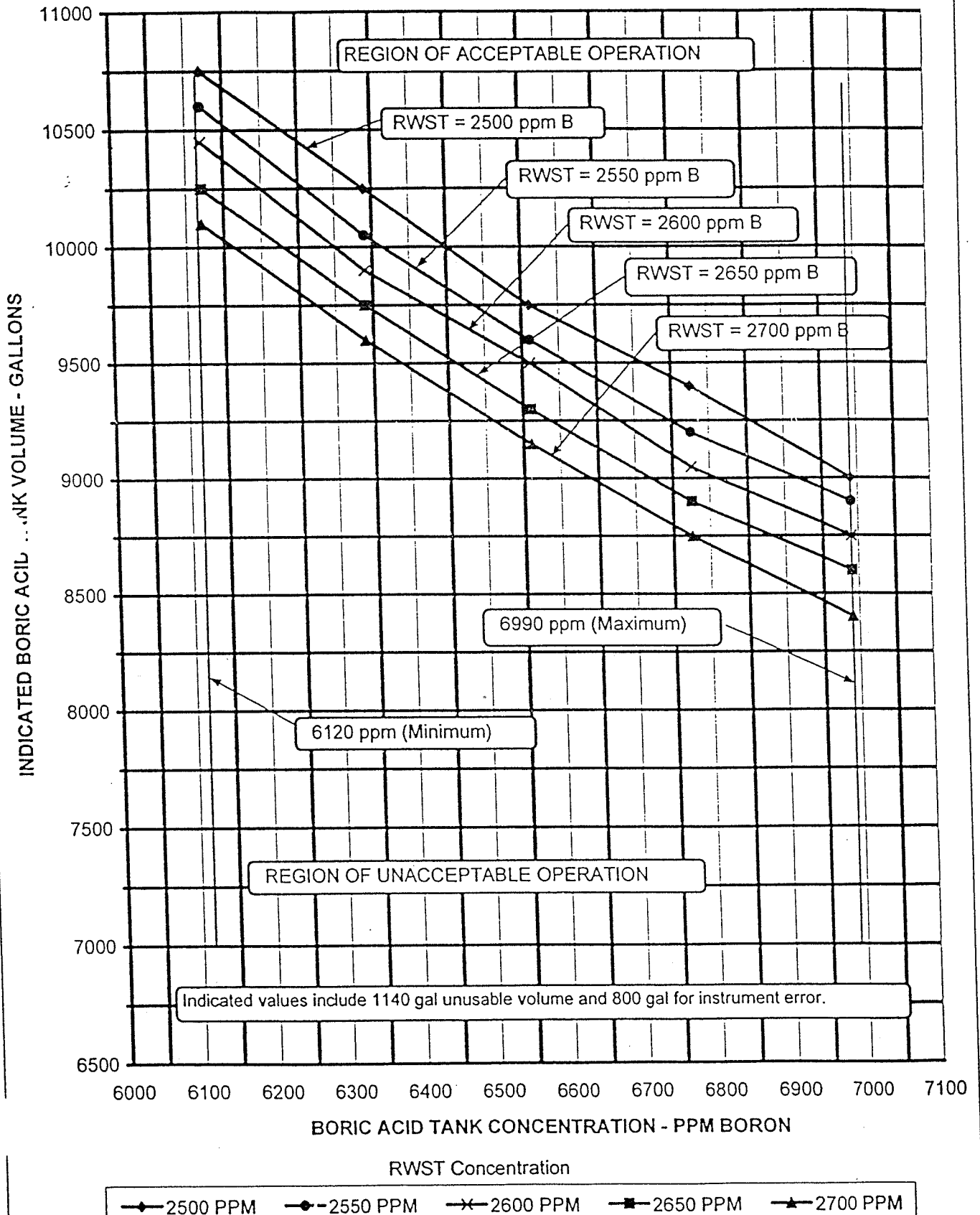
TR 4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

- a. For the boric acid storage system, when it is the source of borated water by:
 1. Verifying the boron concentration at least once per 7 days,
 2. Verifying the borated water volume at least once per 7 days, and
 3. Verifying the boric acid storage tank solution temperature is greater than or equal to 63°F at least once per 7 days by verifying the area temperature to be greater than or equal to 63°F, or
 4. Whenever the boric acid tank area temperature is less than 63°F and the boric acid storage system being used as the source of borated water, within 6 hours and every 24 hours thereafter, verify the boric acid tank solution temperature to be greater than or equal to 63°F until the boric acid tank area temperature has returned to greater than or equal to 63°F.

- b. For the refueling water storage tank by:
 1. Verifying the boron concentration at least once per 7 days,
 2. Verifying the borated water volume at least once per 7 days, and
 3. Verifying the solution temperature at least once per 24 hours.

This figure is for Boric Acid Tank A only.

TRM FIGURE 3.1.2.6 (Units 1 & 2)
BORIC ACID TANK LIMITS
BASED ON RWST BORON CONCENTRATION



This figure is for Boric Acid Tank B only.

BORIC ACID TANK REQUIRED VOLUME vs. BORIC ACID TANK CONCENTRATION

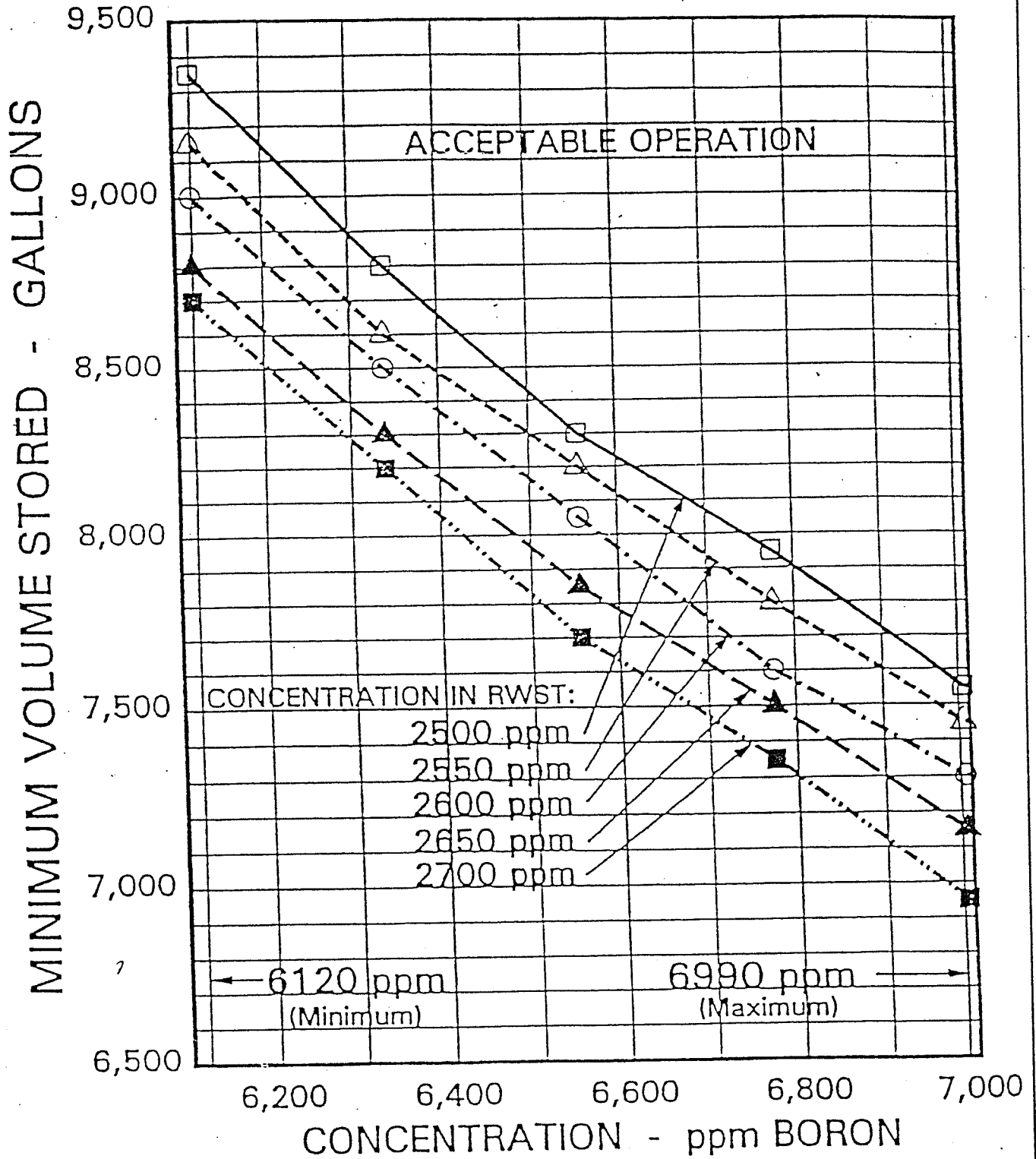
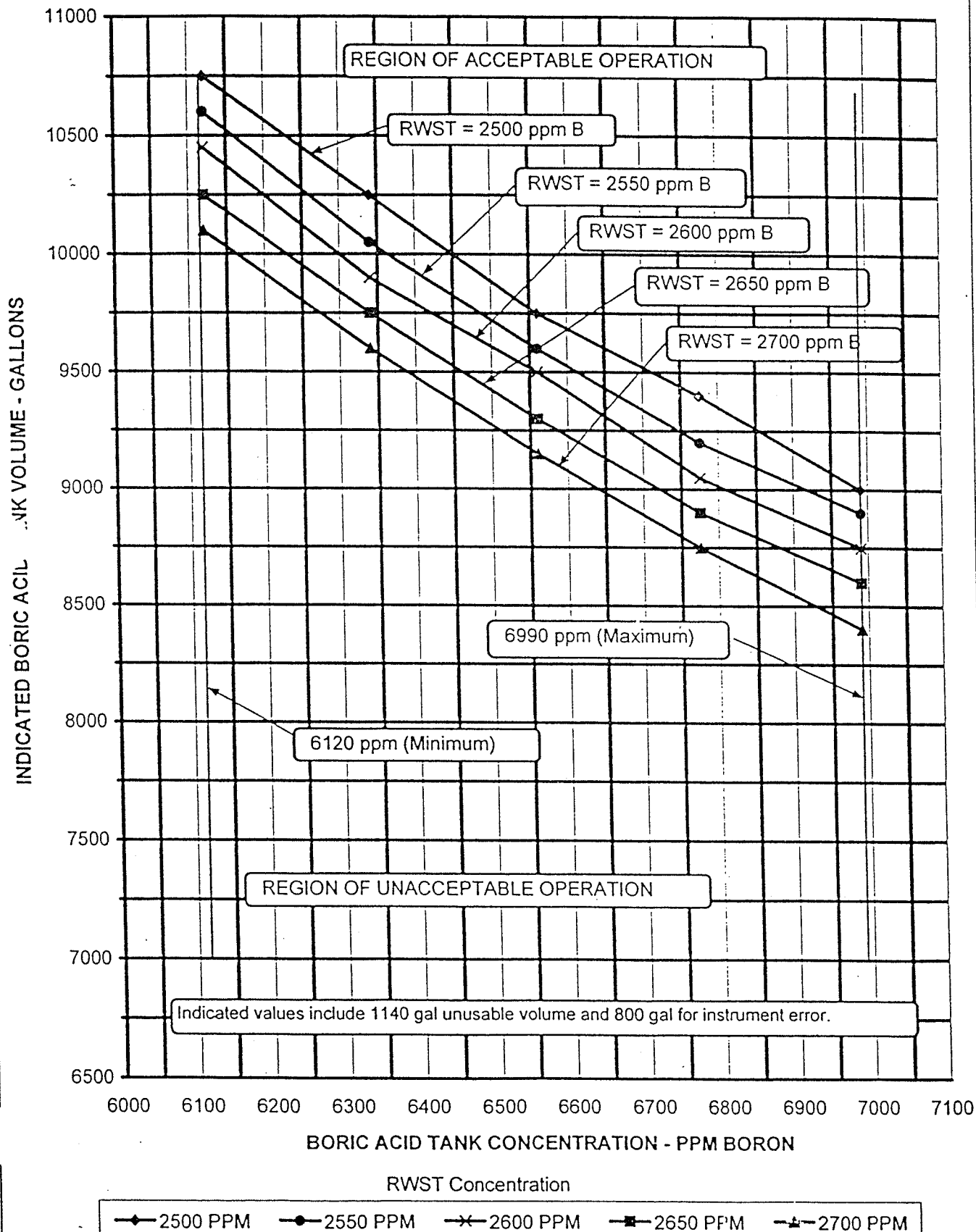


FIGURE 3.1.2.6

This figure is for Boric Acid Tank C only.

TRM FIGURE 3.1.2.6 (Units 1 & 2)
BORIC ACID TANK LIMITS
BASED ON RWST BORON CONCENTRATION



Attachment 1
Operations Information (Chemistry)
 (Page 1 of 2)

Sample Point	Units	Boron	Date / Time	Goal	Limit
U1 RCS	ppm	1130	11/3/09 1130	Variable	Variable
U2 RCS	ppm	2476	11/4/09 0856	Variable	Variable
U1 RWST	ppm	2597	11/01/09 0330	2550 – 2650	2500 – 2700
U2 RWST	ppm	2608	10/30/09 1114	2550 – 2650	2500 – 2700
BAT A	ppm	6869	11/02/09 0320	Variable	Variable
BAT B	ppm	6932	11/05/09 0202	Variable	Variable
BAT C	ppm	6763	11/02/09 0320	Variable	Variable
U1 CLA #1	ppm	2549	11/02/09 0330	2470-2630	2400-2700
U1 CLA #2	ppm	2558	11/02/09 0336	2470-2630	2400-2700
U1 CLA #3	ppm	2567	11/02/09 0320	2470-2630	2400-2700
U1 CLA #4	ppm	2555	11/02/09 0341	2470-2630	2400-2700
U2 CLA #1	ppm	2539	9/28/09 0947	2470-2630	2400-2700
U2 CLA #2	ppm	2455	10/14/09 1455	2470-2630	2400-2700
U2 CLA #3	ppm	2542	9/28/09 1115	2470-2630	2400-2700
U2 CLA #4	ppm	2534	9/28/09 1023	2470-2630	2400-2700
Spent Fuel Pool	ppm	2285	11/04/09 0215	≥ 2050	≥ 2000
Lithium Results			Date / Time	Goal	Midpoint
U1 RCS Lithium	ppm	4.02	11/3/09 1130	3.84-4.14	3.99
U2 RCS Lithium	ppm	0.09	10/26/09 1615		

Attachment 1
Operations Information (Chemistry)
(Page 2 of 2)

Primary to Secondary Leakrate Information (Total CPM RM-90-99/119)					
Indicator	Units	U1	Date / Time	U2	Date/Time
SI 50 S/G Leakage?	Yes/No	No	11/05/09 0005	No	11/02/09 0001
SI 137.5 CVE Leakrate	gpd	<0.1	11/03/09 0500	<5.0	10/6/09 0310
5 gpd leak equivalent	cpm	71	11/03/09 0500	135	10/03/09 0130
30 gpd leak equivalent	cpm	224	11/03/09 0500	612	10/03/09 0130
50 gpd leak equivalent	cpm	347	11/03/09 0500	993	10/03/09 0130
75 gpd leak equivalent	cpm	500	11/03/09 0500	1469	10/03/09 0130
CVE Air Inleakage	cfm	26	11/03/09 0500	8.4	10/03/09 0130
Bkgd on 99 /119	cpm	40	11/03/09 0500	40	10/03/09 0130
Correlation Factor 99/119	cpm/gpd	6.13	11/03/09 0500	19.06	10/03/09 0130
Steady State conditions are necessary for an accurate determination of leak rate using the CVE Rad Monitor					

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT

SURVEILLANCE INSTRUCTION

1-SI-OPS-000-003.W

WEEKLY SHIFT LOG

Revision 43

QUALITY RELATED

PREPARED/PROOFREAD BY: H J RICKS

RESPONSIBLE ORGANIZATION: OPERATIONS

APPROVED BY: W. T. LEARY

EFFECTIVE DATE: 04/08/09

LEVEL OF USE: CONTINUOUS

VFL Today
JD

REVISION

DESCRIPTION: Updated breaker UNID on 125VDC Vital Batt Bd III feeder to Vital Inverter 1-III per WO 07-777880-000 (EDCN E22208A)

SQN	WEEKLY SHIFT LOG	1-SI-OPS-000-003.W Rev. 43 Page 24 of 46
1		

Date
____/____/____

APPENDIX A
Page 17 of 22

DESCRIPTION	SR REFERENCE	MODE	NOTE	T.S. LIMITS	INSTRUMENT	UNITS	DATA	REMARKS
Spent Fuel Pit Water Level el 734	4.9.11	With spent fuel in pool	1	Min of 23 ft above fuel	Local depth gage	(√)		
Boric Acid Tank "A" Level 1-M-6	TR 4.1.2.5.a.2	4,5,6	11	≥ 6400 gal.	1-LI-62-238	Gal.		
	TR 4.1.2.6.a.2	1,2,3	12	TRM figure 3.1.2.6		Gal.		
Boric Acid Tank "C" Level 1-M-6	TR 4.1.2.5.a.2	4,5,6	11	≥ 6400 gal.	0-LI-62-242	Gal.		
	TR 4.1.2.6.a.2	1,2,3	12	TRM figure 3.1.2.6		Gal.		
RWST Level 1-M-6	TR 4.1.2.6.b.2 4.5.5.a.1	1,2,3,4	13,14	≥370,000 ≤375,000	1-LI-63-46 or 1-LI-63-49	Gal.		
	TR 4.1.2.5.b.2	4,5,6	11	10% or ≥ 55,000 gal	1-LI-63-50 or 1-LI-63-51 or 1-LI-63-52 or 1-LI-63-53	% or Gal.		
UO/RO REVIEW								

NOTES

1. Verify water level is above bottom mark on depth gage installed on west wall of spent fuel pit. If water level is in the bottom of the normal range, then contact MCR to make-up to high in the normal range per 0-SO-78-1
11. In modes 4, 5 and 6 one boric acid storage tank or RWST is required to be operable.
12. In modes 1, 2, and 3 one boric acid storage tank is to be operable if required by LCO 3.1.2.2.
13. In modes 1, 2, 3, and 4 RWST is required to be operable.
14. If deviation of ≥1000 gallons exists between channels, then submit work request (WR) to have transmitter repaired.

SEQUOYAH NUCLEAR PLANT JOB PERFORMANCE MEASURE

RO ADMIN A.3

**Determine Potential Total Dose
for Valve Alignment**

RO/SRO
JOB PERFORMANCE MEASURE

Task: Determine Potential Total Dose For Valve Alignment.

Task #: (RO) 3430290302

Task Standard: Determine total dose which will occur while operating 1-FCV-63-39, and based on that determination, state whether the projected dose posted on the RWP will be exceeded and whether the administrative dose limit will be exceeded.

Alternate Path: YES: _____ NO: X

Time Critical Task: YES: _____ NO: X

K/A Reference/Ratings: G2.3.4 (3.2/3.7)

Method of Testing:

Simulated Performance: _____ Actual Performance: X

Evaluation Method:

Simulator _____ In-Plant _____ Classroom X

Main Control Room _____ Mock-up _____

Performer: _____
Trainee Name

Evaluator: _____ / _____
Name / Signature DATE

Performance Rating: SAT: _____ UNSAT: _____

Validation Time: _____ Total Time: _____

Performance Time: Start Time: _____ Finish Time: _____

COMMENTS

SPECIAL INSTRUCTIONS TO EVALUATOR:

1. Critical steps are identified in step SAT/UNSAT column by bold print 'Critical Step'.
2. Any UNSAT requires comments
3. Task should be performed in a class room.

Tools/Equipment/Procedures Needed:

RCI-03
1-PI-OPS-000-003.0, pages 67 thru 71
Survey Map of Unit 1 Aux Bldg elevation 609', Pipe Chase
Calculator

References:

	Reference	Title	Rev No.
1.	RCI-03	Personnel Monitoring	48

=====

READ TO OPERATOR

DIRECTIONS TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM. 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. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

1. Unit 1 is at Mode 6 with fuel movement in progress.
2. A CVCS resin transfer is underway at this time.
3. It has just been discovered that the stroke test for 1-FCV-63-39, CCPIT Inlet Valve, is required to be performed to comply with AOP-N.08, Appendix R Fire, requirements.
4. An RWP that was developed for this test has a dose limit of 75 mr.
5. A survey map is available for elevation 690' of the auxiliary building, Pipe Chase, showing dose rates and projected travel times to reach 1-FCV-63-39.
6. Assume that it will take the maximum allowed time stroke 1-FCV-63-39 from Open to Close.
7. You are also expected to return the valve to its original position before exiting the area.

INITIATING CUES:

1. You have been directed to perform the alignment of 1-FCV-63-39, in accordance with 1-PI-OPS-000-003.0, "Periodic Stroking of Unit 1 Time Critical Valves," to satisfy the testing requirements of AOP-N.08.
2. Your total dose for the year to date is 715 mr.
3. Calculate your estimated total dose to perform this job using the attached survey map.
4. Determine if you can perform this job without exceeding the allotted dose per the planned RWP and/or your administrative dose limit. Show all work.

Job Performance Checklist:

STEP / STANDARD	SAT / UNSAT
<p>Evaluator Note: If the applicant asks if this is a planned special exposure, respond that the applicant should re-read the initiating cues.</p> <p>Candidate must calculate 2 way travel time thru all transient areas.</p>	
<p>STEP 1: Calculate exposure during transit thru the pipe chase to the valve and return.</p> <p><u>STANDARD:</u></p> <p><u>Transit to STEP OFF PAD</u> $(250 \text{ mr/hr})(.5 \text{ min})(2)(\text{hr}/60\text{min}) = \underline{4.166 \text{ mr}}$</p> <p><u>Time at Step Off Pad</u> $(450 \text{ mr/hr})(2.5 \text{ min})(\text{hr}/60\text{min}) = \underline{18.75 \text{ mr}}$</p> <p><u>Operate valve</u> $(5\text{min})(450\text{mr/hr})(2)(\text{hr}/60\text{min}) = \underline{75 \text{ mr}}$</p> <p>Critical step to calculate the expected dose accurately in order to avoid exceeding dose limits.</p> <p>COMMENTS:</p>	<p>_____ SAT</p> <p>_____ UNSAT</p> <p>Critical Step</p>

Job Performance Checklist:

STEP / STANDARD	SAT / UNSAT
<p>Evaluator Note: Some rounding of numbers is allowed as long as the examinee follows sound mathematical standards. The acceptable total is 97.91 to 98 mr.</p>	
<p><u>STEP 2.:</u> Calculate the total exposure received while performing the task.</p> <p><u>STANDARD:</u> Individual doses received are added up.</p> <p>4.166 mr + 75 mr + 18.75 mr = <u>97.91mr.</u></p> <p>Rounding values</p> <p>4 + 75 + 19 = <u>98 mr</u></p> <p>Acceptable Range - <u>97.91 - 98 mrem</u></p> <p>Step is critical to avoid exceeding dose limits.</p> <p><u>COMMENTS:</u></p>	<p>_____ SAT</p> <p>_____ UNSAT</p> <p>Critical Step</p>
<p><u>Step 3.</u> Applicant determines that estimated exposure is greater than that allowed by RWP.</p> <p><u>STANDARD:</u> Total dose calculated in previous step is compared to Total Dose allowed on RWP: 97.9 mr (estimated dose from job) 75 mr (total dose for the job allowed on RWP) and reports to US that projected dose exceeds that allowed by RWP.</p> <p>Critical step to identify that valve alignment could <u>not</u> be performed under the current RWP.</p> <p><u>COMMENTS:</u></p>	<p>_____ SAT</p> <p>_____ UNSAT</p> <p>Critical Step</p>

Job Performance Checklist:

STEP / STANDARD	SAT / UNSAT
<p><u>Step 4.</u> Applicant calculates his/her total estimated exposure.</p> <p>STANDARD: Total dose calculated in previous step is added to applicant's dose for the year: 97.9 mr (estimated dose from job) + 715 mr (total dose for the year) = 812.9 mr (813 mr).</p> <p>Critical step to calculate dose accurately in order to avoid exceeding dose limits.</p> <p><u>COMMENTS:</u></p>	<p>_____ SAT</p> <p>_____ UNSAT</p> <p>Critical Step</p>
<p><u>STEP 5.</u> Applicant determines admin dose limit will not be exceeded if the job is performed. Applicant notifies supervisor of findings, stating that admin dose limit will be exceeded.</p> <p>STANDARD: Admin dose limit for the year is 1000 mrem. Potential dose received is 98 mrem. Total is 812 to 813 mrem, which does not exceed admin limit.</p> <p><u>COMMENTS:</u></p> <p><u>Cue:</u> This completes the JPM.</p>	<p>_____ SAT</p> <p>_____ UNSAT</p>

Key DO NOT HAND to APPLICANT

SURVEY DATA:

- 1-FCV-63-39 is shown on the Survey map.
- Travel time from the pipe chase door thru the room to the Step Off Pad (SOP) is 30 seconds.
- Estimated time at valve is 10 minutes (5 minutes to Close/ 5 minutes to Open. (assumed max time from surveillance.)
- Estimated time at SOP to exit is 2.5 min. (given in turnover data)
- General Area Dose rates are listed on the survey map.

RESULTS:

Pipe Chase door to SOP and back	$(250\text{mr/hr})(0.5\text{ min})(2)(\text{hr}/60\text{min})$	<u>4.16 mr (4.2 or 4)</u>
Operate valve (Close –then– Open)	$(450\text{mr/hr})(5\text{min})(2)(\text{hr}/60\text{min})$	<u>75 mr</u>
Time at SOP	$(450\text{mr/hr})(2.5\text{min})(\text{hr}/60\text{min})$	<u>18.75 mr</u>

Accept 97.91 - to 98 mr as dose expected to perform task.

When compared to dose allowed by RWP of 75 mr, the RWP limit WILL BE EXCEEDED. Work cannot be performed under this RWP.

When added to current dose for the year of 715 mr, the administrative dose limit of 1000 mrem WILL NOT BE EXCEEDED (total 813 mrem).

READ TO OPERATOR

DIRECTIONS TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM. 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. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

1. Unit 1 is at normal full power alignment.
2. A CVCS resin transfer is underway at this time.
3. It has just been discovered that the stroke test for 1-FCV-63-39, CCPIT Inlet Valve, is required to be performed to comply with AOP-N.08, Appendix R Fire, requirements.
4. An RWP that was developed for this test has a dose limit of 75 mr.
5. A survey map is available for elevation 690' of the auxiliary building, Pipe Chase, showing dose rates and projected travel times to reach 1-FCV-63-39.
6. Assume that it will take the maximum allowed time stroke 1-FCV-63-39 from Open to Close.
7. You are also expected to return the valve to its original position before exiting the area.

INITIATING CUES:

1. You have been directed to perform the alignment of 1-FCV-63-39, in accordance with 1-PI-OPS-000-003.0, "Periodic Stroking of Unit 1 Time Critical Valves," to satisfy the testing requirements of AOP-N.08.
2. Your total dose for the year to date is 715 mr (TEDE).
3. Calculate your estimated total dose to perform this job using the attached survey map.
4. Determine if you can perform this job without exceeding the allotted dose per the planned RWP and/or your administrative dose limit. Show all work.

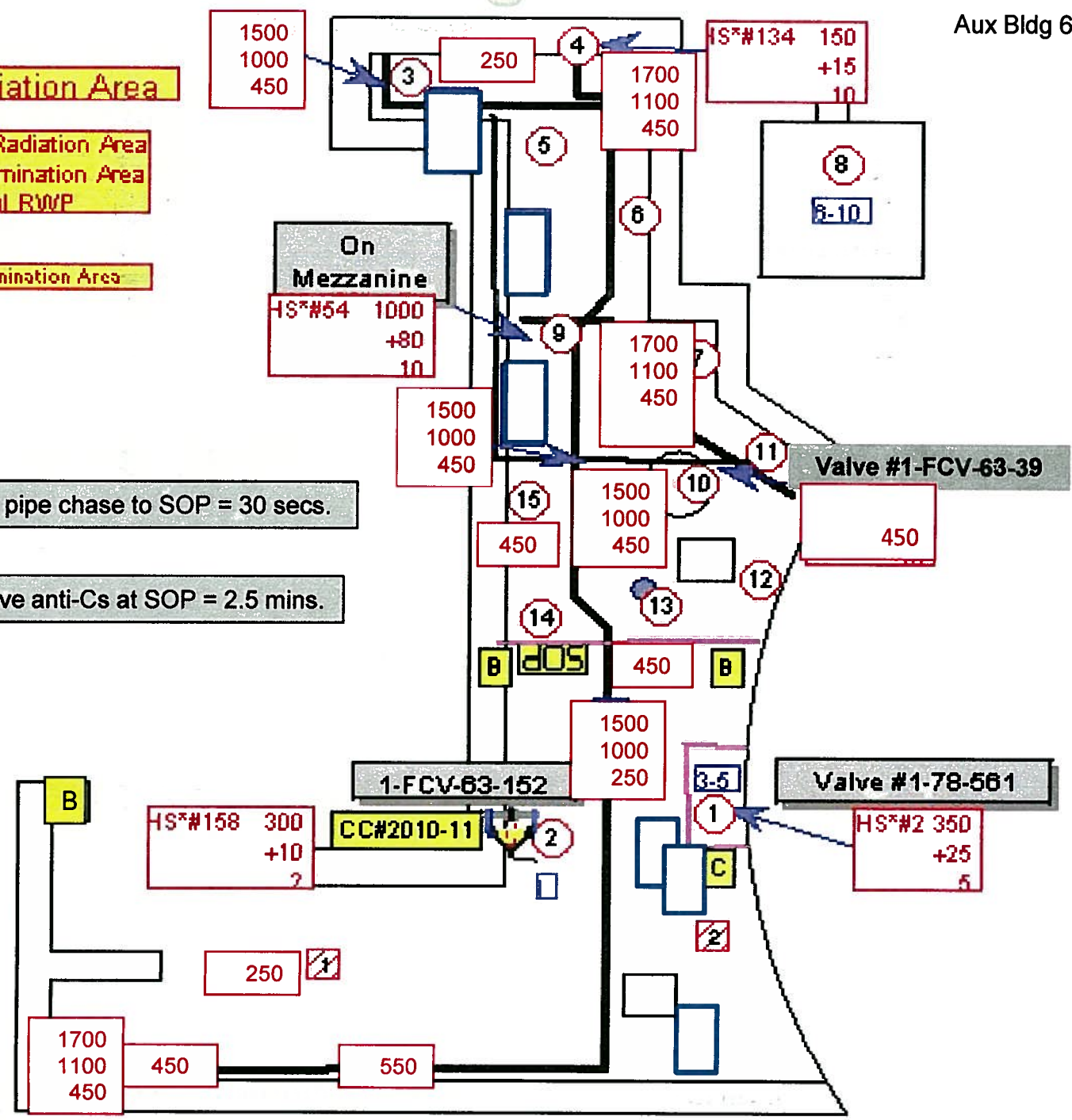
A= Radiation Area

**B= High Radiation Area
Contamination Area
Special RWP**

C= Contamination Area

Travel time thru pipe chase to SOP = 30 secs.

Time to remove anti-Cs at SOP = 2.5 mins.



SQN Unit 1	PERIODIC STROKING OF UNIT 1 TIME CRITICAL VALVES	1-PI-OPS-000-003.0 Rev. 0002 Page 67 of 101
-----------------------	-------------------------------------------------------------	------------------------------------------------------------

**Appendix O
(Page 1 of 5)**

Stroke Testing 1-FCV-63-39 Using Handwheel

NOTE

Testing of CCPIT inlet valve is NOT required during normal periodic performance of this PI. This appendix is for testing as needed to satisfy AOP-N.08 validation requirements.

Valve Nomenclature: CCPIT Inlet Valves

Location: Unit 1 690 Pipe Chase, behind CCPIT

Timing Position: From OPEN to CLOSED

Maximum Allowable Stroke Time: 5 minutes

Maximum Allowable Stroke Time Determined By: 13 minute action time
(ref. AOP-N.08) minus travel time and margin

Drawing References: 47W811-1

1.0 PRECAUTIONS

- A. This valve is located in a high radiation area and contamination zone. Entry must be coordinated with Radcon. Time in the area should be minimized.
- B. Motor-Operated valves which have been manually seated using handwheel are considered inoperable until valves are stroked electrically. Refer to GOI-6.
- C. Small scaffold is installed in 690 Pipe Chase to facilitate access over interfering piping and conduit. Scaffold inspection is required to be current.
- D. This section relies upon 1-SI-SXP-062-203.0 to establish CCPIT flow.
- E. Valve stroke time limit must be capable of being met by any qualified AUO who could be assigned this task. AUOs performing valve stroke tests should keep this in mind when assessing the force required to turn valves.

SQN Unit 1	PERIODIC STROKING OF UNIT 1 TIME CRITICAL VALVES	1-PI-OPS-000-003.0 Rev. 0002 Page 68 of 101
---------------	-----------------------------------------------------	---------------------------------------------------

Appendix O
(Page 2 of 5)

2.0 PRELIMINARY ACTIONS

NOTE

Throughout this Instruction, where an **IF/THEN** statement exists, the step should be **N/A** if the stated condition does **NOT** exist.

- [1] **ENSURE** all precautions reviewed. _____
- [2] **VERIFY** Unit 1 is in one of the following conditions:
 - Mode 6 with head removed _____
 - OR**
 - Defueled. _____

NOTE

If only one AUO will be entering 690 Pipe Chase, placing stopwatch on a lanyard may make timing easier.

- [3] **OBTAIN** a stopwatch **AND**
RECORD TVA ID. [_____]
- [4] **ENSURE** CCPIT flowpath established in accordance with 1-SI-SXP-062-203.0, CCP 1A-A and 1B-B Comprehensive Performance and Check Valve Test
- [5] **ENSURE** Unit Supervisor permission obtained to perform handwheel closure test on 1-FCV-63-39.

SRO Signature Date

SQN Unit 1	PERIODIC STROKING OF UNIT 1 TIME CRITICAL VALVES	1-PI-OPS-000-003.0 Rev. 0002 Page 69 of 101
---------------	-----------------------------------------------------	---------------------------------------------------

Appendix O
(Page 3 of 5)

3.0 INSTRUCTIONS

- [1] **ENSURE** Section 2.0 Preliminary Actions completed. _____
- [2] **ENSURE** **[1-FCV-63-39]** FULL OPEN. _____
- [3] **PLACE** breaker for 1-FCV-63-39 in OFF position:

BREAKER	LOCATION	OFF	
		_____	_____
1-BCTD-63-39-A	Rx MOV Board 1A1-A Compt 12E	1st	CV

CAUTION

Excessive seating force could cause actuator damage.

NOTES

- 1) Stopwatch should used to measure stroke time.
- 2) Valve is expected to require ~145 handwheel turns for closure.
- 3) Closing 1-FCV-63-39 will reduce ECCS injection flow. 1-HCV-74-34 may require adjustment to maintain refueling cavity level.

- [4] **CLOSE** **[1-FCV-63-39]** CCPIT Inlet Valve **AND**

RECORD stroke time. _____

ACCEPTANCE CRITERIA	
	ACCEPTABLE RANGE
	less than 5 min
Stroke Time _____	<input type="checkbox"/>

- [5] **IF** 1-HCV-74-34 is throttled open to maintain cavity level,
THEN
NOTIFY local operator to adjust **[1-HCV-74-34]** as necessary.

SQN Unit 1	PERIODIC STROKING OF UNIT 1 TIME CRITICAL VALVES	1-PI-OPS-000-003.0 Rev. 0002 Page 70 of 101
---------------	-----------------------------------------------------	---------------------------------------------------

Appendix O
(Page 4 of 5)

3.0 INSTRUCTIONS (continued)

NOTE

The following step unseats 1-FCV-63-39 with handwheel to prevent motor overload and possible damage during subsequent electrical operation.

[6] **OPERATE** 1-FCV-63-39 handwheel in the open direction approximately 20 turns.

[7] **PLACE** breaker for 1-FCV-63-39 in ON position:

BREAKER	LOCATION	ON	
1-BCTD-63-39-A	Rx MOV Board 1A1-A Compt 12E	1st	CV

[8] **FULLY OPEN** [1-FCV-63-39] USING [1-HS-63-39A]. _____

NOTE

Steps 3.0[9], 3.0[10], and 3.0[11] may be marked N/A if CCPIT valves and HCV-74-34 must remain open for other activities and valves will be restored by another procedure.

[9] **CLOSE** [1-FCV-63-39] USING [1-HS-63-39A]. _____
1st

IV

[10] **NOTIFY** local operator to close [1-HCV-74-34].

[11] **ENSURE** [1-HCV-74-34] CLOSED and LOCKED. _____
1st

CV

SQN Unit 1	PERIODIC STROKING OF UNIT 1 TIME CRITICAL VALVES	1-PI-OPS-000-003.0 Rev. 0002 Page 71 of 101
---------------	-----------------------------------------------------	---------------------------------------------------

Appendix O
(Page 5 of 5)

4.0 REMARKS

5.0 ACCEPTANCE CRITERIA

5.1 Evaluation of Test Performance

[1] Acceptable stroke time recorded?
Yes No

NOTE

The intent of step 5.1[2] is that no binding or erratic action was noted which might be an indicator of a problem although stroke time was acceptable.

[2] Valve performance was acceptable?
Yes No N/A

Initials Date

NOTE

If valve CANNOT be operated in the required time by any qualified operator, a PER is needed to evaluate the impact on regulatory compliance and reportability.

[3] **IF** step 5.1[1] or 5.1[2] is NO,
THEN
PERFORM the following:

[3.1] **NOTIFY** Unit 1 US. _____

[3.2] **INITIATE** WO and/or PER.
WO or PER # _____

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT

RADIOLOGICAL CONTROL INSTRUCTION

RCI-03

PERSONNEL MONITORING

Revision 48

QUALITY RELATED

PREPARED BY: Terry F. Johnston

RESPONSIBLE ORGANIZATION: Radiation Protection

APPROVED BY: JOHN VINCELLI

EFFECTIVE DATE: 02/24/05

VERIFICATION DATE: N/A

LEVEL OF USE: **INFORMATION ONLY**

REVISION
DESCRIPTION

This revision updates the references, includes organizational title changes, removes the previous requirement to list the SSN on TLDs (PER #76092) and updates general information. This revision is an intent revision.

Attachment 01, Prenatal Radiation Exposure Program, is revised to include organizational title changes. This revision is an intent revision.

Attachment 03, Area TLD Monitoring Program, is revised to include organizational title changes. This revision is an intent revision.

SQN	PERSONNEL MONITORING	RCI-03 Revision 48 Page 2 of 16
------------	-----------------------------	-----------------------------------------------------

1.0 PURPOSE

The purpose of this Instruction is to provide guidelines for monitoring personnel external radiation exposures.

2.0 SCOPE

This Instruction establishes the requirements for personnel dose monitoring (PCs, EDs, TLDs), extremity dose monitoring, Administrative Dose Levels (ADLs), emergency exposure guidance, details the prenatal radiation exposure program, provides the requirements for calculation of skin doses, and details the area TLD monitoring program.

3.0 REFERENCES

- A. 10CFR19, Notices, Instructions, and Reports to Workers; Inspection and Investigations
- B. 10CFR20, Standards for Protection Against Radiation
- C. NUREG CR-4418, Dose Calculation for Contamination of the Skin Using the Computer Code VARSKIN
- D. NUREG CR-5569, Health Physics Position Database
- E. NUREG CR-5873, VARSKIN MOD2 and SADDE MOD2: Computer Codes for Assessing Skin Dose from Skin Contamination
- F. NUREG CR-6204, Questions and Answers Based on Revised 10CFR20
- G. Regulatory Guide 1.109, Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purposes of Evaluating Compliance with 10CFR50, Appendix I
- H. Regulatory Guide 8.13, Instruction Concerning Prenatal Radiation Exposure
- I. NRC IE Notice 86-23, Excessive Skin Exposures Due to Contamination of Hot Particles
- J. NRC IE Notice 90-48, Enforcement Policy for Hot Particle Exposures
- K. SPP-2.3, Document Control
- L. SPP-5.1, Radiological Controls
- M. SPP-5.9, Radiological Control and Radioactive Material Shipment Augmented Quality Assurance Program
- N. ANSI N413-1974, Guidelines for the Documentation of Digital Computer Programs

SQN	PERSONNEL MONITORING	RCI-03 Revision 48 Page 3 of 16
-----	----------------------	---------------------------------------

3.0 REFERENCES (Continued)

- O. INPO 91-014, Guidelines for Radiological Protection at Nuclear Power Stations
- P. SQN Technical Specifications, Unit 1 and Unit 2
- Q. EPIP-15, Emergency Exposure Guidelines
- R. RCI-01, Radiation Protection Program
- S. RCI-04, Radiological Respiratory Protection Program
- T. RCI-11, Bioassay Program
- U. RCI-15, Radiological Postings
- V. RCI-23, Radiation Protection Records
- W. RCI-24, Control of Very High Radiation Areas
- X. RCI-27, External Dosimetry Services (TEDS) Laboratory Quality Manual
- Y. RCI-28, Control of Locked High Radiation Areas
- Z. RCI-29, Control of Radiation Protection Keys
- AA. ANI/MAELU Engineering Bulletin 88-3A, Nuclear Liability Guidance for Hot Particle Contamination
- AB. Revision of Technical Basis for Skin Dose Assessment Process, RIMS L78 880209 800
- AC. Counting Efficiencies for GM Detectors, RIMS L78 871105 800
- AD. Revision of Factors for Calculating Skin Dose from Contamination, RIMS L09 891215 800
- AE. Skin Dose Due to Beta Particles from Noble Gases, RIMS L78 870505 801
- AF. Validation Testing of the Lotus 1-2-3 Application SKINDOSE.WK1, RIMS L78 0606 801
- AG. Response of Panasonic Dosimeters to Submersion Exposure by Xe-133, Hoffman, J.M., and Catchen, G.L., Health Physics, Vol. 58, No. 1
- AH. A Method of Adding Nuclides to VARSKIN and QUINCE Skin Dose Calculation Software, Radiation Protection Management, Vol. 4, No. 6
- AI. RCTP-105, Personnel Inprocessing and Dosimetry Administrative Processes

SQN	PERSONNEL MONITORING	RCI-03 Revision 48 Page 4 of 16
-----	----------------------	---------------------------------------

4.0 DEFINITIONS/ABBREVIATIONS

Calendar Quarter - A calendar quarter is a normal 13 consecutive week period. The first calendar quarter of each year shall begin in the first week of January, and subsequent calendar quarters shall be such that no day is included in more than one calendar quarter, or omitted from inclusion within a calendar quarter.

Committed Effective Dose Equivalent (CEDE) - The sum of the products of the weighting factors applicable to each of the body organs or tissues that are irradiated and the committed dose equivalent to these organs or tissues.

Dose Equivalent - The product of the absorbed dose in tissue, quality factor, and all other necessary modifying factors at the location of interest. The units of dose equivalence are the rem and the Sievert (Sv).

Dose Estimate - (as used in this Instruction) A written estimate of dose received during the current calendar quarter and/or year.

Dosimetry - Personnel monitoring devices used to determine the occupational radiation dose received by an individual.

Electronic Dosimeter (ED) - An electronic dose measuring device worn next to the TLD that is used to give the wearer immediate dose readout. The ED dose is the actual deep dose received and may be used as official dose in the event the TLD is lost.

Form 4 - TVA Form 40763, Form-4 Equivalent, Cumulative Occupational Exposure History. This form is provided as **RCTP-105-7**.

Permanent TLD - A TLD assigned to personnel whose assignment on site is greater than 90 consecutive days. Each TLD (front insert) will contain the employee's name and TLD ID number.

Pocket Chamber (PC) - An ion chamber radiation detection device that is used to give the wearer an immediate dose readout. The pocket chamber dose may be used as official dose in the event that the TLD is lost. A pocket chamber is sometimes called a dosimeter or a DRD (direct reading dosimeter).

Radiation Operations (Rad Ops) - New organizational title for the previous Field Operations section.

Radiation Protection (Rad Protection) - New organizational title for the previous RADCON and Radwaste organizations.

Responsible Onsite Supervisor - A supervisor who is responsible for a specific individual and is on the list of site supervisors authorized to admit personnel to the site.

SQN	PERSONNEL MONITORING	RCI-03 Revision 48 Page 5 of 16
-----	----------------------	---------------------------------------

4.0 DEFINITIONS/ABBREVIATIONS (Continued)

Temporary TLD - A TLD assigned to personnel whose assignment on site is less than or equal to 90 consecutive days. Each TLD will contain an ID number. Upon issuance, the employee's name is recorded on the TLD.

Total Effective Dose Equivalent (TEDE) - The sum of the deep dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures).

Whole Body (or Total Body) - For the purposes of external exposure, the head, trunk (including male gonads), arms above the elbow, or legs above the knee.

5.0 RESPONSIBILITIES

5.1 Radiation Operations Manager

The Rad Ops Manager is responsible for ensuring that all requirements for personnel monitoring are maintained as specified in this Instruction.

5.2 Site Emergency Director (SED)

The SED is responsible for providing written authorization for all emergency radiation doses that may exceed the limits of 10CFR20 (Reference 3.B).

5.3 Site Section Supervisors

The site Section Supervisors are responsible for ensuring that personnel under their supervision comply with all TVAN procedures and instructions concerning radiation dose control.

5.4 Radiation Protection

Rad Protection is responsible for controlling, tracking, monitoring, reviewing, and reporting personnel radiation dose, to include:

- A. Maintenance of employee's radiation dose records. A Rad Protection computer-based records storage system shall be implemented to track and control worker radiation exposure.
- B. Issuance of periodic employee exposure summary reports.

SQN	PERSONNEL MONITORING	RCI-03 Revision 48 Page 6 of 16
-----	----------------------	---------------------------------------

5.0 RESPONSIBILITIES (Continued)

Note Written estimates of current year exposure are provided to an employee upon request. Estimates of dose in the absence of finally determined personnel monitoring results must be clearly indicated as such.

- C. Issuance of a written report regarding the radiation dose received at SQN during the current year to each individual having monitoring records at TVA. This report is provided annually to current employees and upon request to former employees.
- D. Investigation of TLD, PC and/or ED reading discrepancies.
- E. Ensuring that dosimetry values are correct, properly assigned, and entered into the appropriate Rad Protection records data base.
- F. Maintaining dosimetry processing accreditation under the National Voluntary Laboratory Accreditation Program (NVLAP).

5.5 Individual Employee

The individual employee is responsible for complying with all regulations concerning radiation exposure control. Employees are responsible for:

- A. Properly wearing prescribed dosimetry.
- B. Wearing only that dosimetry assigned to them.
- C. Proper care and handling of dosimetry, equipment, and instrumentation.
- D. Pick up and return of dosimetry.
- E. Notifying Rad Ops in the event of lost or damaged dosimetry.
- F. Processing through Rad Protection when arriving, transferring, or terminating at SQN.
- G. Informing Rad Protection whenever radiation exposure or medical radionuclide injections have been, or will be received, from a source other than TVA.

6.0 REQUIREMENTS

6.1 Precautions and Limitations

- A. During normal operations, no individual or group of individuals shall be permitted to receive a radiation dose that exceeds the limits specified in this Instruction.
- B. During emergency situations dose limitations will be as described in **EPIP-15** (Reference 3.Q).

SQN	PERSONNEL MONITORING	RCI-03 Revision 48 Page 7 of 16
-----	----------------------	---------------------------------------

6.0 REQUIREMENTS (Continued)

- C. TLD results are used as the official record of radiation exposure. In the event of a lost or damaged TLD, individuals are responsible for immediately reporting the condition to Rad Ops. In some instances, due to the loss or damage of a monitoring device or the inability of the monitoring device to measure certain types of radiation, it will be necessary to calculate an individual's dose. All calculations shall be documented and included in the employee's personal exposure history record.
- D. If a PC and/or ED is lost, damaged, or offscale, it shall be immediately reported to Rad Ops.
- E. When a PC reads in excess of 3/4 scale, or an ED alarms, the wearer shall report to Rad Ops to have the reading recorded and the PC and/or the ED reset.
- F. Individuals shall verify possession of their assigned dosimetry prior to entering a Radiologically Controlled Area (RCA).
- G. Any individual who enters an RCA shall sign in on an appropriate active **Radiation Work Permit** (RWP) and be monitored for radiation exposure with a TLD and a secondary dosimeter (e.g., ED), unless waived by Rad Protection management.
- H. Dosimetry shall normally be worn on the front of the person between the neck and belt line. It shall be in a clearly visible position. When worn in combination, a PC and/or ED should be located within six inches of the TLD. The beta window side of the TLD should normally face outward. When in a Radiation Area, High Radiation Area, or higher radiological zone classification, the PC and/or ED should be placed in a location that will allow the user to frequently read them.
- I. Extremity doses shall be measured when an individual's extremity dose exceeds, or is expected to exceed, 10% of the annual limit as indicated in **10CFR20**. Extremity dosimetry will be issued when the whole body TLD is not an appropriate monitor of extremity dose.
- J. If it is determined by Rad Ops that the portion of the body most likely to receive the greatest exposure is not in the area of the normal placement location of the TLD, the TLD will be moved to the more appropriate area, and/or multiple TLDs provided.

SQN	PERSONNEL MONITORING	RCI-03 Revision 48 Page 8 of 16
-----	----------------------	---------------------------------------

6.0 REQUIREMENTS (Continued)

- K. Any employee whose radiation dose exceeds any established limits shall not be permitted to enter any RCA for the remainder of the specific monitoring period.
- L. During a site emergency, storage, pickup and collection of dosimetry may be performed at alternate locations as conditions warrant.

6.2 General

- A. Dosimetry processing equipment, TLDs, PCs, and EDs shall be calibrated in accordance with approved procedures.
- B. As a minimum, all assigned TLDs are read at least **semi-annually**. Special TLD readouts are performed as necessary.
- C. TLDs are not required to be stored in their specified storage location upon exiting the plant site each day. However, each individual is responsible for maintaining possession of their TLD and ensuring that it is worn in accordance with the requirements of this Instruction.
- D. Any individual permitted to enter a posted High Radiation Area, Locked High Radiation Area, or Very High Radiation Area shall comply with the requirements of **RCI-15** (Reference 3.U), **RCI-24** (Reference 3.W), **RCI-28** (Reference 3.Y), and/or **RCI-29** (Reference 3.Z), as appropriate. [C.1][C.3]
- E. Dosimetry may be issued by Rad Protection after a **Form-4** (RCTP-105-7, TVA Form 40763), or equivalent, has been initiated and signed by the individual and all applicable requirements have been met (i.e., bioassay, training, etc.).
- F. For individuals requiring an Administrative Dose Level (**ADL**) of less than 500 mrem per year, current year and lifetime dose estimates must be provided and signed by the individual. A **Request for Dosimetry Issuance** (RCTP-105-10, TVA Form 40823), or equivalent, will be used to document current year and lifetime estimates.

SQN	PERSONNEL MONITORING	RCI-03 Revision 48 Page 9 of 16
-----	----------------------	---------------------------------------

6.0 REQUIREMENTS (Continued)

- G. All individuals who have a permanent/temporary TLD at SQN must checkout through Rad Protection prior to terminating work at SQN. In addition, individuals who will visit another licensee or TVA plant, and require a TLD, must checkout prior to leaving SQN.
- H. Area TLDs will be controlled in accordance with the requirements of **Attachment 03, Area TLD Monitoring Program**.

6.3 Assignment of Radiation Dose Limits

- A. If an employee is assigned to work at a non-TVA installation where an exposure to radiation is incurred, the employee shall inform Rad Protection of this assignment. The employee shall turn in their dosimetry, obtain any required bioassays, and complete any requested documentation. When the employee returns, they must report to Rad Protection to update their exposure records.
- B. When visitor or contract personnel have more restrictive dose limits than TVA, the more restrictive limits will be used. It is the responsibility of the contractor to provide written notification to Rad Protection of any company administrative limit.

6.4 Administrative Dose Levels (ADLs)

In addition to the limits of **10CFR20**, ADLs shall be used. The following ADLs shall be observed for routine work:

- A. To ensure that ADLs are not exceeded an administrative control system has been established (refer to **Section 6.7**).
- B. ADLs are based on dosimeters used in determining the reported dose. Results which exceed an ADL, based on other dosimeter data, do not violate the ADL.

6.0 REQUIREMENTS (Continued)

C. An individual's dose shall be controlled by the ADLs listed in the following table:

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 the individual	Not applicable
Up to 1.0 TEDE (or 3.0 LDE or 10 SDE) all sources	Form 4 (or equivalent) to document current year and previous years dose equivalent	Not Applicable
To exceed 1.0 TEDE (or 3.0 LDE or 10 SDE) all sources	Same as above	Site Radiation Protection Manager / RSO
To exceed 5.0 ³ TEDE all sources	Form 4 information must be verified and a Planned Special Exposure initiated	Site Rad Protection Mgr / RSO, Plant Manager ¹ , and Site VP ² or SED, as appropriate
To exceed 1N ⁴ all sources	Form 4 must be verified	Site Rad Protection Mgr / RSO, Plant Manager ¹ , and Site VP ² or SED, as appropriate

Legend

- 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 TEDE should not exceed 1N rem, where N equals the individual's age in years at their last birthday, without authorization signatures delineated.

SQN	PERSONNEL MONITORING	RCI-03 Revision 48 Page 11 of 16
-----	----------------------	----------------------------------------

6.0 REQUIREMENTS (Continued)

- D. Individuals under the age of **18** shall not be granted RCA access.
- E. Individuals whose lifetime accumulated TEDE is $\geq 1\text{N rem}$ shall be limited to **1,000 mrem/yr.** The administrative controls of previous **Table 1** are applicable.
- F. To authorize ADL increases, an **Administrative Dose Level Extension** (RCTP-105-1, TVA Form 40757) must be completed in accordance with **SPP-5.1** (Reference 3.L). At the discretion of the **Rad Protection Manager**, other methods may be utilized (i.e., a memo covering a group of people). Alternate methods shall include the information required on the **RCTP-105-1**. ADL extensions are tracked on an **ADL Tracking Sheet** (RCTP-105-6, TVA Form 40762).
- G. Any personnel exposure received which is in excess of the limits of **10CFR20** shall be reported by the site **Rad Protection Manager** to the Radiation Effects Advisory Group (REAG) and the appropriate area chief physician for an examination. A medical examination and authorization from the Chief Nuclear Officer and Executive Vice President are required before resumption of duties in RCAs for individuals who have received five times the annual limit of **10CFR20**.
- H. Prenatal exposure will be controlled as described in **Attachment 01, Prenatal Radiation Exposure Program.**
- I. Employees shall be instructed during RADCON training to report to their local TVA medical facility and site Rad Protection whenever they receive medical external radiation therapy or internal radionuclides for diagnosis or treatment (routine diagnostic x-rays need not be reported). Rad Ops shall be contacted and requested to perform a radiation survey on the worker. Based on the results of this survey, the individual may be restricted from entry into the RCA. RCA access will be granted when it can be determined, through bioassay and direct surveys, that the medical treatment does not interfere with the ability to monitor the individual's occupational dose.

SQN	PERSONNEL MONITORING	RCI-03 Revision 48 Page 12 of 16
-----	----------------------	----------------------------------------

6.0 REQUIREMENTS (Continued)

- J. Individuals who have or are undergoing therapeutic radiation exposures can have their ADL lowered to **500 mrem**, absent other circumstances which warrant a higher or lower ADL, upon their written request. A **Therapeutic Medical Radiation Exposure** (RCTP-105-8, TVA Form 40764) can be used for this request. The ADLs for individuals receiving therapeutic medical radiation exposures and individuals with radiologically related medical restrictions should be evaluated on a case-by-case basis. It is recommended that the opinion and recommendations of the individual's treating specialist be solicited. The treating specialist would be most aware of the individual diagnosis, specific therapy, the attendant risks, as well as any unusual susceptibility or precautions necessary regarding workplace radiation exposure. The individual and their supervisor will be counseled by Medical Services. A written record of this counseling shall be made and maintained along with all other supporting documentation. It will be included in the individual's personal history file. For individuals receiving therapeutic medical radiation exposures the individual should have risks clearly explained and be encouraged, but not required, to be placed on a lower ADL.
1. If the individual chooses to be placed on a lower ADL, the individual shall be informed that reasonable accommodations will be made to retain their present job status; however, their present job status cannot be guaranteed.
 2. For individuals with radiologically related medical restrictions, Medical Services, in consultation with the Rad Protection Manager (or designee), will determine if occupational exposure should be administratively restricted.

6.5 Skin Dose From Contamination

Skin dose calculations shall be performed in accordance with the requirements of **RCTP-106, Special Dosimetry Operations**.

6.0 REQUIREMENTS (Continued)

6.6 Emergency Exposure Guidance

- A. It is consistent with the risk concept to accept exposures leading to doses in excess of those appropriate for routine operation when recovery from an accident or major operational difficulty is necessary. Saving of a life, measures to circumvent substantial exposure to the general public, or the preservation of valuable installations may be sufficient cause for accepting above normal exposures. Dose limits for an emergency cannot be specified, but they should be commensurate with the significance of the objective and held to the lowest practical level that the emergency permits.
- B. Any decision to embark on emergency operations which would result in exposures in excess of **10CFR20** should be done in consultation with the most senior member of Rad Protection who is available on a timely basis. The guidelines that should be utilized when assigning administrative exposure limits for emergency conditions are listed below. Actual guidance for emergency situations is described in **EPIP-15**.

Table 2 Maximum Limiting Whole Body Dose Equivalent to Radiation Workers During Extreme Emergency	
Dose Equivalent	Remarks
10 rem	Taken only to prevent serious damage to the plant or hazard to personnel
25 rem	Taken to save a life

- C. Personnel must be made aware of possible consequences of such an exposure and selected on a voluntary basis. Emergency team members who are expected to respond to a radiological emergency must be aware of the consequences of such exposure.

SQN	PERSONNEL MONITORING	RCI-03 Revision 48 Page 14 of 16
-----	----------------------	----------------------------------------

6.0 REQUIREMENTS (Continued)

6.7 Administrative Control of Radiation Exposure

Note When deemed necessary, Rad Protection Support shall perform a special TLD analysis.

To minimize the potential for an overexposure, Rad Protection Support shall notify the responsible section supervisor in writing when an individual in that supervisor's section is approaching Action Level 1, or has exceeded Action Level 2:

A. Action Level 1

An individual has exceeded **80%** of the ADL. The responsible supervisor shall not use that individual in a posted Radiation Area, High Radiation Area, or higher radiological zone classification, unless no other qualified personnel with lower exposures are available.

B. Action Level 2

An individual has exceeded **90%** of the ADL. The individual shall be restricted from the RCA.

C. Removal of either Action Level restriction requires the completion of an **Administrative Dose Level Extension** and approval of the **Rad Protection Manager**.

SQN	PERSONNEL MONITORING	RCI-03 Revision 48 Page 15 of 16
-----	----------------------	----------------------------------------

7.0 QUALITY ASSURANCE (QA) RECORDS

7.1 QA Records

The following records are QA records and shall be completed, handled, and stored in accordance with **RCI-23** (Reference 3.V):

Request for Dosimetry Issuance

RCTP-105-10 TVA Form 40823

Form-4 Equivalent, Cumulative Occupational Exposure History

RCTP-105-7 TVA Form 40763

Administrative Dose Level Extension

RCTP-105-1 TVA Form 40757

Therapeutic Medical Radiation Exposure

RCTP-105-8 TVA Form 40764

7.2 Non-QA Records

The following records are non-QA records and shall be completed, handled, and stored in accordance with **RCI-23**:

ADL Tracking Sheet

RCTP-105-6 TVA Form 40762

8.0 APPENDICES/ATTACHMENTS

Attachment 01 Prenatal Radiation Exposure Program

Attachment 02 Calculation of Skin Dose

Attachment 03 Area TLD Monitoring Program

SOURCE NOTES

SQN	PERSONNEL MONITORING	RCI-03 Revision 48 Page 16 of 16
-----	----------------------	----------------------------------------

Source Notes

Implementing Statement	Requirements Document	Requirements Statement
C.1	RIMS A02 871116 013 RIMS S53 880208 994	Revise RCI-3 to indicate that each individual entering a high radiation area shall be equipped with a survey meter or alarming dosimeter unless the work is continuously monitored by a RADCON representative with an appropriate survey meter.
C.2	IE Notice #88-063 NER 910813001	Annotated as a reference to indicate high radiation area controls are addressed in the implementation of this Instruction. Cancelled by Revision 44.
C.3	Self-Assessment #SQ-RP-00-002	Transfer information denoted in previous Source Note C.1 to RCI-15 per this Self-Assessment.

Facility: <u>Sequoyah Nuclear Station 1 & 2</u>		Date of Examination: <u>09/13/2010</u>
Examination Level: RO <input type="checkbox"/> SRO <input checked="" type="checkbox"/>		Operating Test Number: <u>2010302</u>
Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	M, R, P	2.1.5 Ability to use procedures related to shift staffing, such as minimum crew compliment, overtime limitations, etc. (2.9*/3.9) A.1.a Evaluate Overtime Restrictions (Both RO/SRO)
Conduct of Operations	N, R	2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc. (3.9/4.2) A.1.b Review of Estimated Critical Position Calculation
Equipment Control	M, R	2.2.43 Knowledge of the process used to track inoperable alarms. (3.0/3.3) A.2 Review and Approve a Disabled Alarm Checklist
Radiation Control	D, R	2.3.4 Knowledge of radiation exposure limits under normal and emergency conditions. (3.2/3.7) A.3 Evaluate Worker Exposure
Emergency Procedures/Plan	D, R	2.4.38 Ability to take actions called for in the facility emergency plan, including supporting or acting as emergency coordinator if required. (2.4/4.4) A.4 Classify the Event per the REP (SGTR with Failed S/G Safety)
NOTE: All items (5 total) are required for SROs. RO applicants require only 4 items unless they are retaking only the administrative topics, when all 5 are required.		
* Type Codes & Criteria: (C)ontrol room, (S)imulator, or Class(R)oom (D)irect from bank (≤ 3 for ROs; ≤ 4 for SROs & RO retakes) (N)ew or (M)odified from bank (≥ 1) (P)revious 2 exams (≤ 1 ; randomly selected)		

A.1.a Evaluate Overtime Restrictions – This JPM has the candidate review a list of times for scheduled work and determine if any of the scheduled times exceed the procedure requirements for allowable work. This JPM is a Modified Bank JPM that was used on Feb 2009 exam.

A.1.b Review of Estimated Critical Position Calculation – This JPM has the candidate review an ECP calculation to identify the potential errors in data collection and/or calculations. This is a New JPM.

A.2. Review and Approve a Disabled Alarm Checklist – This JPM has the candidate review a work package to remove an annunciator from service. This is a Modified Bank JPM from the Fall of 2009 Watts Bar NRC exam.

A.3 Evaluate Worker Exposure – This JPM has the candidate review several workers exposure and determine whether the job that they have been assigned to perform will cause any of them to exceed the Administrative guidelines and if so what authorization would be required to allow the workers to continue. This is a Bank JPM.

A.4 Classify the Event per the REP (SGTR with Failed S/G Safety) – This JPM has the candidate determine the REP classification for the data presented and also make the correct recommended offsite personnel protective action recommendation. This is a Bank JPM.

SEQUOYAH NUCLEAR PLANT
September 2010 NRC Exam

SRO A.1.B

Review ECP Calculation

RO/SRO
JOB PERFORMANCE MEASURE

Task: Review ECP Calculation
Task # 0010020101 (SRO)

K/A Ratings: 2.1.25 (3.9/4.2)

Task Standard: Prior to taking the reactor critical, verify estimated critical position calculation in accordance with 1-SI-NUC-000-001.0, Estimated Critical Conditions.

Time Critical Task: YES: _____ NO: X

Method of Testing:

Simulated Performance: _____ Actual Performance: X

Evaluation Method:

Simulator _____ In-Plant _____ Classroom X

Main Control Room _____ Mock-up _____

Performer: _____
Trainee Name

Evaluator: _____ / _____
Name / Signature DATE

Performance Rating: SAT: _____ UNSAT: _____

Validation Time: 25 minutes Total Time: _____

Performance Time: Start Time: _____ Finish Time: _____

COMMENTS

SPECIAL INSTRUCTIONS TO EVALUATOR:

1. Critical Steps are identified in step SAT/UNSAT column by bold print "Critical Step."
2. Any UNSAT requires comments
3. Ensure operator performs the following required actions for **SELF-CHECKING**;
 - a. Identifies the correct unit, train, component, etc.
 - b. Reviews the intended action and expected response.
 - c. Compares the actual response to the expected response.

Tools/Equipment/Procedures Needed: NDR
0-SI-NUC-000-001.0
TI-33

REFERENCES:

	Reference	Title	Rev No.
A	0-SI-NUC-000-001.0	Estimated Critical Conditions	5
B	NDR	Nuclear Design Report for Sequoyah Unit 1 Cycle 17	Cycle 17
C	TI-33	Xenon Worth Calculation	25

READ TO OPERATOR

Directions to Trainee:

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. All steps shall be performed for this task. 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. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

1. Unit 1 is in Mode 3 and preparations are in progress to start up and take the reactor critical.
2. Core Average Burnup is 4000 MWD/MTU.
3. The present RCS Boron concentration is 1600 ppm.
4. Unit 1 tripped from full power at 0100 hours on 09/20/2010.
5. Unit 1 had been at 100% power for 2 weeks prior to the trip.
6. The ECP procedure will be applicable from 0100 until 0300 hours on 09/27/2010.
7. ECC bias is not applicable.

INITIATING CUES:

You are the Unit SRO and the Reactor Engineer has requested you to review 0-SI-NUC-000-001.0, Estimated Critical Conditions, Appendix B, Data Sheet 2, ECC Calculation Using NDR Data.

1. Identify any and all errors on the provided Estimated Critical Conditions, Appendix B data sheet.

Job Performance Checklist:

STEP/STANDARD	SAT/UNSAT
<p><u>STEP 1:</u> Determine boron concentration, core burnup, shutdown time and power fraction entries are correct.</p> <p><u>STANDARD:</u> SRO uses the turnover page cue sheet and verifies data correct.</p> <p><u>COMMENTS:</u></p>	<p>___ SAT</p> <p>___ UNSAT</p> <p>Start time ____</p>
<p><u>STEP 2:</u> Determine change in samarium/plutonium worth from equilibrium to peak.</p> <p><u>STANDARD:</u> SRO references the NDR Table 6-39 for 4000 MWD/MTU to identify that the samarium change is incorrect and identifies the correct number to be a -385 pcm samarium change.</p> <p>This step is critical to ensure ECP calculation integrity.</p> <p><u>COMMENTS:</u></p>	<p>___ SAT</p> <p>___ UNSAT</p> <p>Critical Step</p>
<p><u>STEP 3:</u> Calculate the change in Sm/Pu combined worth.</p> <p><u>STANDARD:</u> SRO notes the calculation is incorrect due to the values used in previous step and determines the corrected change in Sm/Pu worth to be -164 pcm. (+ or - 2 pcm)</p> <p>This step is critical to ensure ECP calculation integrity.</p> <p><u>COMMENTS:</u></p>	<p>___ SAT</p> <p>___ UNSAT</p> <p>Critical Step</p>
<p>Note: The NDR table 6-40 only shows time after trip values to 100 hours. At 100 hours the value is -13 pcm.</p>	

Job Performance Checklist:

STEP/STANDARD	SAT/UNSAT
<p><u>STEP 4:</u> Calculate Xenon worth using TI-33</p> <p><u>STANDARD:</u> SRO references TI-33 or the NDR.</p> <p><u>COMMENTS:</u></p>	<p>___ SAT</p> <p>___ UNSAT</p>
<p><u>STEP 5:</u> From the NDR, determine the design critical boron concentration for the cycle burnup value 4000 MWD/MTU.</p> <p><u>STANDARD:</u> SRO references the NDR table 6-1 and verifies the concentration given in the reactor engineer's data is correct.</p> <p><u>COMMENTS:</u></p>	<p>___ SAT</p> <p>___ UNSAT</p>
<p><u>STEP 6:</u> Determine from the NDR the ARO, HZP Differential boron worth for the current cycle burnup and boron concentration recorded.</p> <p><u>STANDARD:</u> SRO uses the NDR table 6-8 to determine the Reactor Engineer read the table incorrectly. The correct worth is -6.765 pcm .</p> <p>This step is critical to ensure ECP calculation integrity.</p> <p><u>COMMENTS:</u></p>	<p>___ SAT</p> <p>___ UNSAT</p> <p>Critical Step</p>
<p><u>STEP 7:</u> Determine if ECC bias is applicable.</p> <p><u>STANDARD:</u> SRO will N/A this step per the initial conditions.</p> <p><u>COMMENTS:</u></p>	<p>___ SAT</p> <p>___ UNSAT</p>

Job Performance Checklist:

STEP/STANDARD	SAT/UNSAT
<p>STEP 8: Calculate the reactivity worth difference between Critical and current boron concentrations.</p> <p>STANDARD: SRO will determine the reactor engineers's calculation is incorrect due to incorrect differential boron worth used in the equation. He should arrive at 737 pcm (+ or - 1)</p> <p>This step is critical to ensure ECP calculation integrity.</p> <p>COMMENTS:</p>	<p>___ SAT</p> <p>___ UNSAT</p> <p>Critical Step</p>
<p>STEP 9: Calculate the necessary inserted reactivity worth of control banks for the current boron concentration.</p> <p>STANDARD: SRO will determine that the entry is incorrect based on incorrect data input and arrive at the correct value of -573 pcm (+ or - 1)</p> <p>This step is critical to ensure ECP calculation integrity.</p> <p>COMMENTS:</p>	<p>___ SAT</p> <p>___ UNSAT</p> <p>Critical Step</p>
<p>STEP 10: Record the HZP peak Xe worth at for the current cycle burnup.</p> <p>STANDARD: SRO will use Table 6-38 of the NDR and determine the Reactor Engineer's value is correct.</p> <p>COMMENTS:</p>	<p>___ SAT</p> <p>___ UNSAT</p>

Job Performance Checklist:

STEP/STANDARD	SAT/UNSAT
<p><u>STEP 11:</u> <u>Calculate the ECP rod position.</u></p> <p><u>STANDARD:</u> SRO interpolates data using NDR table 6-32 to determine the Reactor Engineers ECP is incorrect. The correct ECP is 138 steps on Control Bank D. (+ or – 2 steps)</p> <p> This step is critical to ensure ECP calculation integrity.</p> <p><u>COMMENTS:</u></p>	<p>___ SAT</p> <p>___ UNSAT</p> <p>Critical Step</p> <p>Stop time ____</p>

End Of JPM

KEY

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 22 of 42
----------	-------------------------------	----------------------------------------------

APPENDIX B
Page 5 of 10

DATA SHEET 2
ECC CALCULATION USING NDR DATA

6.1.2 ECC, Unit Conditions

[1] Unit 1 ECC applicable from 9/27/10/0100 to 9/27/10/0300
Date Time Date Time

[2] DETERMINE THE FOLLOWING INFORMATION

A. Present RCS Boron concentration 1600 ppm

B. Core average burnup 4000 MWD/MTU

C. Date and time of shutdown. 9/20/10/0100
Date Time

D. Record average fraction of rated power for last 4 days before shutdown.

Power fraction = 1.00

E. Record the change in samarium/plutonium worth from equilibrium to peak from the HZP samarium/plutonium worth table in the NDR at the cycle burnup recorded in step 6.1.2[2]B.(Table 6-39)

Sm Worth = -385 pcm Pu Worth = 204 pcm

KEY

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 23 of 42
----------	-------------------------------	----------------------------------------------

APPENDIX B

Page 6 of 10

DATA SHEET 2

ECC CALCULATION USING NDR DATA

6.1.2 ECC, Unit Conditions (Continued)

- F. Calculate the change in samarium/plutonium worth from equilibrium using values recorded in step 6.1.2[2]E above and the equation below. There are examples of this calculation in Appendix D.

The change in samarium/plutonium worth = [Average fraction of rated power for the last 4 days, step 6.1.2[2]D] x [(change in samarium worth from equilibrium to peak, step 6.1.2[2]E) x (1 - e^{-λ PM149t}) + (change in plutonium worth from equilibrium to peak, step 6.1.2[2]E) x (1 - e^{-λ NP239t})]

λPM149 = .01305 hours⁻¹

λNP239 = .01229 hours⁻¹

t = time since shutdown in hours = 168 hrs

[1.00] * [(-385) * (1 - e^{-0.01305 * 168}) + (204) * (1 - e^{-0.01229 * 168})]

Change in Sm/Pu Worth = -164

Change in Sm/Pu Worth = -164 pcm

- G. Calculate the Xenon worth at time t recorded in step 6.1.2[2]F using TI-33.

Xenon worth = 0 pcm

- H. From NDR determine the design critical boron concentration (ARO, HZP, No Xe, HFP Eq Sm and Pu) at the cycle burnup recorded in step 6.1.2[2]B. (Table 6-1)

1709 ppm

KEY

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 24 of 42
----------	-------------------------------	----------------------------------------------

APPENDIX B

Page 7 of 10

DATA SHEET 2

ECC CALCULATION USING NDR DATA

6.1.2 ECC, Unit Conditions (Continued)

- I. From NDR determine the ARO, HZP Differential boron worth at the cycle burnup recorded in step 6.1.2[2]B and the boron concentration recorded in step 6.1.2[2]A. (Table 6-8)

$$\underline{-6.765} \text{ pcm/ppm}$$

- J. Based upon the unit cycle experience with previous ECC's or deviations of predicted RCS boron with burnup, adjustments between -100 ppm and +100 ppm may be necessary to reflect any nuclear design bias due to calculational methods or B-10 depletion. (Refer to Appendix E for bias calculation)

$$\text{ECC bias} = \underline{N/A} \text{ ppm (if applicable)}$$

NOTE1 If the results of Step 6.1.2[2]K are Negative, criticality can NOT be attained and the present RCS boron concentration needs to be reduced.

- K. Calculate reactivity worth of difference between Critical Boron Concentration (ARO, HZP, No Xe, HFP Eq Sm and Pu) and present RCS boron concentration.

$$\begin{aligned} \text{Boron Difference Reactivity Worth} &= \left[\begin{array}{l} \text{Present} \\ \text{Boron Concent.} - \left(\begin{array}{l} \text{Critical boron} \\ \text{Concentration} \\ \text{ARO, HZP, No Xe} \\ \text{Eq Sm (Step 6.1.2[2]H)} \end{array} \right) + \text{ECC Bias} \\ \text{STEP 6.1.2[2]A} \qquad \qquad \qquad \text{Step 6.1.2[2]J} \end{array} \right] \times \left[\begin{array}{l} \text{ARO, HZP} \\ \text{Differential} \\ \text{Boron Worth} \\ \text{Step 6.1.2[2]I} \end{array} \right] \\ &= \left[\frac{1600}{\text{Step 6.1.2[2]A}} \text{ ppm} - \left(\frac{1709}{\text{Step 6.1.2[2]H}} \text{ ppm} + \frac{0}{\text{Step 6.1.2[2]J}} \text{ ppm} \right) \right] \times \left[\frac{-6.765}{\text{Step 6.1.2[2]I}} \text{ pcm / ppm} \right] \\ &= \underline{737} \text{ pcm} \end{aligned}$$

KEY

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 25 of 42
----------	-------------------------------	----------------------------------------------

APPENDIX B
Page 8 of 10

DATA SHEET 2
ECC CALCULATION USING NDR DATA

6.1.2 ECC, Unit Conditions (Continued)

NOTE2 Results should be negative. If results are positive, boron concentration is still too high.

L. Calculate necessary inserted reactivity worth of control banks at the present RCS boron concentration.

$$\begin{aligned}
 \text{Inserted Control Bank Reactivity Worth} &= - \left(\left[\begin{array}{c} \text{Boron Diff.} \\ \text{Reactivity Worth} \\ \text{(Step 6.1.2[2]K)} \end{array} \right] + \left[\begin{array}{c} \text{Xenon} \\ \text{Worth} \\ \text{Step 6.1.2[2]G)} \end{array} \right] + \left[\begin{array}{c} \text{Difference from} \\ \text{Sm / Pu Worth} \\ \text{(Step 6.1.2[2]F)} \end{array} \right] \right) \\
 &= - \left(\frac{737}{\text{Step 6.1.2[2]K}} \text{ pcm} + \frac{0}{\text{Step 6.1.2[2]G}} \text{ pcm} + \frac{-164}{\text{Step 6.1.2[2]F}} \text{ pcm} \right) \\
 &= \underline{-573} \text{ pcm}
 \end{aligned}$$

M. Record HZP peak Xe worth at the cycle burnup recorded in step 6.1.2[2]B from the NDR table of HZP equilibrium and peak Xe worth. (Table 6-38)

$$\underline{-3980} \text{ pcm}$$

N. Calculate ECP rod position by interpolating between the NDR HZP integral rod worth tables with peak and no Xe using the peak Xe worth recorded in step 6.1.2[2]M and the Xe worth recorded in step 6.1.2[2]G. Interpolate within the tables based on the integral rod worth recorded in step 6.1.2[2]L and the cycle burnup recorded in step 6.1.2[2]B. (Tables 6-32 and 6-33)

$$\text{steps} = \underline{138}$$

READ TO OPERATOR

Directions to Trainee:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM. 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. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

1. Unit 1 is in Mode 3 and preparations are in progress to start up and take the reactor critical.
2. Core Average Burnup is 4000 MWD/MTU.
3. The present RCS Boron concentration is 1600 ppm.
4. Unit 1 tripped from full power at 0100 hours on 09/20/2010.
5. Unit 1 had been at 100% power for 2 weeks prior to the trip.
6. The ECP procedure will be applicable from 0100 until 0300 hours on 09/27/2010.
7. ECC bias is not applicable.

INITIATING CUES:

You are the Unit 1 SRO and the Reactor Engineer has requested you to review 0-SI-NUC-000-001.0, "Estimated Critical Conditions," Appendix B, Data Sheet 2, "ECC Calculation Using NDR Data."

1. Identify any and all errors on the provided Estimated Critical Conditions, Appendix B data sheet.

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 22 of 42
----------	-------------------------------	----------------------------------------------

APPENDIX B
Page 5 of 10

DATA SHEET 2
ECC CALCULATION USING NDR DATA

6.1.2 ECC, Unit Conditions

~~(11)~~ Unit 1 ECC applicable from 9/27/10/ 0100 to 9/27/10/ 0300
Date Time Date Time

~~(12)~~ DETERMINE THE FOLLOWING INFORMATION

~~(A)~~ Present RCS Boron concentration 1600 ppm

~~(B)~~ Core average burnup 4000 MWD/MTU

~~(C)~~ Date and time of shutdown. 9/20/10/ 0100
Date Time

~~(D)~~ Record average fraction of rated power for last 4 days before shutdown.
Power fraction = 1.00

~~(E)~~ Record the change in samarium/plutonium worth from equilibrium to peak from the HZP samarium/plutonium worth table in the NDR at the cycle burnup recorded in step 6.1.2[2]B.(Table 6-39)

Sm Worth = -415 pcm Pu Worth = 204 pcm

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 23 of 42
----------	-------------------------------	----------------------------------------------

APPENDIX B
Page 6 of 10

DATA SHEET 2
ECC CALCULATION USING NDR DATA

6.1.2 ECC, Unit Conditions (Continued)

~~F~~ Calculate the change in samarium/plutonium worth from equilibrium using values recorded in step 6.1.2[2]E above and the equation below. There are examples of this calculation in Appendix D.

The change in samarium/plutonium worth = [Average fraction of rated power for the last 4 days, step 6.1.2[2]D] x [(change in samarium worth from equilibrium to peak, step 6.1.2[2]E) x (1 - e^{-λ PM149t}) + (change in plutonium worth from equilibrium to peak, step 6.1.2[2]E) x (1 - e^{-λ NP239t})]

λPM149 = .01305 hours⁻¹

λNP239 = .01229 hours⁻¹

t = time since shutdown in hours = 168 hrs

$[1.00] * [(-415)] * (1 - e^{-0.01305 * 168}) + (204) * (1 - e^{-0.01229 * 168})$

Change in Sm/Pu Worth = -190.5

Change in Sm/Pu Worth = -190.5 pcm

~~G~~ Calculate the Xenon worth at time t recorded in step 6.1.2[2]F using TI-33.

Xenon worth = 0 pcm

~~H~~ From NDR determine the design critical boron concentration (ARO, HZP, No Xe, HFP Eq Sm and Pu) at the cycle burnup recorded in step 6.1.2[2]B. (Table 6-1)

1709 ppm

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 24 of 42
----------	-------------------------------	----------------------------------------------

APPENDIX B
Page 7 of 10

DATA SHEET 2
ECC CALCULATION USING NDR DATA

6.1.2 ECC, Unit Conditions (Continued)

~~I~~ From NDR determine the ARO, HZP Differential boron worth at the cycle burnup recorded in step 6.1.2[2]B and the boron concentration recorded in step 6.1.2[2]A. (Table 6-8)

-6.723 pcm/ppm

~~J~~ Based upon the unit cycle experience with previous ECC's or deviations of predicted RCS boron with burnup, adjustments between -100 ppm and +100 ppm may be necessary to reflect any nuclear design bias due to calculational methods or B-10 depletion. (Refer to Appendix E for bias calculation)

ECC bias = NIA ppm (if applicable)

NOTE1 If the results of Step 6.1.2[2]K are Negative, criticality can NOT be attained and the present RCS boron concentration needs to be reduced.

~~K~~ Calculate reactivity worth of difference between Critical Boron Concentration (ARO, HZP, No Xe, HFP Eq Sm and Pu) and present RCS boron concentration.

$$\begin{aligned}
 \text{Boron Difference Reactivity Worth} &= \left[\begin{array}{l} \text{Present} \\ \text{Boron Concent.} - \left(\begin{array}{l} \text{Critical boron} \\ \text{Concentration} \\ \text{ARO, HZP, No Xe} \\ \text{Eq Sm (Step 6.1.2[2]H)} \end{array} \right) + \begin{array}{l} \text{ECC Bias} \\ \text{Step 6.1.2[2]J} \end{array} \end{array} \right] \times \left[\begin{array}{l} \text{ARO, HZP} \\ \text{Differential} \\ \text{Boron Worth} \\ \text{Step 6.1.2[2]I} \end{array} \right] \\
 &= \left[\frac{1600}{\text{Step 6.1.2[2]A}} \text{ ppm} - \left(\frac{1709}{\text{Step 6.1.2[2]H}} \text{ ppm} + \frac{\text{NIA}}{\text{Step 6.1.2[2]J}} \text{ ppm} \right) \right] \times \left[\frac{-6.723}{\text{Step 6.1.2[2]I}} \text{ pcm / ppm} \right] \\
 &= \underline{732.8} \text{ pcm}
 \end{aligned}$$

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT

TECHNICAL INSTRUCTION

TI-33

XENON WORTH CALCULATION

Revision 25

QUALITY RELATED

PREPARED BY: KATHRYN ALLEN

RESPONSIBLE ORGANIZATION: Reactor Engineering

APPROVED BY: MICHAEL R. HOWARD

EFFECTIVE DATE: 4/22/09

VERIFICATION DATE: N/A

LEVEL OF USE: REFERENCE USE

SQN 1,2	XENON WORTH CALCULATION	TI-33 Rev 25 Page 2 of 29
------------	-------------------------	---------------------------------

REVISION
DESCRIPTION

Revised to update the Point Model Uncertainty Factors
for Unit 1 Cycle 17.

TABLE OF CONTENTS
Page 1 of 1

Section Title	Page
TABLE OF CONTENTS	3
1.0 INTRODUCTION	4
1.1 Purpose	4
1.2 Scope.....	4
2.0 REFERENCES.....	4
2.1 Performance References	4
2.2 Developmental References.....	4
3.0 THEORY OF XENON-135 REACTIVITY ON CORE	5
4.0 INSTRUCTIONS/GUIDANCE	13
4.1 Methods of Obtaining Xenon	13
4.2 Xenon for Estimated Critical Boron (ECB) or Estimated Critical Condition (ECC)	14
4.3 Xenon for Shutdown Margin Calculation (SDM)	16
4.3.1 Obtain Xenon From REACTF to Use in SDM Calculation	16
4.3.2 Obtain Xenon From ICS to Use in SDM Calculation	18
4.3.3 Obtain Xenon From NDR to Use in SDM Calculation.....	19
4.3.4 Corrections to Xenon Worth Used in SDM Calculation	20
4.4 Xenon for Use in Reactivity Balance of 0-SO-62-7.....	23
5.0 RECORDS	23
APPENDIXES	
APPENDIX A: XENON CALCULATION DATA SHEET FOR ECB OR ECC	25
APPENDIX B: XENON CALCULATION DATA SHEET FOR SDM.....	26
APPENDIX C: XENON MANAGEMENT NEAR END OF CYCLE	27
APPENDIX D: UNIT 1 AND 2 POINT MODEL UNCERTAINTY FACTORS	28
APPENDIX E: CORRELATION OF XENON SOURCES TO XENON USES AND THE ASSOCIATED CORRECTIONS.....	29

SQN 1,2	XENON WORTH CALCULATION	TI-33 Rev 25 Page 4 of 29
------------	-------------------------	---------------------------------

1.0 INTRODUCTION

1.1 Purpose

This Technical Instruction provides the guidance and methods necessary for the calculation of xenon worth for core reactivity calculations. The methods explained will be:

- A. Use of Xenon tables in the Nuclear Design Report (NDR).
- B. Use of the REACTF computer program.
- C. Use of the plant computer (ICS).

1.2 Scope

This TI is used to determine the method of calculation and any corrections to Xenon Worth based on the power history and length of time from shutdown until time of reactivity calculation.

2.0 REFERENCES

2.1 Performance References

- A. Applicable Unit/Cycle NDR or other vendor data
- B. REACTF Users Guide
- C. Operator's Manual for plant computer
- D. 0-SO-62-7, "Boron Concentration Control"

2.2 Developmental References

- A. Framatome Technologies Technical Document 64-1245312-00
- B. Introduction to Nuclear Power - Tennessee Valley Authority, Second Edition

SQN 1,2	XENON WORTH CALCULATION	TI-33 Rev 25 Page 5 of 29
------------	-------------------------	---------------------------------

3.0 THEORY OF XENON-135 REACTIVITY ON CORE

The reactivity effect of xenon (Xe)-135 and samarium (Sm)-149 are things which occur relatively slowly and their influence is felt over periods of time ranging from hours to days and years.

Xe-135 is but one of the ~200 possible fission products. Its importance is a result of the fact that its microscopic absorption cross section, σ_a , is ~3,500,000 barns. It is of note that most isotopes, excepting the small group considered along with Xe-135 as nuclear poisons, have values of σ_a in the range of 0.01 to 10 barns. The fact that σ_a for Xe 135 is high would be of no particular consequence if there were little or no atoms of this isotope formed. However, Xe-135 is one of the most common of all fission products. It is formed to a small extent (0.3%) as a direct fission product. The percentage indicates the number of Xe-135 nuclei formed per 100 fissions. The major source of Xe-135 however, is the decay by β^- emission of iodine (I)-135, an isotope which is formed in 5.9% of all fissions⁽¹⁾. The half-life of I-135 is 6.7 hours. The combination of these two sources makes the total effective fission yield of Xe-135 equal to 6.2%. That is, an average of 6.2 atoms of Xe-135 result from each 100 fissions.

- (1) Actually, tellurium-135 (Te-135), not I-135 is the direct fission product. Te-135 decays by β^- emission to I-135 with a two minute half-life according to the equation:



Since the half-life of Te-135 is so short, the effect of its presence is of little consequence, and most analyses assume that I-135 is the direct fission product.

3.0 THEORY OF XENON-135 REACTIVITY ON CORE (Continued)

Xe-135 and its poisoning effect on the reactor are removed by two mechanisms: 1) its decay by β^- emission with a 9.2 hour half-life to Cs-135 (cesium), and 2) its capture of a neutron with the resulting formation of stable Xe-136. When a reactor of the type we are considering is operated at high power, very roughly half of the Xe-135 formed is removed by each of these mechanisms.

Figure 3.1 summarizes the life history of Xe-135, and an understanding of it is fundamental to comprehension of the effect of Xe-135 on a reactor. The net amount of xenon in the core is dependent upon the relative rates of its formation and disappearance. In other words:

$$\left[\begin{array}{c} \text{Rate of change of} \\ \text{xenon level} \end{array} \right] = \left[\begin{array}{c} \text{Rate of production from direct} \\ \text{fission and from decay of iodine} \end{array} \right] - \left[\begin{array}{c} \text{Rate of removal by decay} \\ \text{and by neutron capture} \end{array} \right]$$

When the rate of production exceeds the rate of removal, the xenon inventory in the core will be increasing, and vice-versa. With this background let us consider a number of the more important XE-135 transients.

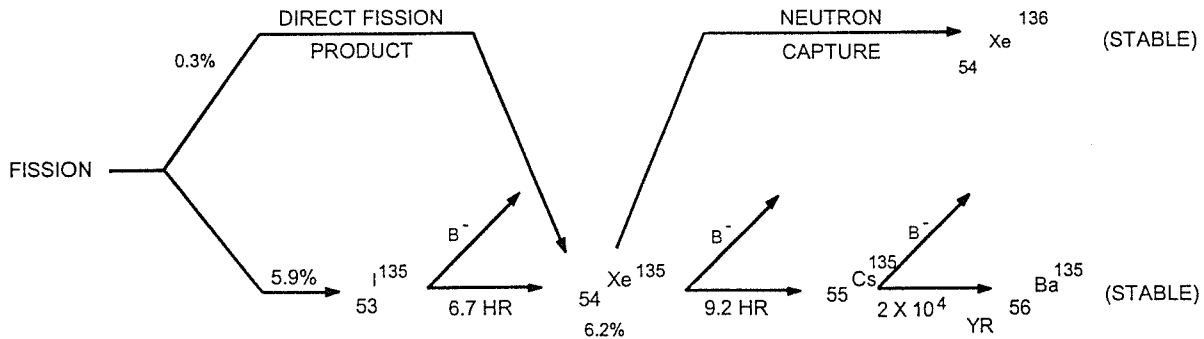


Figure 3-1 Life History of Xenon-135 in a Reactor

3.0

THEORY OF XENON-135 REACTIVITY ON CORE (Continued)

[1] Startup from the Xe free condition

A reactor which has never run at power before, or a reactor which has been shutdown for several days is said to be in the Xe free condition.

Figure 3.2 shows the reactivity effect of xenon as such a reactor is quickly brought to full power.

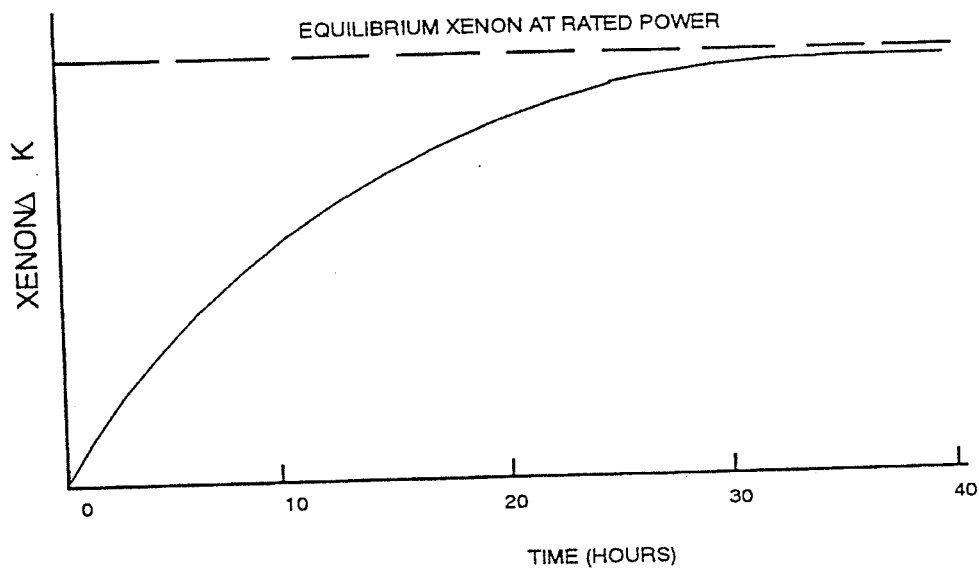


Figure 3.2: Xenon Transient after Startup from Zero Power to Full Power of a Reactor which was Initially Xenon Free

3.0 THEORY OF XENON-135 REACTIVITY ON CORE (Continued)

The figure shows that as soon as the reactor reaches power, the xenon starts building up in the core and gradually approaches an equilibrium value. This equilibrium value is reached when the rate of production of xenon just equals the rate at which it is removed. Note that both the rate of production and the rate of removal of xenon depend upon reactor power (or equivalently upon neutron flux). It follows from this that for each value of power or neutron flux there is a corresponding value of equilibrium xenon reactivity worth. In general, the higher the power, the higher the equilibrium xenon worth. This relationship is shown in figure 3-3. It can be seen that the equilibrium reactivity effect of xenon is negligible at low power.

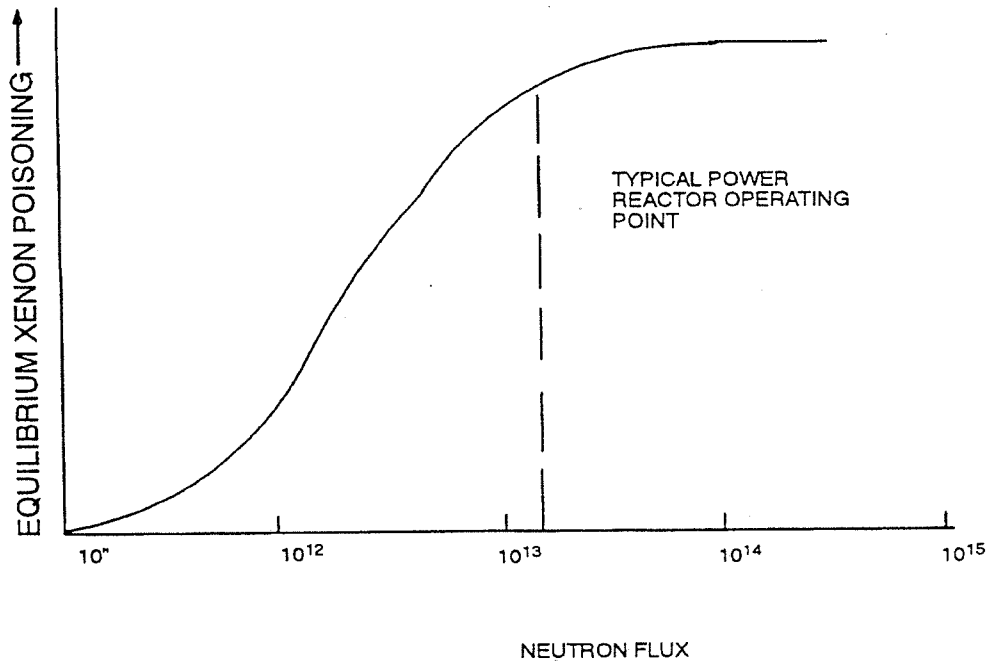


Figure 3-3: Equilibrium Xenon Worth versus Neutron Flux

3.0 THEORY OF XENON-135 REACTIVITY ON CORE (Continued)

The buildup of xenon to its equilibrium value from the xenon free condition at startup can be explained by referring to the diagram of xenon's life history. It can be seen that most of the xenon is formed by the decay of I-135. Although creation of significant quantities of I-135 commences immediately upon attainment of full power, the xenon appears at a more gradual rate due to the half-life of I-135. Then, as the amount of xenon in the core increases, the rate at which it is removed also increases (the rate at which the xenon absorbs neutrons is proportional to the quantity of xenon present as is the rate at which it decays radioactively) until finally it catches up with the rate of production and equilibrium is reached. Actually, from a theoretical standpoint, xenon equilibrium is never reached. From figure 3-2 it can be seen that xenon worth approaches its equilibrium value asymptotically. For practical purposes, xenon is so close to equilibrium after 30 or 40 hours that it can be considered at equilibrium.

[2] Reactor shutdown from equilibrium xenon

If a reactor is shut down after having run long enough to reach equilibrium xenon, the behavior shown in figure 3-4 occurs. Sometime between 8 and 12 hours after shutdown the amount of xenon in the core, or equivalently its reactivity worth, increases to a peak value and then decays away to the xenon free condition with a half-life that eventually approaches the 9.2 hour half-life of Xe-135.

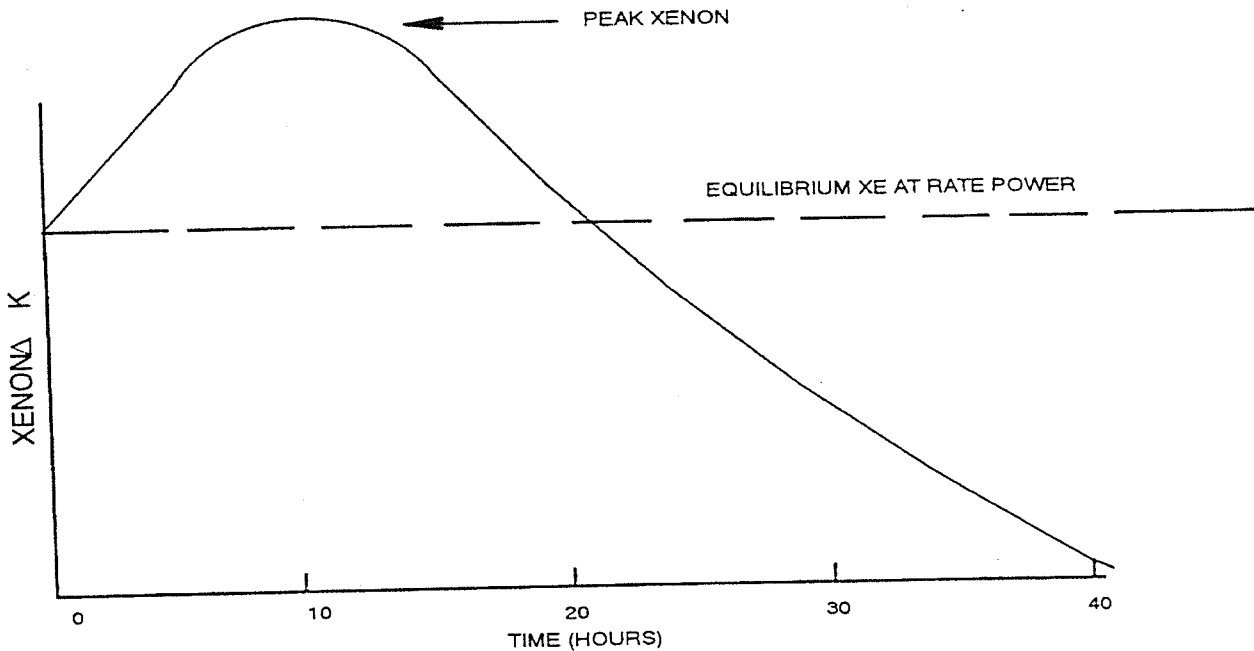


Figure 3-4: Xenon Behavior Following Reactor Shutdown from Full Power, Equilibrium Xenon condition

SQN 1,2	XENON WORTH CALCULATION	TI-33 Rev 25 Page 10 of 29
------------	-------------------------	----------------------------------

3.0 THEORY OF XENON-135 REACTIVITY ON CORE (Continued)

This behavior can be explained as follows:

- A. At the time the reactor is shutdown, equilibrium amounts of I-135 and Xe-135 are present. At the shutdown, the production of I-135 and Xe-135 by direct fission stops. However, because of the I-135 already in the core, the production of Xe-135 by decay of I-135 continues at essentially the same rate as before the shutdown. Furthermore, after the shutdown, the Xe-135 is no longer removed by neutron capture to any significant extent. The only removal mechanism for Xe-135 is by radioactive decay. Therefore, since the production of Xe-135 continues at almost the equilibrium rate, and since the rate at which Xe-135 is removed is significantly diminished, the concentration of Xe-135 in the core initially builds up following the shutdown.
- B. As time passes, the rate at which Xe-135 is being produced diminishes because the I-135 concentration diminishes. Eventually a point is reached where the Xe-135 production and removal rates are equal, and the amount of Xe-135 in the core levels off. This is the point of peak xenon.
- C. Still later in time, the rate at which xenon is removed by decay exceeds the rate at which it is produced from I-135. Therefore the xenon inventory in the core begins to decrease. Finally, as the I-135 in the core is nearly depleted, the rate at which the Xe-135 inventory decreases is governed by, and approaches, its 9.2 hour half-life.

If a reactor is able to start up when xenon is at its peak, it is said to have sufficient reactivity in control rods to "override" peak xenon. A typical power reactor usually can override peak xenon during early core life. Towards the end of core life, it may not be able to do this. After a scram from power, it would be necessary to get started up and reach power within, say two hours, or else the reactor could not again be made critical for about 20 hours.

3.0 THEORY OF XENON-135 REACTIVITY ON CORE (Continued)

[3] Reactor startup with xenon in the core

Let us further pursue the case of a startup after a shutdown from power with xenon in the core. This behavior is shown in Figure 3-5. Between the time the reactor is shutdown and reaches power again, the first portion of the xenon buildup curve after shutdown (Figure 3-4) can be seen. Just as soon as the reactor returns to power, the xenon inventory begins to drop. This is due to the fact that once the reactor reaches power, xenon again begins to be removed because of neutron capture (as well as by decay). However, its rate of production has fallen off from the equilibrium value because decay during the shutdown time has depleted the I-135 inventory. Temporarily, therefore, the rate of xenon removal exceeds the rate of production. The I-135 inventory starts to recover immediately upon the return to power. Its half-life, however, introduces a lag of several hours before the xenon begins to recover. Eventually, the xenon also returns to the equilibrium value for the power level in question. The initial drop in xenon inventory following a return to power is often called xenon burnout.

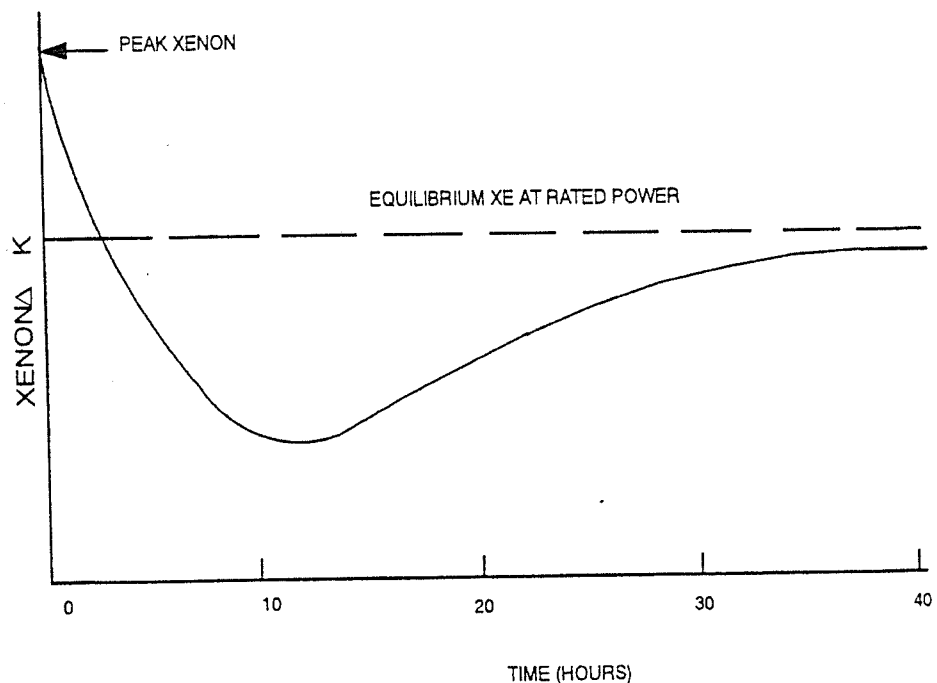


Figure 3-5: Xenon Behavior for Reactor Startup with Xenon in the Core

3.0 THEORY OF XENON-135 REACTIVITY ON CORE (Continued)

[4] Xenon behavior on load changes

With the information presented so far, the effect of xenon during load increases and reductions, as shown in Figures 3-6 and 3-7, should be understandable to the reader.

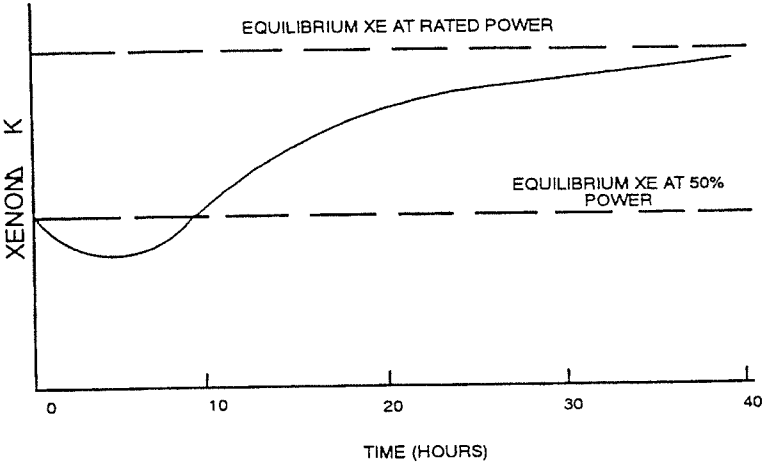


Figure 3-6: Xenon Behavior on a Load Change from 50% Power to 100% Power

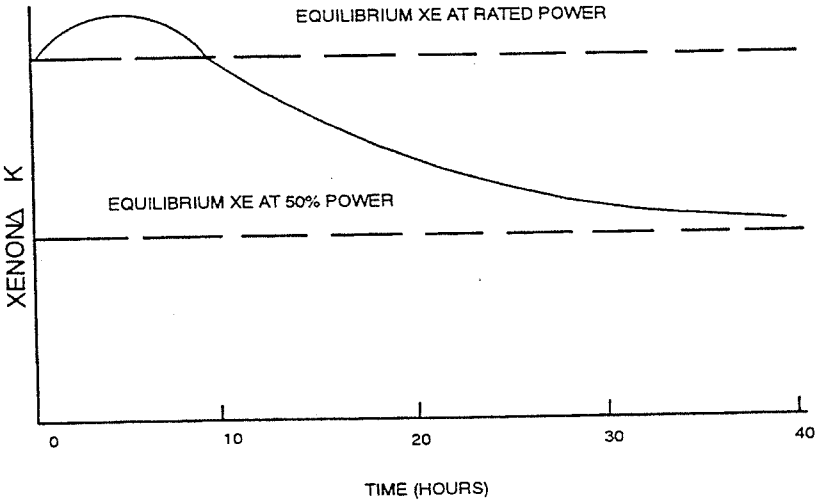


Figure 3-7: Xenon Behavior on a load Change form 100% Power to 50% Power

SQN 1,2	XENON WORTH CALCULATION	TI-33 Rev 25 Page 13 of 29
------------	-------------------------	----------------------------------

4.0 INSTRUCTIONS/GUIDANCE

NOTE Refer to Appendix E for a correlation of xenon sources, xenon uses and necessary corrections to the xenon worth.

4.1 Methods of Obtaining Xenon

NOTE The xenon worth and/or xenon percent of full power concentration can be obtained from the sources provided below.

A. Use of REACTF

REACTF is located on PC computers in the Control room or in the Reactor Engineering office. Familiarization with the Users Guide for REACTF will allow proper utilization of Xenon Calculations.

Select windows icon for REACTF or go to the appropriate directory and execute the appropriate executable file and perform calculations.

B. Use of plant computer (ICS).

Familiarization with the Operator's Manual will allow proper utilization of calling up values.

Computer points of interest:

U2108-Xenon Worth pcm

U2107-Xenon Concentration %

U2106-Iodine Concentration %

C. Use of Nuclear Design Report (NDR)

Choose the correct value from the appropriate table based on burnup and Reactor Power, interpolation may be required. The NDR can only be used for either equilibrium reactor power or trips from equilibrium reactor power conditions. To obtain xenon percent of full power concentration, divide the xenon worth in question by the 100% reactor power xenon worth (which can be found from REACTF or the NDR).

SQN 1,2	XENON WORTH CALCULATION	TI-33 Rev 25 Page 14 of 29
------------	-------------------------	----------------------------------

4.0 INSTRUCTIONS/GUIDANCE (continued)

4.2 Xenon for Estimated Critical Boron (ECB) or Estimated Critical Condition (ECC)

NOTE 1 The xenon used in REACTF for an ECB or ECC is for the time that the unit went subcritical. The input for xenon into REACTF is the xenon percent concentration of full power.

NOTE A xenon worth calculated at the beginning of the four hour surveillance interval for ECC may be off by as much as 500 pcm if the criticality occurs at the end of the surveillance interval and xenon is changing rapidly during the four hour surveillance interval.

NOTE 3 For ECC calculations, it is suggested that a xenon worth calculation be performed at the closest approximation of the time to when criticality will occur within the four hour surveillance interval.

NOTE 4 For ECC calculations, it is advisable to use the initial xenon worth from steady state conditions and to use this value to calculate the xenon worth at time of startup using the power history of the unit.

NOTE 5 DO NOT apply the Point model Uncertainty Factor (PUF) to the Xenon worth's used for ECB or ECC since a best estimate value instead of a conservative value is needed.

SQN 1,2	XENON WORTH CALCULATION	TI-33 Rev 25 Page 15 of 29
------------	-------------------------	----------------------------------

4.0 INSTRUCTIONS/GUIDANCE (continued)

4.2 Xenon for Estimated Critical Boron (ECB) or Estimated Critical Condition (ECC) (continued)

- A. Obtain the best estimate value of xenon percent concentration of full power by either:
 - 1. Using the value at time of trip or subcriticality from the ICS (this value can be compared to the value generated by REACTF).
 - 2. Using REACTF by modeling the power history prior to unit going subcritical (using an initial xenon worth from steady state conditions).

- B. If using REACTF to calculate ECB and/or ECC then N/A or discard Appendix A. If not using REACTF to calculate ECB and/or ECC then document xenon worth or xenon % concentration of full power for an ECB or ECC on Appendix A.

SQN 1,2	XENON WORTH CALCULATION	TI-33 Rev 25 Page 16 of 29
------------	-------------------------	----------------------------------

4.0 INSTRUCTIONS/GUIDANCE (continued)

4.3 Xenon for Shutdown Margin Calculation (SDM)

NOTE 1 For SDM calculations, it is necessary that a xenon worth calculation be performed at the correct time (starting time of surveillance interval) and another xenon worth at the surveillance interval end time. The lesser of the xenon worths (least negative) shall be used in the SDM calculation for xenon. However, if using the computer code REACTF, the Xenon worth used in the code MUST be at the time of trip or when the unit goes subcritical from a controlled shutdown. REACTF automatically takes the lesser of the xenon worths for the time interval of the surveillance

NOTE Section 4.3.4 explains the conditions affecting the reduction of Xenon Worth for Shutdown Margin.

1. 4.3.1 Obtain Xenon From REACTF to Use in SDM Calculation

NOTE For a Shutdown Margin Calculation, the Xe and I worths (% concentration of full power) used in the computer code REACTF MUST be entered for time of trip or when the unit goes subcritical from a controlled shutdown.

A. If xenon is taken from the Xenon calculation option of REACTF and used in the SDM option of REACTF then perform the following steps:

NOTE For steady state Rx conditions use equilibrium Xe and I at time of shutdown. Do not use steady state Rx conditions and "Input Xe and I at time of shutdown."

1. Execute REACTF for the appropriate plant conditions.
2. Any corrections to xenon required due to plant conditions for the SDM calculation will be made by REACTF.
3. Appendix B is not required to be filled out. It may be N/A'd or discarded.

SQN 1,2	XENON WORTH CALCULATION	TI-33 Rev 25 Page 17 of 29
------------	-------------------------	----------------------------------

4.0 INSTRUCTIONS/GUIDANCE (continued)

4.3 Xenon for Shutdown Margin Calculation (SDM) (continued)

4.3.1 Obtain Xenon From REACTF to Use in SDM Calculation (continued)

NOTE It is unlikely that the following step would be performed since if REACTF is operating one would use it to perform the SDM calculation rather than performing it by hand.

B. If xenon is taken from REACTF and used in a hand calculation for SDM using the SDM procedure then perform the following steps:

1. Execute the appropriate portion of REACTF to obtain the xenon worth for each end of the surveillance interval that calculation is being performed for.
2. Determine if PUF and/or 10% reduction in xenon worth is necessary by referring to section 4.3.4 and then find the minimum xenon worth.
3. Complete Appendix B and include with the performance of SDM procedure.

SQN 1,2	XENON WORTH CALCULATION	TI-33 Rev 25 Page 18 of 29
-------------------	--------------------------------	----------------------------------

4.0 INSTRUCTIONS/GUIDANCE (continued)

4.3 Xenon for Shutdown Margin Calculation (SDM) (continued)

4.3.2. Obtain Xenon From ICS to Use in SDM Calculation

- A. If xenon is taken from ICS and used in REACTF for SDM calculation (transient condition) then perform the following steps (only xenon and iodine values of % concentration of full power is used for REACTF):

NOTE In order to use the Xe and I % concentration of full power in REACTF, the Rx conditions prior to shutdown must be transient.

1. Obtain the xenon % concentrations of full power and iodine % concentrations of full power (if needed) from ICS and enter into the REACTF SDM calculation.
 2. No corrections are necessary to xenon by the performer because REACTF will make corrections if needed.
 3. Appendix B is not required to be filled out. It may be N/A'd or discarded.
- B. If xenon is taken from ICS and used in a hand calculation for SDM using the SDM procedure then perform the following steps (the xenon worth, in pcm, is used in the SDM procedure hand calculation):
1. Obtain the xenon worth from ICS for each end of the surveillance interval that the SDM calculation is being performed for (will need to use the prediction portion of ICS xenon calculation).
 2. Determine if PUF and 10% reduction in xenon worth is necessary by referring to section 4.3.4 and then find the minimum xenon worth.
 3. Complete Appendix B and include with the performance of SDM procedure.

SQN 1,2	XENON WORTH CALCULATION	TI-33 Rev 25 Page 19 of 29
------------	-------------------------	----------------------------------

4.0 INSTRUCTIONS/GUIDANCE (continued)

4.3 Xenon for Shutdown Margin Calculation (SDM) (continued)

4.3.3. Obtain Xenon From NDR to Use in SDM Calculation

NOTE 1 Use of the NDR for xenon values is for equilibrium-based plant conditions.

NOTE 2 Xenon values from the NDR require no corrections (i.e. no corrections for PUF and no corrections for transient conditions).

NOTE 2 Use NDR table 6-40, 6-41, or 6-42 for the appropriate burnup to obtain the Xenon worths.

A. If xenon is taken from NDR and used in a hand calculation for SDM using the SDM procedure then perform the following steps:

1. Obtain the minimum xenon from the appropriate table of NDR (interpolation may be necessary) at plant conditions, burnup, time after plant trip or unit went subcritical, surveillance interval, and boron concentration.
2. Write the xenon worth in Appendix B and identify the method used. There is no PUF reduction or 10% reduction since the data is from equilibrium conditions, refer to section 4.3.4, thus N/A step A.3 of Appendix B.

<p style="text-align: center;">SQN 1,2</p>	<p style="text-align: center;">XENON WORTH CALCULATION</p>	<p>TI-33 Rev 25 Page 20 of 29</p>
------------------------------------------------	------------------------------------------------------------	-------------------------------------------

4.0 INSTRUCTIONS/GUIDANCE (continued)

4.3 Xenon for Shutdown Margin Calculation (SDM) (continued)

4.3.4. Corrections to Xenon Worth Used in SDM Calculation

NOTE 1 The Shutdown Margin option of REACTF Computer Program performs the appropriate reduction of xenon worth, if needed, in the program.

NOTE 2 For FCF methodology, a Point model Uncertainty Factor (PUF) should be applied when Xenon worth's are calculated by the point model and used for shutdown margin. The point model is used by the plant computers and the "XENON CALCULATION" option of REACTF. REACTF will automatically apply the PUF factor to the Xenon worths used for a SDM calculation if it is needed. The PUF adds additional conservatism to the Xenon worth to account for the inaccuracies of the point model. If using the equilibrium Xenon worth's from the Nuclear Design Report (NDR) it is not necessary to apply the PUF since the values from the NDR are calculated using a three dimensional model and not the point model.

The PUF factor is unit and cycle dependent and will be supplied before the start of each cycle by Nuclear Fuel. It accounts for non-conservative errors in the point model when compared to the three dimensional model which is used to calculate xenon worth's in the NDR. The PUF factors are provided in Appendix D.

SQN 1,2	XENON WORTH CALCULATION	TI-33 Rev 25 Page 21 of 29
------------	-------------------------	----------------------------------

4.0 INSTRUCTIONS/GUIDANCE (continued)

4.3 Xenon for Shutdown Margin Calculation (SDM) (continued)

4.3.4. Corrections to Xenon Worth Used in SDM Calculation (continued)

NOTE 3 Equilibrium power history is considered operating within $\pm 5\%$ about a power level for the past three days. If not at equilibrium power history then it is considered a transient power history. It is important to distinguish between the two since it impacts the correction factor applied to the SDM for conservatism.

- A. No correction to xenon is required for values taken from the NDR since the NDR values are calculated from a three-dimensional (3-D) model and are for equilibrium reactor conditions.

NOTE The SDM option of REACTF performs the appropriate reduction of xenon, if needed, in the program.

- B. If using the computer code REACTF (refer to Flowchart 1) then:

1. If the 3-D model (equilibrium conditions/steady state tables) is being used then no corrections are required.
2. For the point model case, if the time since shutdown is less than or equal to 15 hours and the reactor shutdown from unsteady conditions, the calculated xenon worth will include the standard 10% FCF xenon reduction and the corresponding PUF factor if any one or more of the following conditions exist:
 - a. Boron dilution is expected within the surveillance interval

SQN 1,2	XENON WORTH CALCULATION	TI-33 Rev 25 Page 22 of 29
------------	-------------------------	----------------------------------

4.0 INSTRUCTIONS/GUIDANCE (continued)

4.3 Xenon for Shutdown Margin Calculation (SDM) (continued)

4.3.4. Corrections to Xenon Worth Used in SDM Calculation (continued)

- b. The minimum expected core average temperature is less than HZP (547 deg F), or
 - c. The xenon transient is not well understood, causing the accuracy of the pretrip boron concentration to be in question.
3. For the point model, if the time since shutdown is greater than 15 hours and the reactor shutdown from unsteady conditions, the PUF and 10% reduction factors are automatically applied to the calculated xenon worth.

C. For xenon from the ICS:

1. If the time since shutdown is less than or equal to 15 hours and the reactor shutdown from unsteady conditions or steady state/equilibrium conditions, the calculated xenon worth shall include the standard 10% FCF xenon reduction factor and the corresponding PUF factor if any one or more of the following conditions exists:
 - a. Boron dilution is expected within the surveillance interval,
 - b. The minimum expected core average temperature is less than HZP (547 deg F), or
 - c. The xenon transient is not well understood, causing the accuracy of the pretrip boron concentration to be in question.
2. If the time since shutdown is greater than 15 hours and the reactor shutdown from unsteady conditions or steady state/equilibrium conditions, the PUF and 10% reduction factors shall be applied to the calculated xenon worth values.

SQN 1,2	XENON WORTH CALCULATION	TI-33 Rev 25 Page 23 of 29
------------	-------------------------	----------------------------------

4.0 INSTRUCTIONS/GUIDANCE (continued)

4.4 Xenon for Use in Reactivity Balance of 0-SO-62-7

NOTE No corrections are applied to the Xenon used in 0-SO-62-7 since a best estimate value instead of a conservative value is needed.

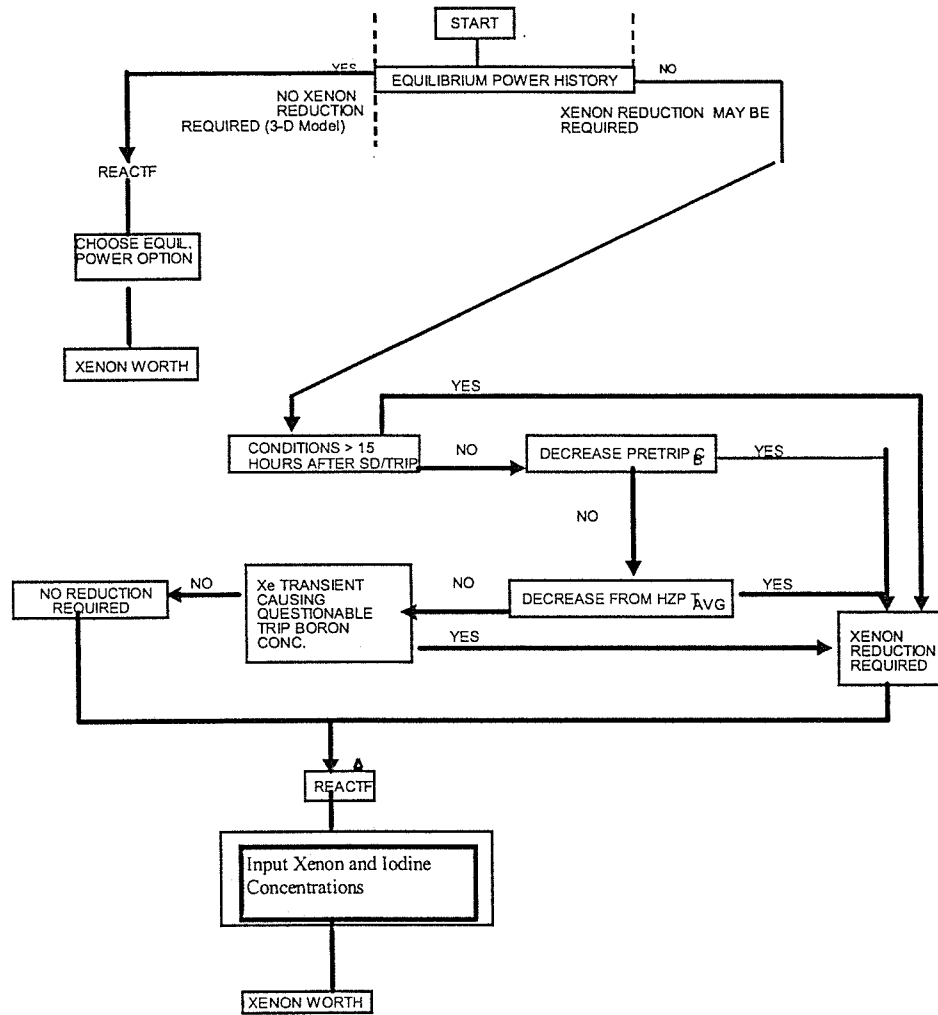
- A. Obtain the Xenon worth from ICS or REACTF and write these values into the appropriate location of 0-SO-62-7.

5.0 RECORDS

Appendix A and/or B (with computer printouts) becomes a QA record when used and is attached to the appropriate Data Package.

FLOWCHART 1

SDM METHODOLOGY XENON CALCULATION USING REACTF (Not Applicable for NDR or ICS)



A. REACTF USING CONDITIONS TAKES INTO CONSIDERATION WHETHER TO USE THE REDUCTION IN XENON WORTH FOR PUF AND ADDITIONAL 10%.

SQN 1,2	XENON WORTH CALCULATION	TI-33 Rev 25 Page 25 of 29
------------	-------------------------	----------------------------------

APPENDIX A
Page 1 of 1

XENON WORTH CALCULATION DATA SHEET FOR ECB OR ECC

UNIT _____

A. Mark method used and write Xenon worth or Xenon % Concentration of Full Power and attach to ECB or ECC printout.

1. Method used:

- REACTF
- Plant computer (ICS)
- Nuclear Design Report (NDR)

2. Xe worth _____ pcm

or

Xenon % Concentration of Full Power _____ %

DATA BY _____ / _____

SQN 1,2	XENON WORTH CALCULATION	TI-33 Rev 25 Page 26 of 29
------------	-------------------------	----------------------------------

APPENDIX B

Page 1 of 1

XENON WORTH CALCULATION DATA SHEET FOR SDM

UNIT _____

- A. Record Xenon worth and mark method used attaching printout.
1. Xe worth _____ pcm
 2. Method used. REACTF (for reduction refer to 4.3.4)
 Plant computer (for reduction refer to 4.3.4)
 Nuclear Design Report (no reduction required, N/A step 3)
 3. Determine if a reduction in xenon is required using Section 4.3.4 and record below (N/A if not required).
 - a. Record the burnup below and determine and record the PUF from Appendix D based on unit and burnup below:

[1] Burnup = _____ MWD/MTU

[2] PUF = _____

- b. Determine xenon worth using the following equation (which includes the 10% reduction)

$$\text{Xe worth} = \text{Step A.1} \times (0.9) \times \text{Step 3.a[2]}$$

$$\text{Xe worth} = \text{_____} \times (0.9) \times \text{_____}$$

$$\text{Xe worth} = \text{_____} \text{ pcm}$$

Data by _____ / _____

APPENDIX C

Page 1 of 1

XENON MANAGEMENT NEAR END OF CYCLE

Management of xenon near the end of an operating cycle (that is, RCS boron concentration 50 ppm or less) can present special challenges. Guidance for operation near end of cycle is contained in Section E (Power Coastdown at End of Life) of 0-GO-5, Normal Power Operation. The purpose of this appendix is to provide additional discussion of some of the potential difficulties associated with reactivity control and xenon management near the end of cycle (EOC). Reactor power level at any time is directly related to the reactivity balance. The primary influences on reactivity near EOC are RCS average temperature (moderator feedback) which is a function of reactor power when RCS temperature is on program, reactor power level (doppler feedback), control rod insertion, xenon worth, and RCS boron concentration until reduced near zero by dilution and demineralization. Another aspect of xenon management near EOC involves axial power stability decrease late in core life. Hence, it is important to dampen any axial flux difference oscillations early with minimal control rod motion. It is especially important near EOC to maintain as steady a reactor power level as possible (or to make any required power changes as slowly as possible, for example less than or equal to one percent per hour) in order to avoid inducing large xenon worth changes. For example, as shown in Figure 3-7, a power decrease transient causes excess xenon build in for the first few hours after the power change. Near EOC that negative reactivity might be difficult to offset with control rod motion or boron dilution which could result in further power decrease until the reactor shuts down. Thus, reactor operation can be constrained near EOC following too large or too fast a power change. Operating Experience report OE7023 reported two large axial flux deviations at EOC following two rapid down power maneuvers that resulted in reactor shutdown in one case and reduction to 26% power for 90 hours in the other case to dampen xenon-induced axial power oscillations. Similarly, reactor restart following a trip near EOC (or especially during power coast down) is difficult because boron is likely to have been added to the RCS to maintain shutdown margin as xenon decays off and large dilution volumes would be needed to reduce an already low RCS boron concentration. Even after restarting, it may not be possible to reach the same reactor power as before the trip because of excess samarium build in (see Figure 3-2 of TI-34) and difficulty removing RCS boron. Consequently, reactor restart following a trip during power coast down might not be advisable and should be carefully evaluated by management.

APPENDIX D
Page 1 of 1

UNIT1 AND 2 POINT MODEL UNCERTAINTY FACTORS

The data in this appendix is cycle dependent and must be updated for each cycle for each unit. The NDR will provide the values below.

UNIT 1 CYCLE 17 XENON POINT MODEL UNCERTAINTY FACTORS (PUF).

UNIT 1 CYCLE 17 NDR

Time in Life	Burnup Range (MWD/MTU)	PUF
BOL	0 - 10500	1.00
MOL	10500 - 15500	1.00
EOL	15500 - 21223	0.99

UNIT 2 CYCLE 16 XENON POINT MODEL UNCERTAINTY FACTORS (PUF).

Unit 2 CYCLE 16 NDR

Time in Life	Burnup Range (MWD/MTU)	PUF
BOL	0 - 10500	.97
MOL	10500 - 15500	1.0
EOL	15500 - 21355	1.0

APPENDIX E
Page 1 of 1

**CORRELATION OF XENON SOURCES TO XENON USES
AND THE ASSOCIATED CORRECTIONS**

The following tables are to assist the user in correlating the xenon sources to xenon uses and necessary corrections to the xenon worth.

Xenon Value Sources	Xenon Value Uses
1. NDR Tables	A. ECB/ECC (REACTF or Hand Calc)
2. REACTF Xenon Calc	B. REACTF SDM Calc
3. ICS Point	C. SDM Hand Calc

CORRELATION/CORRECTION MATRIX			
Xenon Source - Xenon Uses	Correction By User?	Reason	Reference Section
1 - A	No	Best Estimate Desired	4.2
2 - A	No	Best Estimate Desired	4.2
3 - A	No	Best Estimate Desired	4.2
1 - B *	N/A	N/A	N/A
2 - B	No	REACTF does necessary corrections	4.3.1.A
3 - B	No	REACTF does necessary corrections	4.3.2.A
1 - C	No	NDR is steady state, 3-D model, Section 4.3.4.A	4.3.3.A
2 - C **	Yes (if required)	Section 4.3.4.B.1, .2 and .3	4.3.1.B
3 - C	Yes (if required)	Section 4.3.4.C.1 and .2	4.3.2.B

*Not a viable option.

**It is unlikely that this option would be used since, if REACTF is up and running there is no need to perform a hand calculation, use REACTF.



PROPRIETARY

AREVA NP Inc.,
an AREVA and Siemens company

TECHNICAL DOCUMENT

Document No: 61 - 9103163 - 000

**Nuclear Design Report
for
Sequoyah Unit 1, Cycle 17**

AREVA NP INC. PROPRIETARY

This document and any information contained herein, is the property of AREVA NP Inc. (AREVA NP) and is to be considered proprietary and confidential and may not be reproduced or copied in whole or in part. Nor may it be furnished to others without the express written consent and permission of AREVA NP. nor may it be used in any way that is or may be detrimental to AREVA NP. This document and any copies that may have been made must be returned to AREVA NP upon request.



AREVA

AREVA NP Inc.,
an AREVA and Siemens company

Document No.: 61-9103163-000

PROPRIETARY

NDR - S1C17

Table 6-1 ARO Critical Boron Concentration at HZP and HFP Conditions

Critical Boron Concentration, ppm			
Cycle Life MWd/mtU	HZP (No Xenon)	HFP (HFP Equilibrium Xenon)	HFP (HFP Equilibrium Xenon, with Max. B ¹⁰ Depletion)
0	1770	1520 *	1526 *
150	1752	1101	1103
500	1750	1101	1107
1000	1729	1092	1093
2000	1724	1082	1104
4000	1709	1076	1120
6000	1715	1090	1157
8000	1677	1047	1134
10000	1579	928	1025
12000	1428	749	844
14000	1233	530	610
16000	1033	321	377
16201	1013	300	353
18000	835	117	140
19190	722	0	0
21223	524	0	0

Table Notes:

- * ≡ nc Xe
- ARO
- Eq Sm and Pu concentrations
- Data based on best estimate EOL 16 burnup of 19300 MWd/mtU (490.64 EFPD). A +/-2 ppm adjustment may be applied for each EFPD that cycle 16 length is different from the best estimate value.
- HFP Boron is 0 ppm at approximately 19150 MWd/mtU.
- Critical boron concentrations with B¹⁰ depletion are illustrative only. The actual extent of B¹⁰ depletion is dependent upon the reactor operating history, including previous borations.



AREVA NP Inc.,
an AREVA and Siemens company

NDR - S1C17

Table 6-8 HZP Differential Boron Worth

Boron Conc ppm	HZP Differential Boron Worth, pcm/ppm													
	Burnup, MWd/mtU													
	0	500	1000	2000	4000	6000	8000	10000	12000	14000	16000	18000	19180	21223
0	-7.197	-7.180	-7.167	-7.158	-7.188	-7.271	-7.392	-7.540	-7.707	-7.886	-8.078	-8.281	-8.408	-8.646
200	-7.120	-7.104	-7.093	-7.086	-7.120	-7.204	-7.325	-7.472	-7.636	-7.813	-8.000	-8.199	-8.324	-8.556
400	-7.051	-7.036	-7.027	-7.022	-7.058	-7.144	-7.265	-7.410	-7.571	-7.745	-7.928	-8.123	-8.245	-8.473
600	-6.989	-6.975	-6.967	-6.963	-7.002	-7.088	-7.209	-7.352	-7.511	-7.681	-7.860	-8.051	-8.170	-8.394
800	-6.932	-6.920	-6.912	-6.910	-6.950	-7.037	-7.156	-7.297	-7.453	-7.620	-7.796	-7.982	-8.099	-8.319
1000	-6.879	-6.868	-6.861	-6.860	-6.901	-6.987	-7.105	-7.245	-7.398	-7.561	-7.733	-7.916	-8.031	-8.248
1200	-6.830	-6.819	-6.812	-6.812	-6.854	-6.940	-7.056	-7.193	-7.344	-7.504	-7.673	-7.852	-7.965	-8.179
1400	-6.783	-6.772	-6.766	-6.766	-6.809	-6.894	-7.009	-7.143	-7.291	-7.448	-7.613	-7.790	-7.901	-8.112
1600	-6.738	-6.728	-6.722	-6.723	-6.765	-6.849	-6.962	-7.094	-7.239	-7.393	-7.555	-7.729	-7.839	-8.049
1800	-6.695	-6.685	-6.679	-6.680	-6.722	-6.805	-6.916	-7.045	-7.188	-7.339	-7.499	-7.671	-7.780	-7.988
2000	-6.654	-6.644	-6.639	-6.639	-6.681	-6.762	-6.871	-6.998	-7.139	-7.287	-7.445	-7.615	-7.723	-7.931

Table Notes:

1. HZP
2. No Xe
3. HFP Eq Sm and Pu concentrations
4. ARO

NDR – S1C17

Table 6-32 HZP Integral Rod Worth as a Function of Steps Withdrawn and Burnup for Banks CD, CC, and CB in Overlap

Steps Withdrawn			HZP Integral Rod Worth, pcm				
Bank CB	Bank CC	Bank CD	Cycle Burnup, MWd/m!U				
			0	4000	10000	16000	21223
228	228	228	0	0	0	0	0
228	228	220	-16	-12	-13	-29	-39
228	228	212	-67	-53	-62	-130	-168
228	228	204	-129	-105	-133	-264	-323
228	228	196	-191	-163	-217	-411	-481
228	228	188	-254	-223	-307	-557	-632
228	228	130	-314	-282	-397	-691	-766
228	228	172	-371	-340	-482	-805	-877
228	228	164	-426	-397	-562	-899	-968
228	228	156	-479	-452	-636	-978	-1042
228	228	148	-532	-508	-705	-1042	-1102
228	228	140	-584	-583	-768	-1094	-1143
228	228	132	-636	-618	-827	-1133	-1132
228	228	124	-687	-673	-880	-1130	-1233
228	228	116	-737	-727	-928	-1175	-1213
228	228	108	-787	-779	-971	-1184	-1219
228	228	100	-833	-829	-1011	-1194	-1223
228	220	92	-917	-917	-1033	-1241	-1280
228	212	84	-989	-992	-1149	-1332	-1385
228	204	76	-1056	-1030	-1216	-1451	-1521
228	196	68	-1126	-1130	-1289	-1581	-1665
228	188	60	-1203	-1208	-1374	-1705	-1795
228	180	52	-1288	-1291	-1466	-1819	-1907
228	172	44	-1374	-1376	-1561	-1918	-2000
228	164	36	-1458	-1458	-1653	-2002	-2075
228	156	28	-1536	-1535	-1737	-2068	-2134
228	148	20	-1608	-1605	-1815	-2121	-2130
228	140	12	-1672	-1669	-1884	-2163	-2216
228	132	4	-1729	-1728	-1945	-2197	-2244
228	128	0	-1758	-1759	-1975	-2209	-2253
228	120	0	-1790	-1795	-2005	-2217	-2258
228	112	0	-1824	-1835	-2037	-2222	-2260
228	104	0	-1859	-1876	-2070	-2229	-2265

Table Notes:

1. No Xe
2. HFP Eq Sm and Pu concentrations
3. HZP, ARO, no Xe, critical boron concentration for the indicated burnup
4. Rod worths based on sequential insertion with overlap

NDR - S1C17

Table 6-33 HZP Integral Rod Worth with Peak Xenon as a Function of Steps Withdrawn and Burnup for Banks CD, CC, and CB in Overlap

Steps Withdrawn			HZP Integral Rod Worth, pcm				
Bank CB	Bank CC	Bank CD	Cycle Burnup, MWd/mtU				
			0	4000	10000	16000	21223
228	228	228	0	0	0	0	0
228	228	220	-23	-21	-21	-34	-46
228	228	212	-99	-90	-96	-150	-191
228	228	204	-187	-177	-199	-295	-359
228	228	196	-276	-268	-316	-443	-523
228	228	188	-361	-359	-437	-596	-674
228	228	180	-440	-444	-552	-728	-804
228	228	172	-511	-521	-655	-838	-911
228	228	164	-576	-591	-747	-928	-997
228	228	156	-635	-654	-829	-1001	-1064
228	228	148	-690	-712	-901	-1051	-1118
228	228	140	-741	-765	-964	-1108	-1158
228	228	132	-790	-814	-1016	-1143	-1187
228	228	124	-836	-860	-1058	-1166	-1204
228	228	116	-879	-902	-1089	-1178	-1211
228	228	108	-919	-941	-1114	-1185	-1215
228	228	100	-957	-977	-1136	-1194	-1223
228	220	92	-1028	-1045	-1187	-1242	-1277
228	212	84	-1034	-1111	-1257	-1339	-1388
228	204	76	-1160	-1179	-1341	-1465	-1531
228	196	68	-1231	-1252	-1435	-1593	-1679
228	188	60	-1310	-1334	-1533	-1722	-1808
228	180	52	-1396	-1422	-1631	-1829	-1913
228	172	44	-1483	-1512	-1724	-1921	-1998
228	164	36	-1559	-1600	-1809	-1995	-2065
228	156	28	-1649	-1682	-1884	-2053	-2116
228	148	20	-1722	-1756	-1949	-2099	-2155
228	140	12	-1787	-1821	-2004	-2135	-2186
228	132	4	-1841	-1877	-2051	-2164	-2210
228	124	0	-1866	-1902	-2070	-2174	-2218
228	120	0	-1890	-1927	-2086	-2179	-2220
228	112	0	-1915	-1953	-2101	-2182	-2219
228	104	0	-1941	-1979	-2116	-2187	-2223

Table Notes:

1. Peak Xe
2. HFP Eq Sm and Pu concentrations
3. HZP, ARO, no Xe, critical boron concentration for the indicated burnup
4. Rod worths based on sequential insertion with overlap

NCR - S1C17

Table 6-38 HZP Equilibrium and Peak Xenon Worth

Cycle Life MWd/mtU	HZP Xenon Worth. pcm	
	Equilibrium	Peak
0	-2678	-3963
150	-2682	-3965
500	-2555	-3991
1000	-2679	-3977
2000	-2675	-3974
4000	-2675	-3980
6000	-2695	-4024
8000	-2730	-4094
10000	-2779	-4222
12000	-2838	-4391
14000	-2833	-4527
16000	-2913	-4593
16201	-2915	-4598
18000	-2932	-4639
19180	-2943	-4680
21223	-3001	-4898

Table Notes:

1. ARO
2. HFP ARO critical boron concentration
3. HFP Eq Sm and Pu concentrations
4. "Equilibrium" is the Xe worth, at HZP conditions, with a Xe concentration resulting from HFP steady state operation (i.e., HFP Eq Xe concentration).
5. "Peak" is the worth of Xe at HZP conditions and a peak Xe concentration resulting from the decay at HZP conditions of the HFP Eq Xe concentration.

NDR – S1C17

Table 6-39 HZP Samarium and Plutonium Worth

Cycle Life MWd/mtU	HZP Samarium Worth, Peak – Equilibrium pcm	HZP Plutonium Worth, Peak - Equilibrium pcm
0	0	0
150	-237	150
500	-327	202
1000	-346	207
2000	-357	205
4000	-385	204
6000	-415	204
8000	-435	198
10000	-446	189
12000	-476	189
14000	-512	190
16000	-531	188
16201	-531	186
18000	-545	184
19180	-552	182
21223	-600	190

Table Notes:
 1. ARO
 2. HZP ARO critical boron concentration

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT

SURVEILLANCE INSTRUCTION

0-SI-NUC-000-001.0

ESTIMATED CRITICAL CONDITIONS

Revision 5

QUALITY RELATED

PREPARED BY: GREGORY KNIEDLER

RESPONSIBLE ORGANIZATION: REACTOR ENGINEERING

APPROVED BY: GARY BAIR

EFFECTIVE DATE: 11/9/2000

VERIFICATION DATE: N/A

LEVEL OF USE: **CONTINUOUS USE**

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 2 of 42
------------------------	--------------------------------------	---------------------------------------------

REVISION

DESCRIPTION

This revision is an intent change. Revised to allow the ECC to be terminated if ECC is off by $\pm > 750$ pcm (this is below the ± 1000 pcm Tech Spec Limit) and revised App. D to perform a Sm/Pu calculation like it is performed in App. B.

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 3 of 42
-----------------	--------------------------------------	---------------------------------------------

TABLE OF CONTENTS
Page 1 of 2

Section Title	Page
TABLE OF CONTENTS	3
1.0 INTRODUCTION	5
1.1 Purpose	5
1.2 Scope.....	5
1.2.1 Surveillance Tests to be Performed.....	5
1.2.2 Requirements Fulfilled.....	5
1.2.3 Modes	6
1.3 Frequency and Conditions.....	6
2.0 REFERENCES.....	6
2.1 Performance References	6
2.2 Developmental References.....	6
3.0 PRECAUTIONS AND LIMITATIONS.....	7
4.0 PREREQUISITE ACTIONS	8
5.0 ACCEPTANCE CRITERIA	9
6.0 PERFORMANCE	9
6.1 ECB and ECC Calculation	10
6.1.1 ECB, Unit Conditions	10
6.1.2 ECC, Unit Conditions.....	11

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 4 of 42
------------------------	--------------------------------------	---------------------------------------------

TABLE OF CONTENTS

Page 2 of 2

Section Title	Page
7.0 POST SURVEILLANCE ACTIVITIES	13
8.0 RECORDS	13
APPENDIXES	
APPENDIX A: ECC CALCULATION FOLLOWING A REFUELING OUTAGE	14
APPENDIX B: DATA SHEET 1 ECB CALCULATION USING NDR DATA	18
APPENDIX C: MONITORING THE APPROACH TO CRITICALITY.....	28
APPENDIX D: SAMARIUM/PLUTONIUM WORTH.....	38
APPENDIX E: ECC AND ECB BIAS CALCULATION.....	42

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 5 of 42
-----------------	--------------------------------------	---------------------------------------------

1.0 INTRODUCTION

1.1 Purpose

This Surveillance Instruction (SI) provides detailed steps for predicting the rod position and/or boron concentration at which core criticality will be achieved. It also provides guidance for monitoring the approach to criticality.

1.2 Scope

1.2.1 Surveillance Tests to be Performed

- A. Section 6.1.1 of this SI is used to Calculate the estimated critical boron concentration using data from Nuclear Design Report (NDR).
- B. Section 6.1.2 of this SI is used to calculate the estimated critical rod position using data from Nuclear Design Report (NDR).
- C. Appendix C of this SI is used to monitor the approach to criticality. This section will be used in conjunction with General Operating Instruction 0-GO-2, *Plant Startup from Hot Standby to Reactor Critical*. For initial startups following a refueling outage, Restart Test Instruction 0-RT-NUC-000-003.0, *Initial Criticality* will be used in lieu of Appendix C.
- D. Appendix A of this SI is used to calculate the acceptance limits for a startup following a refueling outage.

1.2.2 Requirements Fulfilled

- A. Performance of this SI completely fulfills Technical Specifications (TS) Surveillance Requirements (SR) 4.1.1.1.c.
- B. This SI may be used to completely fulfill TS SR 4.1.1.1.b during the transition from mode 3 to mode 2 and immediately following criticality by verifying the estimated and/or actual critical condition is above the zero power rod insertion limit.

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 6 of 42
-----------------	--------------------------------------	---------------------------------------------

1.2.3 Modes

Plant operating modes for which the surveillance requirement covered in this Instruction must be satisfied (applicable modes) and during which this Instruction can be performed (performance modes) are:

- A. Applicable Modes: 2, and 3 (with shutdown banks (S/D) withdrawn).
- B. Performance Modes: Sections 6.1.1, 6.2.1 - all
Appendix C - 2 and 3 (S/D banks withdrawn)
Appendix A - all

1.3 Frequency and Conditions

- A. This SI must be performed within 4 hours before achieving criticality.
- B. This SI may be used to verify adequate shutdown margin while withdrawing control banks by ensuring that the predicted critical control rod position is within the limits of TS LCO 3.1.3.6.

2.0 REFERENCES

2.1 Performance References

- A. 0-GO-2, Plant Startup from Hot Standby to Reactor Critical
- B. 0-RT-NUC-000-003.0, Initial Criticality
- C. TI-28, Curve Book
- D. TI-33, Xenon Worth Calculation
- E. TI-34, Samarium Worth Calculation
- F. TI-44, Boron Tables
- G. Nuclear Design Report for the applicable Unit/fuel cycle.
- H. SPP-10.4, Reactivity Management Program
- I. SPP-2.4, Records Management

2.2 Developmental References

- A. SQN Technical Specifications
- B. Framatome Technical Document 64-1245312-00.

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 7 of 42
----------	-------------------------------	---------------------------------------------

3.0 PRECAUTIONS AND LIMITATIONS

- A. For reactor startup, if criticality is not achieved within 4 hours of the performance of this SI, another ECC calculation must be performed or existing calculation reverified. If additional performance(s) are required, the applicable section(s) must be reperformed and attached to this SI. Reperformance is not required for the first ECC calculation following a refueling outage or when the core is xenon free and the boron concentration and RCS temperature are essentially unchanged. In these cases the ECC calculation must be verified every 4 hours. This procedure assumes a HZP temperature for criticality. If temperature varies greatly from HZP temperature then recover temperature before criticality.
- B. Adequate shutdown margin during plant operation in mode 3 is normally verified by performing 0-SI-NUC-000-038.0, *Shutdown Margin*. If control bank withdrawal is anticipated (i.e., an approach to criticality), an ECC calculation is used to verify adequate shutdown margin by ensuring that the predicted critical control rod position is within the limits of TS LCO 3.1.3.6. If this Instruction is being used to verify adequate shutdown margin, it must normally be performed at least once every 4 hours except for the first ECC calculation following a refueling outage or when the core is xenon free and the boron concentration and RCS temperature are essentially unchanged; in these situations the ECC calculation can be verified at least once every 4 hours rather than reperformed.
- C. When using a verified and validated computer program to calculate estimated critical rod positions or estimated critical boron concentrations, the calculative inputs should be verified to ensure the computer outputs are correct. This action may be accomplished by comparing the computer inputs with the data tables in the Nuclear Design Report to ensure they are within the range of the tables. The same method (e.g., computer calculations) should be used to determine both previous and current parameters (e.g., xenon worth) to ensure accurate estimates of reactivity changes.
- D. This Instruction references data from the Nuclear Design Report (NDR). Care must be exercised to ensure the data used is for the applicable unit and is applicable to the current fuel cycle. If the NDR for the current fuel cycle is not available, other applicable vendor data may be used provided that it is documented. Care must be exercised in all cases to preserve the signs and units when reading the figures and performing the calculations. Xenon, control rods, boron, and samarium are neutron poisons and the reactivity change due to adding a poison is always negative.
- E. When calculating the Estimated Critical Rod Position (ECP), the rod position must be above the HZP Rod Insertion Limits and below the Administrative Rod Withdrawal Limits (if applicable) for criticality.

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 8 of 42
----------	-------------------------------	---------------------------------------------

3.0 PRECAUTIONS AND LIMITATIONS (Continued)

- F. If the approach to criticality is suspended for an extended period of time near the point of criticality, the reactor core shall be made sufficiently subcritical to avoid an inadvertent criticality.
- G. In the event of an unexplained change in reactivity during an approach to criticality, the approach to criticality shall cease and the reactor core shall be made sufficiently subcritical to prevent an inadvertent criticality. Approval of the Plant Manager or his designee is required to resume the approach to criticality.
- H. Substantial Boron 10 depletion can occur after long continuous runs (i.e. 200 days or longer). Therefore, boron letdown curves should be evaluated for this phenomenon and adjustments made, particularly for fast restarts of the reactor when boron dilution would be necessary to account for xenon buildup.
- I. An ECC Upper Termination Limit (UTL) of +750 pcm and an ECC Lower Termination Limit (LTL) of -750 pcm are calculated. These values are within the Upper and Lower Allowable Limits, Tech Spec limits, of ± 1000 pcm. The UTL and LTL shall be used by Reactor Engineering and Operations to determine whether an approach to critical should be terminated and a new ECC calculated. The ECC must be terminated if the Upper and Lower Allowable Limits are approached.

4.0 PREREQUISITE ACTIONS

NOTE 1 During performance of this Instruction, any **IF/THEN** statement may be marked N/A when the corresponding stated condition does not occur.

NOTE 2 Shutdown Banks are withdrawn per 0-GO-2 before control rod pull to critical.

[1] **ENSURE** Instruction to be used is a copy of the effective version, and

RECORD necessary information on STS Cover Sheet. □

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 9 of 42
------------------------	--------------------------------------	---------------------------------------------

4.0 PREREQUISITE ACTIONS (Continued)

- [2] **ENSURE** NDR (or other applicable vendor data) for the current fuel cycle is available.

- [3] **ENSURE** sufficient copies of applicable sections and appendices are available.

5.0 ACCEPTANCE CRITERIA

- A. The actual critical rod position must be equal to or further withdrawn than the zero power rod insertion limit and less withdrawn than the rod position that ensures a negative moderator temperature coefficient.

- B. The absolute difference between the estimated critical conditions and the actual critical conditions must be less than 1000 pcm.

- C. If either criteria stated above is not satisfied, the Shift Manager (SM) shall be notified and the applicable action requirement of LCO 3.1.1.1, 3.1.1.3, or 3.1.3.6 must be satisfied.

- D. If the absolute difference between the estimated critical condition and the actual critical condition is 750 pcm or greater, the basis for termination of startup or for proceeding shall be recorded on Data Sheet C-2.

- E. If the absolute difference between the estimated critical condition and the actual critical condition is greater than 500 pcm, the Reactor Engineering section shall initiate an evaluation to understand the reason for the difference and an entry shall be made into the Reactivity Management database.

6.0 PERFORMANCE

NOTE 1 The Test Director will determine if Section 6.1.1 or 6.1.2 or Appendix A will be performed. Appendix A performs an ECC calculation for plant startup following a refueling outage. Only the section that is performed must be included in the data package.

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 10 of 42
------------------------	--------------------------------------	----------------------------------------------

6.1 ECB and ECC Calculation

NOTE 2 This section must be reperformed if criticality is not achieved within the four-hour time interval recorded in 6.1.1 [1] or 6.1.2 [1]. However, if the core is xenon free and the RCS boron concentration and RCS temperature are essentially unchanged, then the ECC calculation can be verified within the four-hour time interval recorded in step 6.1.1 [1] or 6.1.2 [1] rather than reperforming this section.

[1] **PERFORM** the computer program "REACTF" provided that the tables in the NDR (or other applicable vendor data) are used to verify the inputs to the program are within the range of the Tables, use Appendix E to determine the ECB/ECC bias, and

ATTACH the output to the STS Cover Sheet,

OR

CONTINUE performing by hand.

6.1.1 ECB, Unit Conditions

[1] **RECORD** on Appendix B, Data Sheet 1 the unit and applicable time period for which the ECB calculation is being performed (four hours or less).

[2] **RECORD** on Appendix B, Data Sheet 1, the following parameters.

- A. RCS present boron concentration.
- B. Core Average Burnup.
- C. Date and Time of shutdown.
- D. Average fraction of rated power for the past 4 days before shutdown.

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 11 of 42
----------	-------------------------------	----------------------------------------------

6.1.1 ECB, Unit Conditions (Continued)

- E. Record the change in samarium/plutonium worth from equilibrium to peak from the HZP Sm and Pu worth table in the NDR.(Table 6-39)
- F. ECB bias(Refer to Appendix E)
- G. Calculate samarium/plutonium worth in pcm from NDR. Examples in Appendix D.
- H. Xenon predicted worth.
- I. ARO, HZP, No Xe, HFP Eq Sm +Pu Concentration critical boron concentration from NDR at cycle burnup recorded in step 6.1.1[2]B.(Table 6-1)
- J. ARO, HZP differential boron worth from NDR at cycle burnup recorded in step 6.1.1[2]B and the boron concentration recorded in step 6.1.2[2]A(Table 6-8).
- K. Desired critical control bank position.
- L. Peak Xenon worth from NDR.(Table 6-38)
- M. Reactivity worth of inserted rods.(Table 6-32 and 6-33)
- N. Reactivity worth of combined reactivities.
- O. Estimated boron equivalent of reactivity worth.
- P. ECB calculated value.
- Q. Make-up water or boric acid change required.

6.1.2 ECC, Unit Conditions

[1] **RECORD** on Appendix B, Data Sheet 2, the unit and applicable time period for which the ECC calculation is being performed (four hours or less).



<p style="text-align: center;">SQN 0</p>	<p style="text-align: center;">ESTIMATED CRITICAL CONDITIONS</p>	<p>0-SI-NUC-000-001.0 Rev 5 Page 12 of 42</p>
-----------------------------------------------------	-------------------------------------------------------------------------	-------------------------------------------------------

6.1.2 ECC, Unit Conditions (Continued)

[2] RECORD on Appendix B, Data Sheet 2, the following parameters. □

- A. Present RCS boron concentration.
- B. Core Average Burnup.
- C. Date and Time of shutdown.
- D. Average fraction of rated power for the last 4 days before shutdown.
- E. Record the change in samarium/plutonium worth from equilibrium to peak from the HZP Sm and Pu worth table in the NDR.(Table 6-39)
- F. Calculate samarium/plutonium worth in pcm from NDR. Examples in Appendix D.
- G. Xenon predicted worth.
- H. ARO, HZP, No Xe, HFP Eq Sm and Pu, design critical boron concentration from NDR at cycle burnup recorded in step 6.1.2.[2]B.(Table 6-1)
- I. ARO, HZP differential boron worth from NDR at cycle burnup recorded in step 6.1.2[2]B and the boron concentration recorded in step 6.1.2[2]A.(Table 6-8)
- J. ECC bias.(Refer to Appendix E)
- K. Calculate boron difference worth.
- L. Calculate necessary inserted rod reactivity worth.
- M. Record peak HZP Xenon worth from NDR.(Table 6-38)
- N. Calculate rod position from NDR.(Tables 6-32 and 6-33)
- O. Determine allowable window.
- P. Determine limits on control rods.
- Q. Acceptance Criteria.

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 13 of 42
------------------------	--------------------------------------	----------------------------------------------

6.1.2 ECC Unit Conditions (Continued)

[3] **PERFORM** Appendix C during the approach to criticality.

_____/_____
Initials Date

7.0 POST SURVEILLANCE ACTIVITIES

[1] **COMPLETE** applicable sections of STS Cover Sheet.

_____/_____
Initials Date

8.0 RECORDS

All records shall be maintained per SPP-2.4.

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 14 of 42
----------	-------------------------------	----------------------------------------------

APPENDIX A
Page 1 of 4

ECC CALCULATION FOLLOWING A REFUELING OUTAGE

NOTE 1 The SQN ECC administrative limit of ± 500 pcm is addressed in 0-RT-NUC-000-008.0 by the review criteria of ± 50 ppm for the measured ARO critical boron concentration.

[1] **RECORD** expected date of criticality on Data Sheet A-1.

[2] **RECORD** on Data Sheet A-1 the estimated Critical Boron Concentration (C_B) and D bank position from 0-RT-NUC-000-003.0.

[3] **RECORD** on Data Sheet A-1 the inserted control bank D rod worth using Table 6-32 from the NDR, control bank D position in Step [2] and cycle burnup of zero.

[4] **CALCULATE** on Data Sheet A-1 the Upper Allowable Limit (UAL) and the Lower Allowable Limit (LAL) using the following equations:

UAL = Step [3] + 1000 pcm

LAL = Step [3] - 1000 pcm

NOTE2 If the UAL is positive then the UAL for control rod position is ARO.

[5] **RECORD** on Data Sheet A-1 the UAL and LAL control bank D position using Table 6-32 of NDR, cycle burnup of zero, and pcm from Step [4].

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 15 of 42
----------	-------------------------------	----------------------------------------------

APPENDIX A
Page 2 of 4

ECC CALCULATION FOLLOWING A REFUELING OUTAGE

NOTE3 The following steps may be signed off during the performance of 0-RT-NUC-000-003.0.

[6] **OBTAIN** approval, on Data Sheet A-1, of ECP from unit RO or SRO.

[7] **ENSURE** all data in this section has been verified accurate within four hours of achieving criticality.

NOTE4 These steps are performed once criticality has been achieved and Operations has stabilized reactor power.

[8] **RECORD** on Appendix A, Data Sheet A-1, the following conditions at time of criticality.

- A. Date and Time of criticality
- B. Core Average Temperature
- C. Control Bank position
- D. Boron concentration

[9] **CHECK** appropriate box on Appendix A, Data Sheet A-1, to indicate whether the following acceptance criteria is satisfied:

Acceptance Criteria Actual criticality control bank position step [8]C is within the ± 1000 pcm limits of step [5]?

[10] **IF** acceptance criteria in step [9] was not satisfied, **THEN IMMEDIATELY NOTIFY** the SM that the action requirement of LCO 3.1.1.1 must be satisfied.

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 18 of 42
----------	-------------------------------	----------------------------------------------

APPENDIX B

Page 1 of 10

DATA SHEET 1

ECB CALCULATION USING NDR DATA

6.1.1 ECB, Unit Conditions

[1] Unit _____ ECB applicable from _____ / _____ to _____ / _____
Date Time Date Time

[2] DETERMINE THE FOLLOWING INFORMATION

A. Present RCS Boron concentration _____ ppm

B. Core average burnup _____ MWD/MTU

C. Record Date and time of shutdown. _____ / _____
Date Time

D. Average fraction of rated power for the last 4 days before shutdown.

Power Fraction = _____

E. Record the change in samarium/ plutonium worth from equilibrium to peak from the HZP samarium/plutonium worth table (Table 6-39) in the NDR at the cycle burnup recorded in step 6.1.1[2]B.

SM Worth = _____ pcm PU Worth = _____ pcm

F. ECB bias adjustments between -100 ppm and +100 ppm may be necessary to reflect any nuclear design bias due to calculational methods or B-10 depletion. (Refer to Appendix E for bias calculation)

ECB bias = _____ ppm (if applicable)

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 19 of 42
----------	-------------------------------	----------------------------------------------

APPENDIX B
Page 2 of 10

DATA SHEET 1
ECB CALCULATION USING NDR DATA

6.1.1 ECB, Unit Conditions (Continued)

G. Calculate the change in samarium/plutonium worth from equilibrium using values recorded in step 6.1.1[2]E above and the equation below. There are examples of this calculation in Appendix D.

The change in samarium/plutonium worth = [Average fraction of rated power for the last 4 days, step 6.1.1[2]D] x [(change in samarium worth from equilibrium to peak, step 6.1.1[2]E) x (1 - e^{-λ PM149t}) + (change in plutonium worth from equilibrium to peak, step 6.1.1[2]E) x (1 - e^{-λ NP239t})]

Where: λPM149 = .01305 hours⁻¹

 λNP239 = .01229 hours⁻¹

t = time since shutdown in hours = _____

Change in Sm/Pu Worth =

$$[\text{_____}] * [(\text{_____}) * (1 - e^{-0.01305 * \text{_____}}) + (\text{_____}) * (1 - e^{-0.01229 * \text{_____}})]$$

Change in Sm/Pu Worth = _____ pcm

H. Calculate the Xenon worth at time t recorded in step 6.1.1[2]G using TI-33.

Xenon worth = _____ pcm

I. ARO, HZP, No Xe, HFP Eq Sm and Pu critical boron concentration from NDR at the cycle burnup recorded in step 6.1.1.[2]B.(Table 6-1)

_____ ppm

J. From NDR determine the ARO, HZP Differential boron worth at the cycle burnup recorded in step 6.1.1.[2]B and the boron concentration recorded in step 6.1.1[2]A.(Table 6-8)

_____ pcm/ppm

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 20 of 42
----------	-------------------------------	----------------------------------------------

APPENDIX B

Page 3 of 10

**DATA SHEET 1
ECB CALCULATION USING NDR DATA**

6.1.1 ECB, Unit Conditions (Continued)

NOTE1 The selected bank D position should normally be between 140 and 160 steps withdrawn to minimize RPI inaccuracy, depending on Xenon transient and other factors.

K. Select desired critical control bank position:
_____ steps withdrawn on Control bank D.

L. Record HZP peak Xe worth at the cycle burnup recorded in step 6.1.1[2]B from the NDR table of HZP equilibrium and peak Xe worth.(Table 6-38)
_____ pcm

M. Calculate reactivity worth of inserted rods by interpolating between the NDR HZP integral rod worth tables with peak and no Xe using the peak Xe worth recorded in step 6.1.1[2]L and the Xe worth recorded in step 6.1.1[2]H. Interpolate within the tables based on the control rod position recorded in step 6.1.1[2]K and the cycle burnup recorded in step 6.1.1[2]B.(Tables 6-32 and 6-33)

Inserted Control Bank Worth = _____ pcm

N. Calculate combined reactivity worth of Xenon (step 6.1.1[2]H), samarium/plutonium worth (step 6.1.1[2]G) and desired critical control bank position 6.1.1[2]M):

$$\begin{aligned}
 \text{Combined Reactivity Worth} &= \left[\begin{array}{c} \text{Xe Reactivity} \\ \text{Worth} \\ \text{Step 6.1.1[2]H} \end{array} \right] + \left[\begin{array}{c} \text{Samarium /} \\ \text{Plutonium Worth} \\ \text{Step 6.1.1[2]G} \end{array} \right] + \left[\begin{array}{c} \text{Inserted Control} \\ \text{Bank Worth} \\ \text{Step 6.1.1[2]M} \end{array} \right] \\
 &= \text{_____} + \text{_____} + \text{_____} \\
 &= \text{_____} \text{ pcm}
 \end{aligned}$$

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 21 of 42
----------	-------------------------------	----------------------------------------------

APPENDIX B
Page 4 of 10

DATA SHEET 1
ECB CALCULATION USING NDR DATA

6.1.1 ECB, Unit Conditions (Continued)

O. Calculate the estimated boron equivalent of the combined reactivity worth:

$$\begin{aligned} \text{Estimated Boron Equivalent} &= \frac{\text{Combined Reactivity Worth Step 6.1.1[2]N}}{\text{Differential Boron Worth Step 6.1.1[2]J}} = \underline{\hspace{2cm}} \\ &= \underline{\hspace{2cm}} \text{ ppm} \end{aligned}$$

P. Calculate estimated critical RCS boron concentration (ECB):

$$\text{ECB} = \left[\begin{array}{c} \text{Design ARO,HZP, Critical} \\ \text{Boron Concentration} \\ \text{STEP 6.1.1[2]I} \end{array} \right] + \left[\begin{array}{c} \text{ECB Bias} \\ \text{STEP 6.1.1.[2]F} \end{array} \right] - \left[\begin{array}{c} \text{Estimated} \\ \text{Boron Equivalent} \\ \text{STEP 6.1.1[2]O} \end{array} \right]$$

ECB = _____ ppm + _____ ppm - _____ ppm

ECB = _____ ppm

Q. Determine amount of makeup water or boric acid to change the RCS boron from the boron concentration of step 6.1.1[2]A to the ECB of Step 6.1.1[2]P using TI-44 or REACTF program.

Water/boric acid _____ gal
circle one

Calculated by _____ Date/Time _____

Verified by _____ Date/Time _____

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 23 of 42
----------	-------------------------------	----------------------------------------------

APPENDIX B
Page 6 of 10

DATA SHEET 2
ECC CALCULATION USING NDR DATA

6.1.2 ECC, Unit Conditions (Continued)

- F. Calculate the change in samarium/plutonium worth from equilibrium using values recorded in step 6.1.2[2]E above and the equation below. There are examples of this calculation in Appendix D.

The change in samarium/plutonium worth = [Average fraction of rated power for the last 4 days, step 6.1.2[2]D] x [(change in samarium worth from equilibrium to peak, step 6.1.2[2]E) x (1 - e^{-λ PM149t}) + (change in plutonium worth from equilibrium to peak, step 6.1.2[2]E) x (1 - e^{-λ NP239t})]

$\lambda_{PM149} = .01305 \text{ hours}^{-1}$

$\lambda_{NP239} = .01229 \text{ hours}^{-1}$

t = time since shutdown in hours = _____ hrs

[_____] * [(_____) * (1 - e^{-0.01305 * _____}) + (_____) * (1 - e^{-0.01229 * _____})]

Change in Sm/Pu Worth =

Change in Sm/Pu Worth = _____ pcm

- G. Calculate the Xenon worth at time t recorded in step 6.1.2[2]F using TI-33.

Xenon worth = _____ pcm

- H. From NDR determine the design critical boron concentration (ARO, HZP, No Xe, HFP Eq Sm and Pu) at the cycle burnup recorded in step 6.1.2[2]B. (Table 6-1)

_____ ppm

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 24 of 42
----------	-------------------------------	----------------------------------------------

APPENDIX B
Page 7 of 10

DATA SHEET 2
ECC CALCULATION USING NDR DATA

6.1.2 ECC, Unit Conditions (Continued)

- I. From NDR determine the ARO, HZP Differential boron worth at the cycle burnup recorded in step 6.1.2[2]B and the boron concentration recorded in step 6.1.2[2]A.(Table 6-8)

_____ pcm/ppm

- J. Based upon the unit cycle experience with previous ECC's or deviations of predicted RCS boron with burnup, adjustments between -100 ppm and +100 ppm may be necessary to reflect any nuclear design bias due to calculational methods or B-10 depletion. (Refer to Appendix E for bias calculation)

ECC bias = _____ ppm (if applicable)

NOTE1 If the results of Step 6.1.2[2]K are Negative, criticality can NOT be attained and the present RCS boron concentration needs to be reduced.

- K. Calculate reactivity worth of difference between Critical Boron Concentration (ARO, HZP, No Xe, HFP Eq Sm and Pu) and present RCS boron concentration.

$$\begin{aligned}
 \text{Boron Difference Reactivity Worth} &= \left[\begin{array}{l} \text{Present} \\ \text{Boron Concent.} - \left(\begin{array}{l} \text{Critical boron} \\ \text{Concentration} \\ \text{ARO, HZP, No Xe} \\ \text{Eq Sm (Step 6.1.2[2]H)} \end{array} \right) + \text{ECC Bias} \\ \text{STEP 6.1.2[2]A} \qquad \qquad \qquad \qquad \qquad \qquad \text{Step 6.1.2[2]J} \end{array} \right] \times \left[\begin{array}{l} \text{ARO, HZP} \\ \text{Differential} \\ \text{BoronWorth} \\ \text{Step6.1.2[2]I} \end{array} \right] \\
 &= \left[\frac{\text{_____}}{\text{Step 6.1.2[2]A}} \text{ ppm} - \left(\frac{\text{_____}}{\text{Step 6.1.2[2]H}} \text{ ppm} + \frac{\text{_____}}{\text{Step 6.1.2[2]J}} \text{ ppm} \right) \right] \times \left[\frac{\text{_____}}{\text{Step 6.1.2[2]I}} \text{ pcm / ppm} \right] \\
 &= \text{_____ pcm}
 \end{aligned}$$

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 25 of 42
----------	-------------------------------	----------------------------------------------

APPENDIX B
Page 8 of 10

DATA SHEET 2
ECC CALCULATION USING NDR DATA

6.1.2 ECC, Unit Conditions (Continued)

NOTE2 Results should be negative. If results are positive, boron concentration is still too high.

L. Calculate necessary inserted reactivity worth of control banks at the present RCS boron concentration.

$$\begin{aligned}
 \text{Inserted Control Bank Reactivity Worth} &= - \left(\left[\begin{array}{c} \text{Boron Diff.} \\ \text{Reactivity Worth} \\ \text{(Step 6.1.2[2]K)} \end{array} \right] + \left[\begin{array}{c} \text{Xenon} \\ \text{Worth} \\ \text{(Step 6.1.2[2]G)} \end{array} \right] + \left[\begin{array}{c} \text{Difference from} \\ \text{Sm / Pu Worth} \\ \text{(Step 6.1.2[2]F)} \end{array} \right] \right) \\
 &= - \left(\frac{\quad}{\text{Step 6.1.2[2]K}} \text{ pcm} + \frac{\quad}{\text{Step 6.1.2[2]G}} \text{ pcm} + \frac{\quad}{\text{Step 6.1.2[2]F}} \text{ pcm} \right) \\
 &= \underline{\hspace{2cm}} \text{ pcm}
 \end{aligned}$$

M. Record HZP peak Xe worth at the cycle burnup recorded in step 6.1.2[2]B from the NDR table of HZP equilibrium and peak Xe worth.(Table 6-38)

_____pcm

N. Calculate ECP rod position by interpolating between the NDR HZP integral rod worth tables with peak and no Xe using the peak Xe worth recorded in step 6.1.2[2]M and the Xe worth recorded in step 6.1.2[2]G. Interpolate within the tables based on the integral rod worth recorded in step 6.1.2[2]L and the cycle burnup recorded in step 6.1.2[2]B.(Tables 6-32 and 6-33)

steps=_____

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 26 of 42
----------	-------------------------------	----------------------------------------------

APPENDIX B

Page 9 of 10

**DATA SHEET 2
ECC CALCULATION USING NDR DATA**

6.1.2 ECC, Unit Conditions (Continued)

O. Calculate inserted control bank reactivity worth window for:

1. UPPER ALLOWABLE LIMIT (UAL), Upper Termination Limit (UTL), and Administrative Limit Upper (ALU) for criticality.

$$\begin{aligned}
 & \text{Inserted control} \\
 \text{UAL} &= \text{Bank Worth (Step 6.1.2[2] L)} + 1000 \text{ pcm} \\
 &= \underline{\hspace{2cm}} \text{ pcm} + 1000 \text{ pcm} \\
 &= \underline{\hspace{2cm}} \text{ pcm}
 \end{aligned}$$

$$\begin{aligned}
 & \text{Inserted control} \\
 \text{UTL} &= \text{Bank Worth (Step 6.1.2[2] L)} + 750 \text{ pcm} \\
 &= \underline{\hspace{2cm}} \text{ pcm} + 750 \text{ pcm} \\
 &= \underline{\hspace{2cm}} \text{ pcm}
 \end{aligned}$$

$$\begin{aligned}
 & \text{Inserted control} \\
 \text{ALU} &= \text{Bank Worth (Step 6.1.2[2] L)} + 500 \text{ pcm} \\
 &= \underline{\hspace{2cm}} \text{ pcm} + 500 \text{ pcm} \\
 &= \underline{\hspace{2cm}} \text{ pcm}
 \end{aligned}$$

2. LOWER ALLOWABLE LIMIT (LAL), Lower Termination Limit (LTL), and Administrative Limit Lower (ALL) for criticality.

$$\begin{aligned}
 & \text{Inserted control} \\
 \text{LAL} &= \text{Bank Worth (Step 6.1.2[2] L)} - 1000 \text{ pcm} \\
 &= \underline{\hspace{2cm}} \text{ pcm} - 1000 \text{ pcm} \\
 &= \underline{\hspace{2cm}} \text{ pcm}
 \end{aligned}$$

$$\begin{aligned}
 & \text{Inserted control} \\
 \text{LTL} &= \text{Bank Worth (Step 6.1.2[2] L)} - 750 \text{ pcm} \\
 &= \underline{\hspace{2cm}} \text{ pcm} - 750 \text{ pcm} \\
 &= \underline{\hspace{2cm}} \text{ pcm}
 \end{aligned}$$

$$\begin{aligned}
 & \text{Inserted control} \\
 \text{ALL} &= \text{Bank Worth (Step 6.1.2[2] L)} - 500 \text{ pcm} \\
 &= \underline{\hspace{2cm}} \text{ pcm} - 500 \text{ pcm} \\
 &= \underline{\hspace{2cm}} \text{ pcm}
 \end{aligned}$$

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 27 of 42
----------	-------------------------------	----------------------------------------------

APPENDIX B
Page 10 of 10

DATA SHEET 2
ECC CALCULATION USING NDR DATA

6.1.2 ECC, Unit Conditions (Continued)

P. Use inserted reactivity worth window of Step 6.1.2[2]O to determine the control bank position limits from NDR, interpolating as necessary.

UAU	=	_____	steps withdrawn on control bank		_____
UTL	=	_____	steps withdrawn on control bank		_____
ALU	=	_____	steps withdrawn on control bank		_____
LAL	=	_____	steps withdrawn on control bank		_____
LTL	=	_____	steps withdrawn on control bank		_____
ALL	=	_____	steps withdrawn on control bank		_____

Q. ACCEPTANCE CRITERIA

1. Estimated critical rod position step 6.1.2[2]N is further withdrawn than the zero power insertion limit of T.S. 3.1.3.6.

Yes/No _____ / _____

2. Estimated critical rod position step 6.1.2[2]N is further inserted than the negative MTC withdrawal limits of TI-28.

Yes/No/N/A _____ / _____

3. IF Acceptance Criteria in Step 1 or 2 is no, THEN
REPERFORM Section 6.1.2 with adjusted plant conditions.

_____ / _____

Calculated by/Verified and Approved by:

Calculated By _____ Date/Time _____

SRO or UO Verify/Approval _____ Date/Time _____

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 28 of 42
----------	-------------------------------	----------------------------------------------

APPENDIX C

Page 1 of 10

MONITORING THE APPROACH TO CRITICALITY

This Appendix is used in conjunction with General Operating Instruction 0-GO-2, "Plant Startup from Hot Standby to Reactor Critical." For initial startups following a refueling outage Restart Test Instruction 0-RT-NUC-000-003.0 "Initial Criticality" will be used instead of this Appendix.

[1] **PREPARE** the appropriate Data Sheet C-1, ICRRs during rod withdrawal, for each source range channel to be used for monitoring ICRR by **RECORDING**:

- A. Unit and Date.
 - B. Estimated critical boron concentration and rod position.
 - C. Draw upper/lower ± 750 pcm ECC termination band limits onto the ICRR plot. (Refer to Section 6.1.2)
-

APPENDIX C
Page 2 of 10

MONITORING THE APPROACH TO CRITICALITY

NOTE 1 The Main Control Room Instrumentation Gamma-Metrics shutdown monitor can be used in place of the scaler timer on control room panel M-13 to obtain count rates. If Gamma-Metrics shutdown monitors are used then N/A total counts and intervals in the following step. Do not use source range meters on M-4. ICS could be used if Gamma-Metrics monitor not available.

NOTE 2 The counting interval used in the following step should be long enough to obtain at least 1000 counts. (Not applicable if using Gamma-Metrics shutdown monitor)

[2] OBTAIN five counts for each source range channel to be used and average count rate to determine baseline count rate C_o .

	COUNT NO.	TOTAL COUNTS	INTERVAL (SEC)	COUNT RATE (CPS)
N31	1			
	2			
	3			
	4			
	5			
	N31- C_o			Average

	COUNT NO.	TOTAL COUNTS	INTERVAL (SEC)	COUNT RATE (CPS)
N32	1			
	2			
	3			
	4			
	5			
	N32- C_o			Average

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 30 of 42
----------	-------------------------------	----------------------------------------------

APPENDIX C

Page 3 of 10

MONITORING THE APPROACH TO CRITICALITY

[3] RECORD on Data Sheet C-1, the baseline count rate C_0 for each source range channel. □

- NOTE 1** The remainder of this section is performed concurrently with 0-GO-2.
- NOTE 2** Criticality should be anticipated whenever positive reactivity changes are made.
- NOTE 3** The counting interval used when determining ICRRs may be adjusted to attain approximately 1000 count (not applicable if using Gamma-Metrics shutdown monitor) or more but count rates must be normalized to the same counting interval (e.g., CPS).
- NOTE 4** If the approach to criticality is suspended for an extended period of time near the point of criticality, the reactor core shall be made sufficiently subcritical to reduce the possibility of an inadvertent criticality.
- NOTE 5** In the event of an unexplained change in reactivity during an approach to criticality, the approach to criticality shall cease and the reactor core shall be made sufficiently subcritical to prevent an inadvertent criticality. Approval of the Plant Manager or his designee is required to resume the approach to criticality.
- NOTE 6** Step **[4]** will be repeated many times. Each performance is initialed on Appendix C, Data Sheet C-1.
- NOTE 7** The SRO or Test Director may stop rod withdrawal and request an ICRR plot at any time but stopping points in below table are preferred.

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 31 of 42
----------	-------------------------------	----------------------------------------------

APPENDIX C

Page 4 of 10

MONITORING THE APPROACH TO CRITICALITY

CONTROL BANK	STEP POSTION	TOTAL STEPS
CONTROL BANK A/B	150/22	150
CONTROL BANK B/C	172/44	300
CONTROL BANK C/D	144/16	400
CONTROL BANK C/D	194/66	450
CONTROL BANK D	116	500
CONTROL BANK D	166	550
CONTROL BANK D	216(IF NECESSARY)	600
CONTROL BANK D	FULL OUT(IF NECESSARY)	≈615

[4] WHEN rods have been nominally withdrawn per table above or in accordance with 0-GO-2, **THEN**

CALCULATE and **PLOT** ICRR using the following steps:

[a] STOP rod withdrawal and allow count rate to reach equilibrium (approximately 2 minutes or longer for stabilization) before taking counts data.

[b] IF criticality has been achieved, **THEN** go to Step [4][j] to record critical data.

[c] RECORD number of counts, counting interval (secs) and calculated count rate on Data Sheet C-1.

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 32 of 42
------------------------	--------------------------------------	----------------------------------------------

APPENDIX C

Page 5 of 10

MONITORING THE APPROACH TO CRITICALITY

[d] **RECORD** Controlling Bank and Controlling Bank position on Data Sheet C-1.

[e] **RECORD** ICRR on Data Sheet 1 using the following equation:

$$ICRR_n = \frac{CR_o}{CR_n}$$

Where CR_o = Baseline count rage
CR_n = Individual count rate

NOTE 8 The ICRR plot constructed in the following step is intended to monitor the approach to criticality and is not necessarily a prediction of criticality. However, the general trend of the plot should predict criticality between the upper and lower ECC termination limits drawn in step [1].

[f] **PLOT** ICRR value recorded in step [e] versus Control Bank position from step [d] on Data Sheet C-1.

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 33 of 42
------------------------	--------------------------------------	----------------------------------------------

APPENDIX C

Page 6 of 10

MONITORING THE APPROACH TO CRITICALITY

- [g] EVALUATE** ICRR plot to determine overall trend.

- [h] IF** overall trend of ICRR plot indicates an acceptable ECC, **THEN**
CONTINUE Control Bank withdrawal in accordance with 0-GO-2.

- [i] IF** overall trend of ICRR plot indicates actual critical conditions will fall outside ECC termination band, **THEN**
NOTIFY appropriate Unit US, SM, Duty Plant Manager, and explain why startup should continue or ECC be recalibrated on Data Sheet C-2.

- [j] IF** reactor is critical, **THEN**
RECORD critical data on Data Sheet C-2.

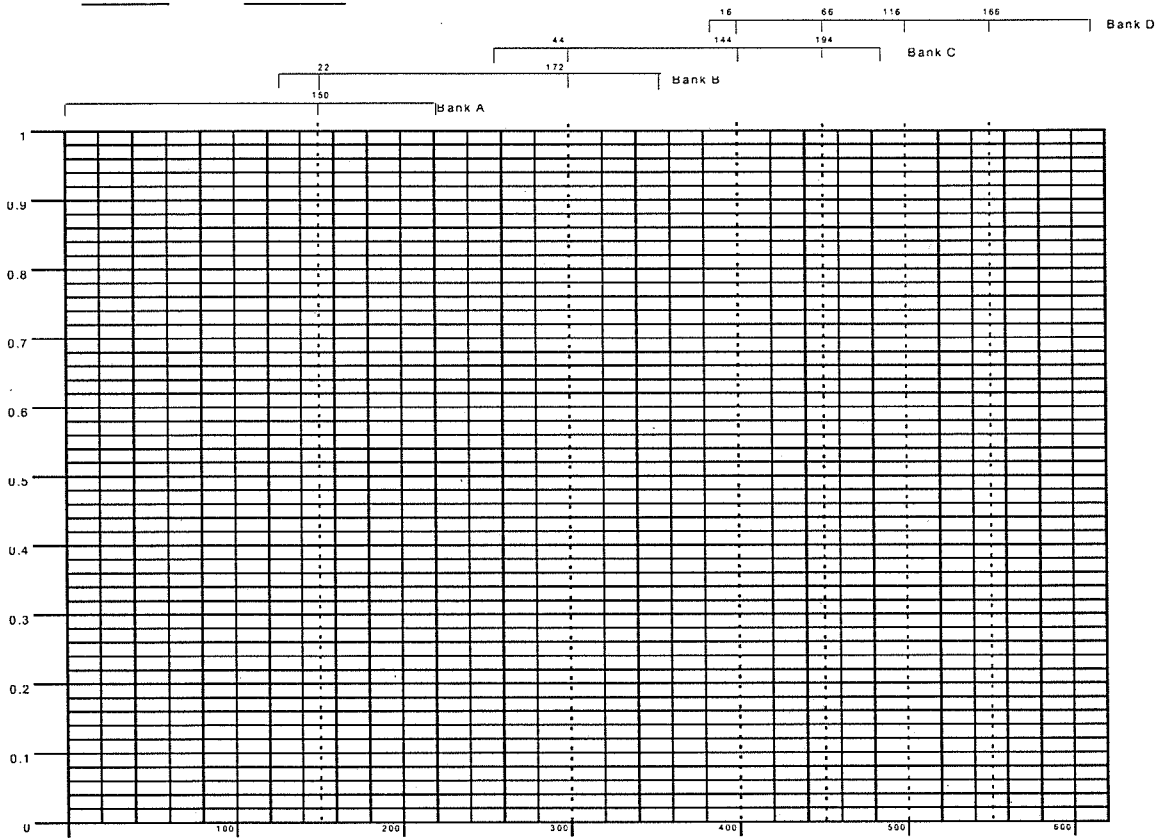
APPENDIX C
Page 7 of 10

MONITORING THE APPROACH TO CRITICALITY

DATA SHEET C-1

ICRR DURING ROD WITHDRAWAL FOR CHANNEL N-31

UNIT _____ CYCLE _____ DATE _____
 ECC: _____ PPM _____ STEPS _____



	COUNT	SECS	Cri	BANK	STEPS	ICRR	INITIALS
CR 0						1.00	
CR 1							
CR 2							
CR 3							
CR 4							
CR 5							
CR 6							
CR 7							
CR 8							
CR 9							

REVIEWED BY _____ DATE _____
 REACTOR ENGR

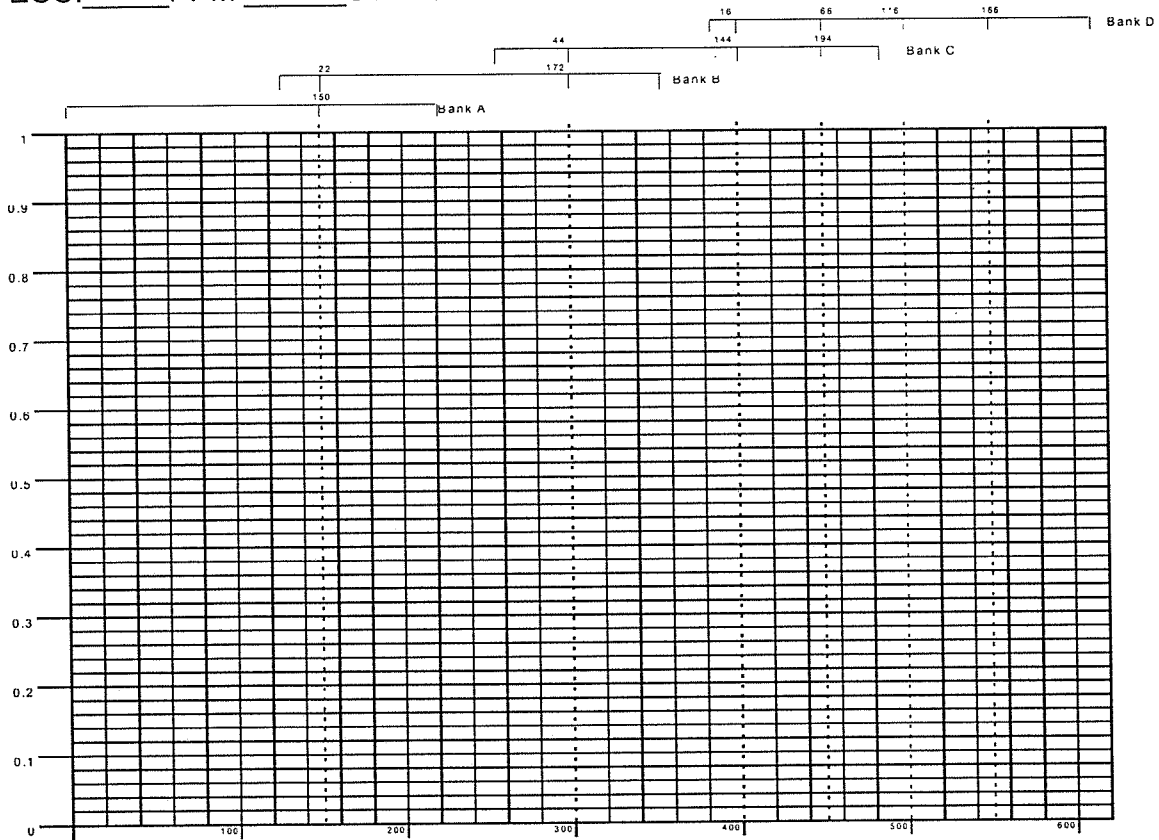
APPENDIX C
 Page 8 of 10

MONITORING THE APPROACH TO CRITICALITY

DATA SHEET C-1

ICRR DURING ROD WITHDRAWAL FOR CHANNEL N-32

UNIT _____ CYCLE _____ DATE _____
 ECC: _____ PPM _____ STEPS _____



	COUNT	SECS	Cri	BANK	STEPS	ICRR	INITIALS
CR 0						1.00	
CR 1							
CR 2							
CR 3							
CR 4							
CR 5							
CR 6							
CR 7							
CR 8							
CR 9							

REVIEWED BY _____ DATE _____
 REACTOR ENGR

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 36 of 42
----------	-------------------------------	----------------------------------------------

APPENDIX C
Page 9 of 10

MONITORING THE APPROACH TO CRITICALITY

DATA SHEET C-2
UNIT CONDITIONS AT CRITICALITY

- [1] Date/Time at criticality. _____
- [2] Core Average Temperature. _____ °F
- [3] Control Bank Position: Bank D at _____ steps
Bank C at _____ steps
- [4] Critical boron concentration. _____ ppm
- [5] ACCEPTANCE CRITERIA
- A. Actual critical rod position is further withdrawn than the ZPIL of TS 3.1.3.6. Yes No
- B. Actual critical rod position is further inserted than the negative MTC withdrawal limits of TI-28. Yes No NA
- C. Actual critical rod position is within the 1000 pcm limits. Yes No
- D. Date/Time of criticality are within the applicable 4 hour time span. Yes No
- E. Actual critical rod position is within the 750 pcm limits. Yes No
- NOTE 9** The following acceptance criteria is an administrative limit:
- F. Actual critical rod position is within the 500 pcm limits. Yes No
- [6] ACCEPTANCE CRITERIA VERIFICATION
- A. If acceptance criteria 5A was not satisfied, notify the SM that the action requirement of LCO 3.1.3.6 must be satisfied. Yes No NA

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 37 of 42
-----------------	--------------------------------------	----------------------------------------------

APPENDIX C
Page 10 of 10

MONITORING THE APPROACH TO CRITICALITY

DATA SHEET C-2
UNIT CONDITIONS AT CRITICALITY

- B. If acceptance criteria 5B was not satisfied, notify the SM that the action requirement of LCO 3.1.1.3 must be satisfied. Yes No NA

- C. If acceptance criteria 5C was not satisfied, notify the SM that the action requirement of LCO 3.1.1.1 must be satisfied. Yes No NA

- D. If acceptance criteria 5D was not satisfied, notify the SM that the action requirement of LCO 3.1.1.1 must be satisfied (see TS 4.0.3). Yes No NA

- E. If acceptance criteria 5E was not satisfied, request Reactor Engineering to perform an evaluation of the reactivity differences. Yes No NA

- F. If acceptance criteria 5F was not satisfied, request Reactor Engineering to perform an evaluation of the reactivity differences. Yes No NA

Comments: _____

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 38 of 42
----------	-------------------------------	----------------------------------------------

APPENDIX D

Page 1 of 4

SAMARIUM/PLUTONIUM WORTH

NOTE 1 The information below represent the minimum necessary to properly account for post-shutdown samarium and plutonium worth. As used herein, the terms Sm worth (or Δ Sm worth), Pu worth (or Δ Pu worth), and Sm/Pu worth (or Δ Sm/Pu worth) refer to the incremental worth due to the change in isotopic number density (i.e., "peaking" from the reference. The reference number density is defined as that resulting from depletion at HFP conditions (i.e., "equilibrium").

FCF recommends a combined accounting for the incremental worth post-shutdown due to number density peaking of these two isotopes (Sm^{149} and Pu^{239}) due to the similarity in the half lives of their respective precursors. Sm^{149} results from beta decay of promethium 149 (Pm^{149}) with half life of 53.1 hours. Pu^{239} results from beta decay of neptunium 239 (Np^{239}) with a half life of 2.35 days (56.4 hours).

FCF has presented the information necessary to make an adjustment for Sm/Pu worth after shutdown for an ECB/ECC calculation for a number of utilities. An example of the presentation of this data is shown below:

NOTE 2 For actual calculations, use the NDR--not the table below.

TABLE 1 - SAMARIUM AND PLUTONIUM-239 WORTHS AT HOT ZERO POWER, CONTROL RODS FULLY WITHDRAWN, NO XENON

Cycle Burnup, MWD/MTU	HZP Samarium Worth, Peak - Equilibrium, pcm	HZP Pu-239 Worth, Peak - Equilibrium, pcm
0	0	0
154	-215	137
963	-334	195
1925	-344	192
7700	-385	180
18249	-489	177

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 39 of 42
----------	-------------------------------	----------------------------------------------

APPENDIX D

Page 2 of 4

SAMARIUM/PLUTONIUM WORTH

NOTE 3 Not accounting for Sm/Pu worth at BOC has less impact on the accuracy of a given ECP than at EOC. The incremental Sm/Pu worth (see first NOTE in this Section) contribution of interest results from the sum of the values in each column for a given burnup. Two other factors in using a table like Table 1 are provided below.

Using the 53.1 hour (2.213 day) half life for the Sm¹⁴⁹ precursor, the following relationship applies to the percentage of the Sm/Pu worth correction that should be used assuming the pre-shutdown condition was nominal HFP operation for at least four days:

$$\text{Sm/Pu Fraction} = 1 - e^{-\lambda t}$$

where λ = the decay constant of the Sm¹⁴⁹ precursor, in this case
= 0.693/2.213 = 0.313 days⁻¹, and
t = time after shutdown, days

Table 2 results directly from this equation:

TABLE 2 - FRACTION OF Sm/Pu WORTH Vs TIME SHUTDOWN

Time After Shutdown (Days)	Fraction of Sm/Pu Worth
2	0.465
4	0.714
6	0.847
8	0.918
10	0.956
>12	1.000

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 40 of 42
----------	-------------------------------	----------------------------------------------

APPENDIX D

Page 3 of 4

SAMARIUM/PLUTONIUM WORTH

NOTE 4 Using the precursor half life for Sm¹⁴⁹ simplifies the calculations required if employing this method. It is acceptable to more precisely determine the Sm/Pu worth by separately treating Sm and Pu and adding together only after the separate decay factors are determined.

NOTE 5 If power level was not at 100% FP prior to shutdown, an additional reduction factor should be applied to account for the fact the HFP equilibrium values were not present prior to shutdown. An adequate approximation for this correction is to average the power level over four days prior to shutdown and use this "fraction or power level" as an additional factor applied to the worth in Table 1.

EXAMPLE #1

A reactor unit has been shutdown for five days, 13 hours. Criticality is scheduled 24 hours from now. Prior to the shutdown, the unit operated at 100% FP, steady state for over a month and achieved a core burnup of 14130 MWD/MTU. Using Table 1, what is the Sm/Pu reactivity contribution for tomorrow's startup?

The Δ Sm Worth and Δ Pu Worth for 14130 MWD/MTU from Table 1 by linear interpolation is:

$$\Delta\text{Sm Worth} = -448 \text{ pcm}$$

$$\Delta\text{Pu Worth} = +178 \text{ pcm}$$

The time of shutdown is projected to be six days, 13 hours, or 157 hours. Use the following Equation to determine the Sm/Pu Worth:

$$\text{Sm/Pu Worth} = [\text{Average fraction of rated power for the last 4 days}] \times [(\Delta\text{Sm Worth}) \times (1 - e^{-\lambda_{\text{PM149}} t}) + (\Delta\text{Pu Worth}) \times (1 - e^{-\lambda_{\text{P239}} t})]$$

Where: $\lambda_{\text{PM149}} = .01305 \text{ hours}^{-1}$

$\lambda_{\text{NP239}} = .01229 \text{ hours}^{-1}$

t = time since shutdown in hours = 157 hours

Therefore, Sm/Pu worth is:

$$\text{Sm/Pu Worth} = [1.00] \times [(-448) \times (1 - e^{-(.01305)(157)}) + (178) \times (1 - e^{-(.01229)(157)})] \text{ pcm}$$

$$\text{Sm/Pu Worth} = [1.00] \times [(-448) \times (0.8711) + (178) \times (0.8548)] \text{ pcm}$$

$$\text{Sm/Pu Worth} = -238 \text{ pcm}$$

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 41 of 42
----------	-------------------------------	----------------------------------------------

**APPENDIX D
PAGE 4 OF 4
SAMARIUM/PLUTONIUM WORTH**

EXAMPLE #2

A reactor unit has been shutdown for two days, 19 hours. Criticality is scheduled 24 hours from now. Prior to the shutdown, the unit operated at the following daily average power levels: 38% FP, 56% FP, 93% FP, and 100% FP. The core burnup is 16794 MWD/MTU. Using Table 1, what is the Sm/Pu reactivity contribution for tomorrow's startup?

The Δ Sm Worth and Δ Pu Worth for 16794 MWD/MTU from Table 1 by linear interpolation is:

$$\Delta\text{Sm Worth} = -475 \text{ pcm}$$

$$\Delta\text{Pu Worth} = +177 \text{ pcm}$$

The time of shutdown is projected to be three days, 19 hours, or 91 hours. Use the following Equation to determine the Sm/Pu Worth:

$$\text{Sm/Pu Worth} = [\text{Average fraction of rated power for the last 4 days}] \times [(\Delta\text{Sm Worth}) \times (1 - e^{-\lambda_{\text{PM149}} t}) + (\Delta\text{Pu Worth}) \times (1 - e^{-\lambda_{\text{P239}} t})]$$

Where: $\lambda_{\text{PM149}} = .01305 \text{ hours}^{-1}$

$\lambda_{\text{NP239}} = .01229 \text{ hours}^{-1}$

$t = \text{time since shutdown in hours} = 91 \text{ hours}$

Therefore, Sm/Pu worth is:

$$\text{Sm/Pu Worth} = [(.38 + .56 + .93 + 1)/4] \times [(-475) \times (1 - e^{-(.01305)(91)}) + (177) \times (1 - e^{-(.01229)(91)})] \text{ pcm}$$

$$\text{Sm/Pu Worth} = [.7175] \times [(-475) \times (0.69503) + (177) \times (0.67319)] \text{ pcm}$$

$$\text{Sm/Pu Worth} = -151 \text{ pcm}$$

SQN 0	ESTIMATED CRITICAL CONDITIONS	0-SI-NUC-000-001.0 Rev 5 Page 42 of 42
----------	-------------------------------	----------------------------------------------

APPENDIX E
Page 1 of 1
ECC AND ECB BIAS CALCULATION

NOTE: Engineering judgment (previous ECC history, time in core life, etc) should be used before ECC or ECB bias is applied. All or part of the bias can be used at test director discretion. Sign convention for Sequoyah is positive for measured RCS boron above the letdown curve and negative if below the letdown curve.

1. Run REACTF ECB for time of criticality with no bias.

ECB= _____

2. In FOLnnn.XLS, under the "Correct Factors" worksheet, Boron Addition Correction Factors Table, copy row and enter new column data with the date, cycle burnup, B-10 atom fraction, initial boron and final boron (Column O, P, Q, T and U); Reference B-10 Depletion Correction Factors Table, copy row and enter new column with cycle burnup (column AA) to calculate B-10 atom fraction in column AI.

Get B-10 Atom fraction (column AI at the correct burnup)= _____ Atom fraction

3. Calculate B-10 depletion.

B-10 depletion = $(1 - \frac{\text{Step 2}}{0.198}) * (\text{Step 1 ppm})$

B-10 depletion = _____ ppm

4. Obtain current design difference (AROCBC) from FOLnnn.XLS spreadsheet, "Data" worksheet, Delta AROCBC (design difference) column K for the correct burnup.

Design difference = _____ ppm

Note : Reverse the sign of design difference ppm by multiplying by -1
(This is applicable for both REACTF and a hand calculation).

5. Calculate Bias.

Bias = $\frac{\text{Step 3}}{\text{Step 3}} \text{PPM} + \frac{\text{Step 4 after sign change}}{\text{Step 4 after sign change}} \text{PPM}$

Bias = _____ PPM

6. Check calculated bias for general agreement with supplementary calculation in FOLnnn.XLS worksheet "Correction Factors" worksheet column AM after inputting ECB value into column AK.

7. Run the ECB with the bias, specifying control bank D height, to determine the ECB. Then run the ECC with the bias and ECB to determine control bank D height.

SEQUOYAH NUCLEAR PLANT
September 2010 NRC Exam

SRO A.2

**Review and Approve a
Disabled Alarm Checklist**

SRO Job Performance Measure

Task: Review and Approve a Disabled Alarm Checklist, per 0-SO-55-1 and OPDP-4.
Task Number: SRO 0001230301

Task Standard: Applicant identifies that the OPDP-4-1 Disabled Alarm Checklist is NOT filled out correctly for Annunciator 1-M4-B Window D1. Applicant DOES NOT sign package at Step 7 as approved.

Time Critical Task: YES: _____ NO: X

K/A Rating(s): 2.2.43 (3.0/3.3)

Method of Testing: _____

Simulated Performance: _____ Actual Performance: X

Evaluation Method: _____

Simulator _____ In-Plant _____ Classroom X

Main Control Room _____ Mock-up _____

Performer: _____
Trainee Name

Evaluator: _____ / _____
Name / Signature DATE

Performance Rating: SAT: _____ UNSAT: _____

Validation Time: 15 minutes Total Time: _____

Performance Time: Start Time: _____ Finish Time: _____

COMMENTS

Tools/Equipment/Procedures Needed:

OPDP- 4 (Entire Procedure)
OPDP- 4-1 (Master Copy Completed) Attachment 1, Disabled Alarm Checklist.
1-AR-M4-B, Annunciator Response, Window D1

REFERENCES:

	Reference	Title	Rev No.
	OPDP- 4	Annunciator Disablement	0004

READ TO APPLICANT

DIRECTION TO APPLICANT:

I will explain the initial conditions, and state the administrative task to be performed. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

1. Unit-1 is in Mode 3 with a normal plant shutdown in progress per 0-GO-7 "Unit Shutdown from Hot Standby to Cold Shutdown."
2. The Operator-at-the-Controls (OATC) has informed you that Annunciator 1-M-4B (D1), XIS-68-387 REAC LEVEL CH-1 HYDRO ISOLATOR TROUBLE, has been received and that an operator has been sent to the local panel and reported that no local indication supports the alarm.
3. This alarm has been designated as an invalid alarm and the OATC has initiated OPDP-4-1 "Disabled Alarm Checklist" and Service Request 199401.
4. You are the Unit 1 SRO.

INITIATING CUES:

Review the OPDP-4-1 Disabled Alarm Checklist for approval. Note any and all discrepancies found during your review.

Job Performance Checklist:

STEP/STANDARD

SAT/UNSAT

<p><u>STEP 1:</u> Obtain a copy of the completed instruction.</p> <p><u>STANDARD:</u> A copy of OPDP-4-1 Attachment 1, Disabled Alarm Checklist.</p> <p><u>CUE:</u> Provide the Applicant a copy of the completed instruction.</p> <p><u>COMMENTS:</u></p>	<p>___ SAT</p> <p>___ UNSAT</p> <p>Start time _____</p>
<p><u>STEP 2:</u> DISABLED ALARM CHECKLIST block reviewed.</p> <p><u>STANDARD:</u> Applicant determines that the correct Panel Number is entered in the block.</p> <p><u>COMMENTS:</u></p>	<p>___ SAT</p> <p>___ UNSAT</p>
<p><u>STEP 3:</u> ALARM LOCATION block is reviewed.</p> <p><u>STANDARD:</u> Applicant determines correct information is entered in the Node/Mux/Pt or SER.</p> <p><u>COMMENTS:</u></p>	<p>___ SAT</p> <p>___ UNSAT</p>
<p><u>STEP 4:</u> Reviews entry in 1. Description of the alarm that is being defeated.</p> <p><u>STANDARD:</u> Applicant determines that an adequate description of the alarm function has been entered.</p> <p><u>COMMENTS:</u></p>	<p>___ SAT</p> <p>___ UNSAT</p>
<p><u>NOTE to evaluator:</u> While not a critical step a comment should be made if the candidate fails to identify the missing Service Request / Work Order.</p>	



Job Performance Checklist:

STEP/STANDARD

SAT/UNSAT

<p><u>STEP 5:</u> Reviews entry in 2. Describe reason for disabling the alarm (include procedure or WO number, if applicable).</p> <p><u>STANDARD:</u> Error Discovery – Inadequate / incomplete reason for disabling alarm.</p> <p> The data entered should include the Service Request number.</p> <p><u>COMMENTS:</u></p>	<p>___ SAT</p> <p>___ UNSAT</p>
<p><u>STEP 6:</u> Reviews entry in 3. Description of how alarm will be disabled.</p> <p><u>STANDARD:</u> Applicant determines that the description is adequate.</p> <p><u>COMMENTS:</u></p>	<p>___ SAT</p> <p>___ UNSAT</p>
<p>NOTE: The NOTE preceding step B of Appendix A states that initiation of a W.O. does not constitute in-process maintenance.</p>	
<p><u>STEP 7:</u> Reviews entry in 4. Is a 10CFR50.59 Review required prior to disabling alarm? (Refer to Appendix A).</p> <p><u>STANDARD:</u> Error Discovery - Incorrect block checked.</p> <p> Applicant determines that a 10CFR50.59 review <u>is required</u> since the alarm is NOT being disabled to support in-process maintenance.</p> <p>This step is critical because this disablement cannot take credit for maintenance that has not yet began. Therefore, a 10CFR50.59 Review is required prior to disabling the alarm.</p> <p><u>COMMENTS:</u></p>	<p>Critical Step</p> <p>___ SAT</p> <p>___ UNSAT</p>

Job Performance Checklist:

STEP/STANDARD

SAT/UNSAT

<p><u>STEP 8:</u> Reviews entry in 5. Is a 10CFR50.59 Review required prior to exceeding 90 days.(alarm disabled for maintenance)?</p> <p><u>STANDARD:</u> Applicant determines that a 10CFR50.59 review would be required prior to exceeding 90 days for a maintenance evolution.</p> <p><u>COMMENTS:</u></p>	<p>___ SAT</p> <p>___ UNSAT</p>
<p><u>STEP 9:</u> Reviews entry in 6. Is a Technical Evaluation (Form OPDP-4-5) required prior to disabling the alarm? (Refer to Appendix A).</p> <p><u>STANDARD:</u> Error Discovery - Incorrect block checked.</p> <p>Applicant determines that a Technical Evaluation (Form OPDP-4-5) is required since no maintenance is in progress and the affected equipment remain in service.</p> <p>This step is critical because the system remains in service with it's single SER point inoperable and no in-progress maintenance. Therefore, a Technical Evaluation is required prior to disabling the alarm.</p> <p><u>COMMENTS:</u></p>	<p>Critical Step</p> <p>___ SAT</p> <p>___ UNSAT</p>
<p><u>STEP 8:</u> The candidate concludes that approval block 7 cannot be approved as presented.</p> <p><u>STANDARD:</u> Applicant states that the package cannot be signed as approved until after the errors have been corrected.</p> <p>This step is critical because the alarm cannot be legally disabled without first having a 10CFR50.59 Review and Technical Evaluation completed.</p> <p><u>COMMENTS:</u></p> <p>JE: This concludes the JPM.</p>	<p>Critical Step</p> <p>___ SAT</p> <p>___ UNSAT</p> <p>Stop time _____</p>

SPO A2 - Key

Attachment 1
(Page 1 of 2)

OPDP-4-1 - Disabled Alarm Checklist

DISABLED ALARM CHECKLIST	
Site <u>SQN</u> Unit <u>1</u> <u>1-M-4B</u> Panel Number	ALARM LOCATION <u>D1</u> Window Number <u>SER 322</u> Node/Mux/Pt or SER/Sensor

Describe the function of this alarm (e.g. provide indication of abnormal operation, equipment failure, indication of an automatic trip, indication of loss of function):
The alarm window D1 on panel 1-M-4B, "XIS-68-387 Reac Level CH1 Hydro Isolator Trouble," has been received and is staying in continuously with no local indication to support the alarm.

Describe reason for disabling alarm/input: (Include procedure or WO number, if applicable)
This alarm is invalid & supplying the operator no meaningful information. Need to maintain "Dark Board" concept.

Describe how this alarm/alarm input will be disabled:
SER point 322 will be removed from scan.

- Is a 10CFR50.59 Review required prior to disabling alarm? (Refer to Appendix A) Yes No
- Is a 10CFR50.59 Review required prior to exceeding 90 days (alarm disabled for maintenance)? Yes No
- Is a Technical Evaluation (Form OPDP-4-5) required prior to disabling alarm? (Refer to Appendix A). Yes No

Prepared By John Doe UO Signature John Doe UO Print Name Today Date

7. Approval for annunciator disablement.	Yes	No	N/A
If required, is 10CFR50.59 Review attached?	<input type="checkbox"/>	<input type="checkbox"/>	<input type="checkbox"/>
If required, is Technical Evaluation (Form OPDP-4-5) attached?	<input type="checkbox"/>	<input type="checkbox"/>	<input type="checkbox"/>
If a Technical Evaluation was performed, is Compensatory Monitoring required and acceptable?	<input type="checkbox"/>	<input type="checkbox"/>	<input type="checkbox"/>
Are steps to enable the alarm provided in the controlling work document?	<input type="checkbox"/>	<input type="checkbox"/>	<input type="checkbox"/>

~~XXX~~ _____ Signature (SM/US) _____ Print Name _____ Date

This alarm must be returned to service by: _____ Date _____ Time N/A if not an LCO

No procedure or WO #

50.59 is required

Tech Eval is Required

Cannot approve

**Attachment 1
(Page 2 of 2)**

OPDP-4-1 - Disabled Alarm Checklist

DISABLED ALARM CHECKLIST				
DISABLED ALARM CHECKLIST		ALARM LOCATION		
Site _____	Unit _____	_____		
_____		Window Number		
Panel Number		Node/Mux/Pt or SER/Sensor		
<p>8. This alarm has been disabled as described in Item 1 of this form and Disabled Alarm Indicators have been placed on affected alarm window(s).</p>				
Performed by:				
	_____	_____	_____	_____
	Signature	Print Name	Date	Time
Verified By:				
	_____	_____	_____	_____
	Signature	Print Name	Date	Time
<p>9. Describe actions necessary to restore annunciator to normal including post-restoration testing.</p> <p>_____</p> <p>_____</p> <p>_____</p>				
Prepared by:				
	_____	_____	_____	_____
	Signature	Print Name	Date	Time
Reviewed & Approved by:				
	_____	_____	_____	_____
	SM/US Signature	Print Name	Date	Time
<p>10. This alarm has been restored to normal and tested in accordance with Item 8 of this form and Disabled Alarm Indicator(s) associated with this alarm have been removed.</p>				
Performed by:				
	_____	_____	_____	_____
	Signature	Print Name	Date	Time
Verified By:				
	_____	_____	_____	_____
	Signature	Print Name	Date	Time
<p>11. Compensatory Monitoring of this alarm is terminated and Unit Supervisor notified. N/A if no Compensatory Monitoring required.</p>				

	Signature	Print Name	Date	Time

DIRECTION TO APPLICANT:

I will explain the initial conditions, and state the administrative task to be performed. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

1. Unit-1 is in Mode 3 with a normal plant shutdown in progress per 0-GO-7 "Unit Shutdown from Hot Standby to Cold Shutdown."
2. The Operator-at-the-Controls (OATC) has informed you that Annunciator 1-M-4B (D1), XIS-68-387 REAC LEVEL CH-1 HYDRO ISOLATOR TROUBLE, has been received and that an operator has been sent to the local panel and reported that no local indication supports the alarm.
3. This alarm has been designated as an invalid alarm and the OATC has initiated OPDP-4-1 "Disabled Alarm Checklist" and Service Request 199401.
4. You are the Unit 1 SRO.

INITIATING CUES:

Review the OPDP-4-1 Disabled Alarm Checklist for approval. Note any and all discrepancies found during your review.

Attachment 1
(Page 1 of 2)

OPDP-4-1 - Disabled Alarm Checklist

DISABLED ALARM CHECKLIST			
<p style="text-align: center;">DISABLED ALARM CHECKLIST</p> <p>Site <u>SQN</u> Unit <u>1</u></p> <p style="text-align: center;"><u>1-M-4B</u></p> <p style="text-align: center;">Panel Number</p>	<p style="text-align: center;">ALARM LOCATION</p> <p style="text-align: center;"><u>D1</u></p> <p style="text-align: center;">Window Number</p> <p style="text-align: center;"><u>SER 322</u></p> <p style="text-align: center;">Node/Mux/Pt or SER/Sensor</p>		
<p>① Describe the function of this alarm (e.g. provide indication of abnormal operation, equipment failure indication of an automatic trip, indication of loss of function):</p> <p><u>The alarm window D1 on panel 1-M-4B, 'XIS-68-387 Reac Level CH1 Hydro Isolator Trouble', has been received and is staying in continuously with no local indication to support the alarm.</u></p>			
<p>② Describe reason for disabling alarm/input: (Include procedure or WO number, if applicable)</p> <p><u>This alarm is invalid & supplying the operator no meaningful information. Need to maintain "Dark Board Concept"</u></p>			
<p>③ Describe how this alarm/alarm input will be disabled:</p> <p><u>SER point 322 will be removed from scan.</u></p>			
<p>④ Is a 10CFR50.59 Review required prior to disabling alarm? (Refer to Appendix A)</p>	<p>Yes <input type="checkbox"/></p>	<p>No <input checked="" type="checkbox"/></p>	
<p>⑤ Is a 10CFR50.59 Review required prior to exceeding 90 days (alarm disabled for maintenance)?</p>	<p><input checked="" type="checkbox"/></p>	<p><input type="checkbox"/></p>	
<p>⑥ Is a Technical Evaluation (Form OPDP-4-5) required prior to disabling alarm? (Refer to Appendix A).</p>	<p><input type="checkbox"/></p>	<p><input checked="" type="checkbox"/></p>	
<p>Prepared By <u>John Doe UO</u></p> <p style="text-align: center;">Signature</p>	<p><u>John Doe UO</u></p> <p style="text-align: center;">Print Name</p>	<p><u>Today</u></p> <p style="text-align: center;">Date</p>	
<p>7. Approval for annunciator disablement.</p>			
<p>If required, is 10CFR50.59 Review attached?</p>	<p>Yes <input type="checkbox"/></p>	<p>No <input type="checkbox"/></p>	<p>N/A <input type="checkbox"/></p>
<p>If required, is Technical Evaluation (Form OPDP-4-5) attached?</p>	<p><input type="checkbox"/></p>	<p><input type="checkbox"/></p>	<p><input type="checkbox"/></p>
<p>If a Technical Evaluation was performed, is Compensatory Monitoring required and acceptable?</p>	<p><input type="checkbox"/></p>	<p><input type="checkbox"/></p>	<p><input type="checkbox"/></p>
<p>Are steps to enable the alarm provided in the controlling work document?</p>	<p><input type="checkbox"/></p>	<p><input type="checkbox"/></p>	<p><input type="checkbox"/></p>
<p style="text-align: center;">_____ Signature (SM/US)</p>	<p style="text-align: center;">_____ Print Name</p>	<p style="text-align: center;">_____ Date</p>	
<p>This alarm must be returned to service by:</p>			<p style="text-align: center;">_____ Date</p>
<p style="text-align: center;">_____ Time</p>		<p style="text-align: center;">N/A if not an LCO</p>	

**Attachment 1
(Page 2 of 2)**

OPDP-4-1 - Disabled Alarm Checklist

DISABLED ALARM CHECKLIST																						
<table style="width: 100%; border-collapse: collapse;"> <tr> <th colspan="2" style="text-align: center; padding: 5px;">DISABLED ALARM CHECKLIST</th> </tr> <tr> <td style="width: 50%; padding: 5px;">Site _____ Unit _____</td> <td style="width: 50%; padding: 5px;"></td> </tr> <tr> <td style="padding: 5px;">_____ Panel Number</td> <td style="padding: 5px;"></td> </tr> </table>	DISABLED ALARM CHECKLIST		Site _____ Unit _____		_____ Panel Number		<table style="width: 100%; border-collapse: collapse;"> <tr> <th colspan="4" style="text-align: center; padding: 5px;">ALARM LOCATION</th> </tr> <tr> <td colspan="4" style="padding: 5px; text-align: center;">_____ Window Number</td> </tr> <tr> <td colspan="4" style="padding: 5px; text-align: center;">_____ Node/Mux/Pt or SER/Sensor</td> </tr> </table>				ALARM LOCATION				_____ Window Number				_____ Node/Mux/Pt or SER/Sensor			
DISABLED ALARM CHECKLIST																						
Site _____ Unit _____																						
_____ Panel Number																						
ALARM LOCATION																						
_____ Window Number																						
_____ Node/Mux/Pt or SER/Sensor																						
<p>8. This alarm has been disabled as described in Item 1 of this form and Disabled Alarm Indicators have been placed on affected alarm window(s).</p>																						
<p>Performed by: _____</p> <table style="width: 100%; border-collapse: collapse;"> <tr> <td style="width: 33%; text-align: center; padding: 5px;">Signature</td> <td style="width: 16.5%; text-align: center; padding: 5px;">Print Name</td> <td style="width: 16.5%; text-align: center; padding: 5px;">Date</td> <td style="width: 16.5%; text-align: center; padding: 5px;">Time</td> </tr> </table>					Signature	Print Name	Date	Time														
Signature	Print Name	Date	Time																			
<p>Verified By: _____</p> <table style="width: 100%; border-collapse: collapse;"> <tr> <td style="width: 33%; text-align: center; padding: 5px;">Signature</td> <td style="width: 16.5%; text-align: center; padding: 5px;">Print Name</td> <td style="width: 16.5%; text-align: center; padding: 5px;">Date</td> <td style="width: 16.5%; text-align: center; padding: 5px;">Time</td> </tr> </table>					Signature	Print Name	Date	Time														
Signature	Print Name	Date	Time																			
<p>9. Describe actions necessary to restore annunciator to normal including post-restoration testing.</p> <p>_____</p> <p>_____</p> <p>_____</p>																						
<p>Prepared by: _____</p> <table style="width: 100%; border-collapse: collapse;"> <tr> <td style="width: 33%; text-align: center; padding: 5px;">Signature</td> <td style="width: 16.5%; text-align: center; padding: 5px;">Print Name</td> <td style="width: 16.5%; text-align: center; padding: 5px;">Date</td> <td style="width: 16.5%; text-align: center; padding: 5px;">Time</td> </tr> </table>					Signature	Print Name	Date	Time														
Signature	Print Name	Date	Time																			
<p>Reviewed & Approved by: _____</p> <table style="width: 100%; border-collapse: collapse;"> <tr> <td style="width: 33%; text-align: center; padding: 5px;">SM/US Signature</td> <td style="width: 16.5%; text-align: center; padding: 5px;">Print Name</td> <td style="width: 16.5%; text-align: center; padding: 5px;">Date</td> <td style="width: 16.5%; text-align: center; padding: 5px;">Time</td> </tr> </table>					SM/US Signature	Print Name	Date	Time														
SM/US Signature	Print Name	Date	Time																			
<p>10 This alarm has been restored to normal and tested in accordance with Item 8 of this form and Disabled Alarm Indicator(s) associated with this alarm have been removed.</p>																						
<p>Performed by: _____</p> <table style="width: 100%; border-collapse: collapse;"> <tr> <td style="width: 33%; text-align: center; padding: 5px;">Signature</td> <td style="width: 16.5%; text-align: center; padding: 5px;">Print Name</td> <td style="width: 16.5%; text-align: center; padding: 5px;">Date</td> <td style="width: 16.5%; text-align: center; padding: 5px;">Time</td> </tr> </table>					Signature	Print Name	Date	Time														
Signature	Print Name	Date	Time																			
<p>Verified By: _____</p> <table style="width: 100%; border-collapse: collapse;"> <tr> <td style="width: 33%; text-align: center; padding: 5px;">Signature</td> <td style="width: 16.5%; text-align: center; padding: 5px;">Print Name</td> <td style="width: 16.5%; text-align: center; padding: 5px;">Date</td> <td style="width: 16.5%; text-align: center; padding: 5px;">Time</td> </tr> </table>					Signature	Print Name	Date	Time														
Signature	Print Name	Date	Time																			
<p>11. Compensatory Monitoring of this alarm is terminated and Unit Supervisor notified. N/A if no Compensatory Monitoring required.</p>																						
<p>_____</p> <table style="width: 100%; border-collapse: collapse;"> <tr> <td style="width: 33%; text-align: center; padding: 5px;">Signature</td> <td style="width: 16.5%; text-align: center; padding: 5px;">Print Name</td> <td style="width: 16.5%; text-align: center; padding: 5px;">Date</td> <td style="width: 16.5%; text-align: center; padding: 5px;">Time</td> </tr> </table>					Signature	Print Name	Date	Time														
Signature	Print Name	Date	Time																			

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT

ANNUNCIATOR RESPONSE

1-AR-M4-B

NIS/ROD CONTROL

1-XA-55-4B

Revision 28

QUALITY RELATED

PREPARED/PROOFREAD BY: CECIL DYER

RESPONSIBLE ORGANIZATION: OPERATIONS

APPROVED BY: J. K. WILKES

EFFECTIVE DATE: 07/24/09

LEVEL OF USE: **CONTINUOUS USE**

REVISION

DESCRIPTION: Added notes to windows A-7, B-7 and D-4 that if LEFM is inoperable rod insertion limit alarms and ICS display are not automatically adjusted. (PER 171355 & PCR 09000845)

PERFORMANCE OF THIS PROCEDURE COULD IMPACT REACTIVITY

Source
SER 322
XIS-68-387

Setpoint
 ± 0.4 cubic inch of fluid

<p>XIS-68-387 REAC LEVEL CH-I HYDRO ISOLATOR TROUBLE</p>

Probable Causes

1. Loss of water or pressure in intermediate or primary reference leg of RVLIS system.

NOTE 1

Amber light indicates a possible leak on the transmitter side of the isolator.

NOTE 2

Red light indicates a possible leak on the RCS side of the isolator.

Corrective Actions

- [1] **DISPATCH** operator to local panel in 690 penetration room to check for leakage.
- [2] **EVALUATE** LCO 3.3.3.7.
- [3] **INITIATE** WO as necessary.

References

45N655-04B-0, 47N668-2, 47W600-287, 47W610-68-7

SQN	Page 29 of 46	1-AR-M4-B
1		Rev. 28



NPG Standard
Department
Procedure

TITLE
Annunciator Disablement

OPDP-4
Rev. 0004
Page 1 of 21

Quality Related Yes No

Effective Date 02-02-2009

Responsible Peer Team/Working Group: Operations

Approved by: O. J. Miller
Corporate Functional Manager

2-2-09
Date

Revision Log

Revision or Change Number	Effective Date	Affected Page Numbers	Description of Revision/Change
0	6/24/99 (COC & BFN) Later SQN YSO 6/29/99 WBN 6/30/99 YSO 8/3/99 SQN 8/6/99	All	Initial issue. Replaces PAI-2.08 (WBN), SSP-12.53, 0-PI-OPS-055-001.0 (SQN), and portions of 0-OI-55 (BFN).
1	08/03/00	2-8, 10, 12	These changes were made as a result of revisions to SPP-9.3 and 9.4. The safety assessment was eliminated in SPP-9.4. SPP-9.3 was revised to add a technical evaluation and determination that a change was safe. The wording in this SDP was revised to eliminate references to the safety assessment. (Minor/editorial changes).
2	03/01/04	All	General Revision. Incorporate site-specific changes for WBN and SQN. Eliminates exclusion for 50.59 reviews for nuisance alarms or alarms occurring to known input malfunctions and clarified exclusion of 50.59 from maintenance activities to be consistent with SPP-9.4. Made Disabled Alarm Index Sheet optional to use for tracking. Deleted Index of Activities in Progress That Affect MCR Annunciators OPDP-4-4. Added Alarm Disablement Technical Evaluation OPDP-4-5. Removed specified verification requirements from this procedure and replaced them with "in accordance with SPP-10.3." Made various other enhancements.
3	12/29/08	2, 4	Revised to clarify the scope of the procedure.

NPG Standard Department Procedure	Annunciator Disablement	OPDP-4 Rev. 0004 Page 3 of 21
-----------------------------------------	-------------------------	-------------------------------------

Revision Log

Revision or Change Number	Effective Date	Affected Page Numbers	Description of Revision/Change
4	02/02/09	All 3, 5-11, 14- 16, 19-21	<p>This document has been converted from Word 95 to Word 2003 (XP) using Rev. 3.</p> <p>BFPER 155697 addresses the issue of BFN U1 Vessel Head Leakoff annunciator disablement. Added new statement 3.1.B & C. 3.1.G reworded. Added NOTE to 3.2 and added new 3.2.F. Added 3.5.C. Section 5.0 inserted definitions, Nuisance Alarm & Valid Alarm. Added section 6.0. Reworked forms OPDP-4-1 & OPDP-4-5. Added Appendix B.</p>

Table of Contents

1.0	PURPOSE	5
2.0	SCOPE	5
3.0	INSTRUCTIONS	5
3.1	GENERAL REQUIREMENTS	5
3.2	Disabling an Alarm.....	6
3.3	Enabling an Alarm.....	8
3.4	Identification of Out-of-Service Annunciators.....	9
3.5	Review and Audit.....	10
4.0	RECORDS	10
4.1	QA-Records	10
4.2	Non-QA Records.....	10
5.0	DEFINITIONS	11
6.0	REQUIREMENTS AND REFERENCES.....	11
Appendix A:	Technical Evaluation and 50.59 Applicability.....	12
Appendix B:	RPV Flange Leak Case Study.....	14
Attachment 1:	OPDP-4-1 - Disabled Alarm Checklist	15
Attachment 2:	OPDP-4-2 - Disabled Alarm Index Sheet (Optional)	17
Attachment 3:	OPDP-4-3 - Disabled Alarm Compensatory Monitoring.....	18
Attachment 4:	OPDP-4-5 - Alarm Disablement Technical Evaluation	19

NPG Standard Department Procedure	Annunciator Disablement	OPDP-4 Rev. 0004 Page 5 of 21
--------------------------------------------------	--------------------------------	----------------------------------------------

1.0 PURPOSE

This procedure establishes the requirements for disabling/enabling alarms, tracking disabled alarms, establishing compensatory monitoring requirements, and the use of disabled alarm identifiers on annunciator windows. This instruction also establishes the requirements for disabling of individual inputs to annunciator windows. Attachments to this instruction provide the required approvals, compensatory monitoring, instructional steps, tracking mechanism, and restoration steps for strict control of this activity.

2.0 SCOPE

This procedure provides administrative instructions to control and identify the status of the control room annunciators. This instruction should not be used to circumvent the permanent design change process. It is intended to apply to alarms or alarm inputs for reasons such as the following:

- to address nuisance alarm conditions.
- to restore alarm functions for alarms with multiple inputs.
- to support maintenance or testing activities.

3.0 INSTRUCTIONS

3.1 GENERAL REQUIREMENTS

- A. Before an annunciator can be disabled, the action must be reviewed to ensure that it will not result in an unsafe condition for equipment, personnel, or the public.
- B. Before an alarm can be bypassed, the ability to monitor the condition the alarm is intended to indicate must be evaluated such that if the condition were to occur as the condition changes, Operations personnel will be made aware of the condition change in a timely fashion. A timely fashion is determined by the urgency of the operator response to the alarm.
- C. Any Annunciator disablement requiring a 10CFR50.59 review or a Technical Evaluation (as specified in Appendix A) shall have necessary paperwork completed before disabling the alarm.
- D. Each alarm input to be disabled will be reviewed to determine its impact on Technical Specifications, TRM, ODCM, FSAR, EOIs, Radiological Monitoring, and Environmental Evaluation equipment.
- E. Verification is required for removal and replacement of alarm points associated with safety systems in accordance with SPP-10.3.
- F. An alarm point may be temporarily placed in service to determine if the condition has cleared or if corrective maintenance was sufficient to correct the deficiency provided the steps for enabling are included in the work document.

3.1 GENERAL REQUIREMENTS (continued)

- G. Numerous alarms receive input from multiple points, any of which may cause the alarm to annunciate. If the alarm does not have "reflash" (i.e., the alarm contacts are "Daisy Chained" in the field), other alarm contacts may be masked while the one alarm contact is in. If possible, the cause of the alarm should be determined, and the individual point removed from scan, or leads disconnected in the field to restore the alarm function from other inputs.
- H. Form OPDP-4-1, "Disabled Alarm Checklist" shall be maintained in the Disabled Annunciator Book with a copy of the 50.59 review and Technical Evaluation (if applicable).
- I. The Shift Manager/Unit Supervisor shall evaluate the acceptability of and approve any compensatory monitoring required for annunciators to be disabled.
- J. If an annunciator is found to be in a failed state or inoperable, the Unit Supervisor or Shift Manager should refer to the appropriate Technical Specification and/or FSAR sections to evaluate system operability.

3.2 Disabling an Alarm

NOTE

Appendix B, "RPV Flange Leak Case Study" provides Operating Experience (OE) on inappropriate alarm disablement and should be referred to prior to starting the remainder of the procedure.

Employees Disabling Alarms

- A. Initiate Form OPDP-4-1 for each alarm to be disabled.
- B. Determine if alarm point can be disabled by Operations:

WBN Only

If at WBN, determine if alarm point can be disabled using input number by looking up (NODE/Mux/Pt) on any of the following:

- MCR Alarm Printer (Address/Real Address)
- 47W610 series prints
- MCR Plant SSDs

Refer to SOI-55.01 for disabling instructions.

SQN Only

NPG Standard Department Procedure	Annunciator Disablement	OPDP-4 Rev. 0004 Page 7 of 21
-----------------------------------------	-------------------------	-------------------------------------

3.2 Disabling an Alarm (continued)

If at SQN, determine if alarm point can be disabled using SER number by looking up SER number in 0-SO-55-1. Refer to 0-SO-55-1 for disabling instructions.

BFN Only

If at BFN, software disable functions are NOT available.

- C. If alarm cannot be disabled by operations, then contact Site Engineering/System Engineer to determine a method for disabling the alarm point.
- D. Complete Form OPDP-4-1, items 1-6, and submit to SM/US for review and approval.

STA/Site Engineering

- E. Perform Technical Evaluation using Form OPDP-4-5, if required by Appendix A.

Shift Manager/Unit Supervisor

- F. Ensure there is a WO to correct the problem and if the alarm is an indication of an equipment deficiency, verify a PER has been initiated.
- G. Review controlling work document and ensure annunciator re-enablement is included as an action before completion.
- H. Ensure 50.59 review (SPP-9.4) and Technical Evaluation (Form OPDP-4-5) are attached, if required by Appendix A.
- I. If Technical Evaluation (Form OPDP-4-5) required, review Form OPDP-4-5. Evaluate compensatory monitoring required when alarm is disabled.
- J. Complete Form OPDP-4-1, item 7, and if approved, sign. Ensure WO or PER initiated, as appropriate.

Designated Operator (RO/SRO)

- K. When all required approvals for disabling alarm are obtained, perform the following and complete Form OPDP-4-1, Item 8:
 1. Review disabling steps.
 2. Initiate Compensatory Monitoring as described in Forms OPDP-4-3 and/or OPDP-4-5, if applicable.

3.2 Disabling an Alarm (continued)

3. Designate a qualified individual to disable alarm(s) using method in Form OPDP-4-1, and another individual to perform verification in accordance with SPP-10.3.
4. Place a disabled input indicator on each affected alarm window to indicate an input to that alarm has been disabled.
5. Sign Form OPDP-4-1, Item 8, "Performed By" and submit to a qualified individual to sign verification requirements.
6. If desired to use for tracking, log alarm disablement on Form OPDP-4-2, "Disabled Alarm Index Sheet."
7. Log alarm disablement in the narrative log for the affected unit(s) when alarm disabled. The narrative log entry shall include alarm location, method used to disable alarm, date and time removed, justification for disablement, and Tech Spec action requirements, if applicable.
8. File the disabled alarm Form OPDP-4-1 (with Technical Evaluation and 50.59 review attached, if applicable) in Disabled Annunciator Book.
9. Notify each affected Unit and Radwaste Operator of disabled alarm for:
 - a. Alarms representing unit-shared systems (example: Radwaste).
 - b. Common annunciators.

3.3 Enabling an Alarm

Employees Enabling an Alarm

- A. Obtain the applicable Form OPDP-4-1 from the Disabled Annunciator Book, and Review Item 1.
- B. Prepare enabling steps on Form OPDP-4-1, page 2, Item 9.
- C. Sign as preparer.
- D. Submit Form OPDP-4-1 to the SM or Unit Supervisor for review and approval.

SM/Unit Supervisor

- E. Review Form OPDP-4-1, Item 9 as necessary.
- F. Approve restoration of the alarm in accordance with Form OPDP-4-1.
- G. Designate a qualified individual to restore alarm and another individual to perform verification in accordance with SPP-10.3

NPG Standard Department Procedure	Annunciator Disablement	OPDP-4 Rev. 0004 Page 9 of 21
--------------------------------------------------	--------------------------------	----------------------------------------------

3.3 Enabling an Alarm (continued)

Employees Enabling Alarms

- H. Restore alarm in accordance with applicable operating instruction or as described in Form OPDP-4-1, Item 9.
- I. Remove disabled input indicator from alarm windows which have been enabled, UNLESS there are other inputs to the affected window which remain disabled by OPDP-4 or another approved plant procedure.
- J. Verify associated problem is corrected, and sign Form OPDP-4-1, Item 10.
- K. Ensure a qualified individual signs verification requirements.
- L. Log the enabled alarm in the affected unit(s) narrative log.
- M. If used for tracking, log alarm returned to service on Form OPDP-4-2, "Disabled Alarm Index Sheet."
- N. Ensure Compensatory Monitoring is terminated, if applicable, and sign Form OPDP-4-1, Item 11. Attach completed Compensatory Monitoring forms (Form OPDP-4-3) to Form OPDP-4-1.
- O. Transfer Form OPDP-4-1, including Technical Evaluation (Form OPDP-4-5), 10CFR50.59 review, and Compensatory Monitoring (Form OPDP-4-3) to Management Services

3.4 Identification of Out-of-Service Annunciators

NOTES

- 1) This section applies to annunciators out of service due to maintenance, disabled and or nuisance alarms, or performance of ongoing maintenance or surveillance activities that are affecting the plant's annunciators. Annunciators that are in alarm because of equipment malfunctions should not have indicators placed until the troubled alarm is actually being worked.
- 2) If a clearance is being hung to work a troubled alarm or the clearance will cause an annunciator to alarm, then an indicator should be placed in conjunction with the clearance.
- 3) This section does not negate the requirement to complete Form OPDP-4-1 as specified in Section 3.2.

Employees Performing Activities That will Bring in Alarms

- A. Provide a list of affected alarms to the Unit Operator before starting work.
- B. IF the annunciator is out of service due to maintenance or other abnormal condition, THEN



3.4 Identification of Out-of-Service Annunciators (continued)

Verify WO written which identifies out-of-service annunciator.

- C. Ensure out-of-service indicator is placed on each applicable annunciator window.

Employees Performing Activities That will Bring in Alarms

- D. When maintenance or surveillance activities are complete, then notify Operations to remove out-of-service indicator from affected annunciator windows.

Unit Operator/Designee

- E. Remove out-of-service indicator on alarm windows which have been enabled, UNLESS there are other inputs to the affected window which remain disabled by OPDP-4 or another approved plant procedure.

3.5 Review and Audit

- A. The Disabled Annunciator Book is reviewed during shift turnover (OPDP-1) to ensure disabled alarms are documented as required.
- B. On a monthly basis, the Disabled Annunciator Book should be audited to verify that 10CFR50.59 reviews have been completed as required. A 10CFR50.59 review shall be completed for any annunciators disabled for maintenance which will exceed 90 days prior to the next review.
- C. Every 90 days operations will conduct a critical review of the justification of the disabled alarms to challenge the following:
 - 1. Technical adequacy
 - 2. Monitoring plan
 - 3. Length of time the alarm is disabled

4.0 RECORDS

4.1 QA-Records

- A. Disabled Alarm Checklist OPDP-4-1
- B. Annunciator Disablement Technical Evaluation OPDP-4-5

4.2 Non-QA Records

- A. Disabled Alarm Index Sheet OPDP-4-2
- B. Disabled Alarm Compensatory Measures OPDP-4-3

**Appendix A
(Page 1 of 2)**

Technical Evaluation and 50.59 Applicability

A. When an annunciator window/input is disabled as directed/allowed in an approved plant procedure (excluding maintenance or surveillance activities), a separate 50.59 review and Technical Evaluation are not required since the procedure has already been reviewed and approved. The following example would be an alarm disablement per an approved plant instruction:

- A system operating instruction directs or allows an alarm disablement due to abnormal conditions which are addressed (and restored) by that instruction.

NOTE

The initiation and processing of a work order does NOT constitute in-process maintenance. Refer to Section 5.0 Definitions.

B. If an annunciator window/input is disabled in support of maintenance or surveillance activities, a 50.59 review is not required UNLESS the annunciator will remain disabled for more than 90 days. If 90 days will be exceeded, a 50.59 review shall be completed prior to exceeding 90 days. A Technical Evaluation is required prior to disablement if alarm functions will be disabled for equipment remaining in service (not removed from service/inoperable for the maintenance activity).

1. The following example would be considered necessary to support maintenance activities and requires a Technical Evaluation:

- A pump is tagged with a clearance for maintenance. Its suction pressure switch will be depressurized and disabling the associated low pressure alarm will disable the alarm function for other equipment that must remain in service.

2. The following examples would be considered necessary to support maintenance activities and do not require a Technical Evaluation provided the parameter is the only input to the alarm:

- A pump is tagged with a clearance for maintenance. Its suction pressure switch will be depressurized and the associated low pressure alarm disabled.
- An instrument is declared inoperable, and any required LCO action(s) are entered for calibration in accordance with an approved maintenance instruction. The alarm from this instrument is disabled.

NPG Standard Department Procedure	Annunciator Disablement	OPDP-4 Rev. 0004 Page 13 of 21
-----------------------------------------	-------------------------	--------------------------------------

**Appendix A
(Page 2 of 2)**

Technical Evaluation and 50.59 Applicability

- C. When an annunciator window/input must be disabled due to degraded or inoperable equipment with maintenance NOT in progress, a 50.59 review is required prior to disabling the alarm EXCEPT when covered by an approved plant procedure (item A). A Technical Evaluation is also required EXCEPT when covered by an approved plant procedure (item A) OR when the affected alarm function is only monitoring equipment which is inoperable/out-of-service and the alarm will be restored prior to declaring the affected equipment operable or returning it to service. The following excerpt from NEI 96-07 is an example of a degraded condition affecting multiple alarm inputs:
- A level transmitter for one Reactor Coolant Pump (RCP) lower oil reservoir failed while at power. The transmitter provides an alarm function, but not an automatic protective action function. The transmitter and associated alarm are described in the UFSAR as protective features for the RCPs, but no technical specification applies. Loss of the transmitter does not result in the loss of operability for any technical specification equipment. The transmitter fails in a direction resulting in a continuous alarm in the control room. The alarm circuitry provides a common alarm for both the upper and lower oil reservoir circuits, so transmitter failure causes a hanging alarm and a masking of proper operation of the remaining functional transmitter. Precautionary measures are taken to monitor lower reservoir oil level as outlined in the alarm manual using available alternate means. An interim compensatory action is proposed to lift the leads (temporary change) from the failed transmitter to restore the alarm function for the remaining functioning transmitter. Lifting the leads is a compensatory action (temporary change) that is subject to 10 CFR 50.59. The 10 CFR 50.59 screening would be applied to the temporary change itself (lifted leads), not the degraded condition (failed transmitter) to determine its impact on other aspects of the facility described in the UFSAR. If screening determines that no other UFSAR-described SSCs would be affected by this compensatory action, the temporary change would screen out, i.e., not require a 10 CFR 50.59 evaluation.
- D. If an annunciator window or input must be disabled for other reasons (e.g. due to actual plant parameters which are known/suspected to be at or exceeding the alarm setpoint), then a 50.59 review and Technical Evaluation are required prior to disabling the alarm, EXCEPT when covered by an approved plant procedure (item A).

NPG Standard Department Procedure	Annunciator Disablement	OPDP-4 Rev. 0004 Page 14 of 21
-----------------------------------------	-------------------------	--------------------------------------

Appendix B
(Page 1 of 1)

RPV Flange Leak Case Study

On initial restart after a 22-year idle period, BFN U1 started experiencing high Drywell (D/W) temperature alarms. These were attributed to changes in the ventilation system that occurred during the idle period. The alarm setpoints were raised to clear the alarm.

The Unit also experienced an RPV Flange leakage pressure high alarm. This was attributed to water trapped between the RPV Flange O-rings heating up. It was actually due to a leaking inner O-ring.

The Unit continued to experience high D/W temperature alarms and again the setpoints were raised. The high temperature alarms were not correlated to a known leaking inner RPV seal O-ring and the possibility of an outer leaking O-ring.

The RPV seal high pressure alarm was disabled using this procedure. The person completing the evaluation believed there were no other indications that could be used to warn the operator of degrading conditions so there were no compensatory monitoring actions specified.

The leaking RPV Flange O-rings resulted in increased D/W unidentified leakage. The station responded to increasing leakage by forming a high impact team. Although the team made four D/W entries to find the source of the leak, the key piece of information that would have helped them, the RPV Flange seal high pressure alarm, had been disabled.

The alarm disablement process should not degrade the operator's (station's) ability to detect degrading plant conditions due to conditions getting worse or detecting additional failures (the RPV seal high pressure alarm did not detect the outer O-ring failure) and there was no monitoring to look for it.

Plant personnel did not connect the high D/W temperatures to the RPV seal high pressure or the unidentified leakage because setpoints had been raised to clear alarms and an alarm was disabled without additional monitoring.

**Attachment 1
(Page 2 of 2)**

OPDP-4-1 - Disabled Alarm Checklist

DISABLED ALARM CHECKLIST				
DISABLED ALARM CHECKLIST		ALARM LOCATION		
Site _____	Unit _____	_____		
_____		_____		
Panel Number		Window Number		
		Node/Mux/Pt or SER/Sensor		
8. This alarm has been disabled as described in Item 1 of this form and Disabled Alarm Indicators have been placed on affected alarm window(s).				
Performed by:	_____	_____	_____	_____
	Signature	Print Name	Date	Time
Verified By:	_____	_____	_____	_____
	Signature	Print Name	Date	Time
9. Describe actions necessary to restore annunciator to normal including post-restoration testing.				

Prepared by:	_____	_____	_____	_____
	Signature	Print Name	Date	Time
Reviewed & Approved by:	_____	_____	_____	_____
	SM/US Signature	Print Name	Date	Time
10 This alarm has been restored to normal and tested in accordance with Item 8 of this form and Disabled Alarm Indicator(s) associated with this alarm have been removed.				
Performed by:	_____	_____	_____	_____
	Signature	Print Name	Date	Time
Verified By:	_____	_____	_____	_____
	Signature	Print Name	Date	Time
11. Compensatory Monitoring of this alarm is terminated and Unit Supervisor notified. N/A if no Compensatory Monitoring required.				
	_____	_____	_____	_____
	Signature	Print Name	Date	Time

Attachment 4
(Page 1 of 3)

OPDP-4-5 - Alarm Disablement Technical Evaluation

ALARM DISABLEMENT TECHNICAL EVALUATION	
<p>Site _____ Unit _____</p> <p>_____</p> <p style="text-align: center;">Panel Number</p>	<p style="text-align: center;">ALARM LOCATION</p> <p style="text-align: center;">_____</p> <p style="text-align: center;">Window Number</p> <p style="text-align: center;">_____</p> <p style="text-align: center;">Node/Mux/Pt or SER/Sensor</p>
<p>1. Describe the function of this alarm (e.g. provide indication of abnormal operation, equipment failure, indication of an automatic trip, indication of loss of function):</p> <p>_____</p> <p>_____</p>	
<p>2. Is the alarm due to :</p> <p style="margin-left: 40px;"><input type="checkbox"/> Instrument malfunction</p> <p style="margin-left: 40px;"><input type="checkbox"/> Valid signal</p>	
<p>3. Describe reason for disabling alarm: (List procedure or work document number)</p> <p>_____</p> <p>_____</p>	
<p>4. Describe how this alarm will be disabled:</p> <p>_____</p> <p>_____</p>	
<p>5. What other direct or inferred indications are available to support recognition of the alarming condition? (e.g. if a low flow alarm was disabled a pump trip alarm would be a direct indication. A component cooled by the flow high temperature alarm would be an inferred indication. If a vibration probe is faulty, a portable vibration monitor would be a direct indication. Other vibration probes on the same component would be an inferred indication):</p> <p>_____</p> <p>_____</p>	
<p>6. If there are no direct or inferred indications available, are there supplemental related indications (N/A if direct available) that could be used to detect degrading conditions? (e.g. if a low pressure alarm is bypassed could a high level or pressure in an interconnected system indicate a problem. If a low level alarm is bypassed, could chemistry sample of drainage paths be used to indicate a tank leak):</p> <p>_____</p> <p>_____</p>	
<p>7. How will the operator know:</p> <p>a. If the condition occurs?(for instrument problems only)</p> <p>_____</p> <p>_____</p> <p>_____</p> <p>b. If the condition worsens or changes? (for valid signals)</p> <p>_____</p> <p>_____</p> <p>_____</p>	

**Attachment 4
(Page 2 of 3)**

OPDP-4-5 - Alarm Disablement Technical Evaluation

ALARM DISABLEMENT TECHNICAL EVALUATION			
Site _____ Unit _____ _____ Panel Number	ALARM LOCATION _____ Window Number _____ Node/Mux/Pt or SER/Sensor		
	Yes	No	N/A
8. Can Compensatory Monitoring be performed with current staffing levels and without impacting other required actions under normal, abnormal, or accident conditions (as applicable)? Explain. _____ _____ _____	<input type="checkbox"/>	<input type="checkbox"/>	<input type="checkbox"/>
9. Does Compensatory Monitoring require use of local indications or portable equipment? If YES, identify and evaluate the potential hazardous environmental conditions expected during accident or abnormal conditions (including high radiation levels). _____ _____ _____	<input type="checkbox"/>	<input type="checkbox"/>	<input type="checkbox"/>
10. What is the frequency of monitoring and how will it be controlled? (e.g. recurring W/O, change to operator rounds, system engineer monitoring on a weekly bases). _____ _____ _____			<input type="checkbox"/>
11. For YES responses above, discuss why disabling the affected alarm function is acceptable from a technical and safety standpoint.			
_____ _____ _____			

**Attachment 4
(Page 3 of 3)**

OPDP-4-5 - Alarm Disablement Technical Evaluation

ALARM DISABLEMENT TECHNICAL EVALUATION		
Site _____ Unit _____ _____ <div style="text-align: center;">Panel Number</div>	<div style="text-align: center;">ALARM LOCATION</div> _____ <div style="text-align: center;">Window Number</div> _____ <div style="text-align: center;">Node/Mux/Pt or SER/Sensor</div>	
<p>12. Identify other compensatory actions (e.g., swapping running equipment or Caution Order clearances) that need to be performed to support disabling the alarm.</p> <p>_____</p> <p>_____</p> <p>_____</p> <p>13. If the alarm will be disabled by lifting leads, evaluate the circuit and the proposed lift location. Discuss any adverse impacts or safety concerns identified.</p> <p>_____</p> <p>_____</p> <p>_____</p> <p>14. Identify applicable FSAR sections reviewed.</p> <p>_____</p> <p>_____</p> <p>_____</p> <p>15. Identify the affected annunciator response procedure(s) and other applicable procedures.</p> <p>_____</p> <p>_____</p> <p>_____</p> <p>16. Are temporary revisions to existing procedures required (system operating, ARP, surveillance, maintenance) to address the disabled alarm? <input type="checkbox"/> YES <input type="checkbox"/> NO If YES, list procedures and required revisions.</p> <p>_____</p> <p>_____</p> <p>_____</p> <p>_____</p> <p>_____</p> <p>_____</p> <p>_____</p> <p>_____</p> <p>_____</p>		
Performed By: _____ <div style="display: flex; justify-content: space-around; width: 100%;"> Signature (STA or Engineering) Print Name Date </div>		
Reviewed By: _____ <div style="display: flex; justify-content: space-around; width: 100%;"> Signature (SM or US) Print Name Date </div>		

SEQUOYAH NUCLEAR PLANT
September 2010 NRC Exam

SRO ADMIN C

Evaluate Worker Exposure

SRO
JOB PERFORMANCE MEASURE

Task: Evaluate worker exposure and apply radiation and contamination safety procedures

Task #: 1190100301 (RO); 3430290302 (SRO)

Task Standard: Candidate will calculate the three workers exposures and determine the required administrative actions.

Time Critical Task: YES: _____ NO: X

K/A Reference/Ratings: 2.3.4 (3.2/3.7)

Method of Testing:

Simulated Performance: _____ **Actual Performance:** X

Evaluation Method:

Simulator _____ **In-Plant** _____ **Classroom** X

Main Control Room _____ **Mock-up** _____

Performer: _____
Trainee Name

Evaluator: _____ / _____
Name / Signature DATE

Performance Rating: SAT: _____ UNSAT: _____

Validation Time: 10 minutes **Total Time:** _____

Performance Time: **Start Time:** _____ **Finish Time:** _____

COMMENTS

SPECIAL INSTRUCTIONS TO EVALUATOR:

1. Critical steps are identified in step SAT/UNSAT column by bold print 'Critical Step'.
2. Any UNSAT requires comments
3. Ensure operator performs the following required actions for **SELF-CHECKING**;
 - a. Identifies the correct unit, train, component, etc.
 - b. Reviews the intended action and expected response.
 - c. Compares the actual response to the expected response.

Tools/Equipment/Procedures Needed:

RCI-03

References:

	Reference	Title	Rev No.
1.	RCI-03	Personnel Monitoring	48
2.	RCTP-105	Personnel Inprocessing and Dosimetry Administrative Processes	0001

=====

READ TO OPERATOR

DIRECTIONS TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM. 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. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

1. Radiation surveys in the Auxiliary Building are as follows:
 - 1A-A Charging Pump Room = 40 MR/hr
2. The 1A-A Charging Pump is OOS for pump shaft replacement.
3. The shaft replacement is expected to take 3 workers 12 hours to complete.
4. Annual exposure for each worker up to this time is as follows:
 - Worker 'A' = 10 MR
 - Worker 'B' = 300 MR
 - Worker 'C' = 720 MR

INITIATING CUES:

Assuming all three workers will spend the entire 12 hours in the charging pump room, determine the following:

1. Each workers projected exposures.
2. Any additional approvals required to perform the work, based upon the calculated exposure.

Job Performance Checklist:

STEP / STANDARD	SAT / UNSAT
<p><u>STEP 1.:</u> Determine projected worker exposure for the job.</p> <p>Note: (40 mr/hr x 12 hrs = 480 mr each worker)</p> <p>STANDARD: Operator calculates projected exposure of 480 MR for each worker.</p> <p>This is a critical step necessary to determine overall exposure for job.</p> <p><u>COMMENTS:</u></p>	<p>_____ SAT</p> <p>_____ UNSAT</p> <p>Start Time:</p> <p>_____</p> <p>Critical Step</p>
<p><u>STEP 2.:</u> Determine each workers projected total annual exposure.</p> <p>STANDARD: Worker 'A' projected total annual exposure is calculated to be 490 MR. (10 mr + 480 mr = 490 mr) Worker 'B' projected total annual exposure is calculated to be 780 MR. (300mr + 480 mr = 780 mr) Worker 'C' projected total annual exposure is calculated to be 1200 MR. (720 mr + 480mr = 1200 mr)</p> <p>This step is critical to determine which, if any, of the workers may need dose extensions.</p> <p><u>COMMENTS:</u></p>	<p>_____ SAT</p> <p>_____ UNSAT</p> <p>Critical Step</p>
<p><u>Step 3.</u> Determine any additional approvals required to perform the work, based upon the projected total annual exposure.</p> <p>STANDARD: Worker 'C' must receive authorization from the Site Radiation Protection Manager/RSO to exceed an annual TEDE of 1.0 rem.</p> <p>This is critical step to determine which, if any, worker(s) would need authorization to exceed admin dose limit (ADL).</p> <p><u>COMMENTS:</u></p> <p>CUE: This completes the JPM</p>	<p>_____ SAT</p> <p>_____ UNSAT</p> <p>Critical Step</p> <p>Stop Time:</p> <p>_____</p>

READ TO OPERATOR

DIRECTIONS TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM. 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. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

1. Radiation surveys in the Auxiliary Building are as follows:
 - 1A-A Charging Pump Room = 40 MR/hr
2. The 1A-A Charging Pump is OOS for pump shaft replacement.
3. The shaft replacement is expected to take 3 workers 12 hours to complete.
4. Annual exposure for each worker up to this time is as follows:
 - Worker 'A' = 10 MR
 - Worker 'B' = 300 MR
 - Worker 'C' = 720 MR

INITIATING CUES:

Assuming all three workers will spend the entire 12 hours in the charging pump room, determine the following:

1. Each workers projected exposures.
2. Any additional approvals required to perform the work, based upon the calculated exposure.

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT

RADIOLOGICAL CONTROL INSTRUCTION

RCI-03

PERSONNEL MONITORING

Revision 48

QUALITY RELATED

PREPARED BY: Terry F. Johnston

RESPONSIBLE ORGANIZATION: Radiation Protection

APPROVED BY: JOHN VINCELLI

EFFECTIVE DATE: 02/24/05

VERIFICATION DATE: N/A

LEVEL OF USE: **INFORMATION ONLY**

REVISION

DESCRIPTION

This revision updates the references, includes organizational title changes, removes the previous requirement to list the SSN on TLDs (PER #76092) and updates general information. This revision is an intent revision.

Attachment 01, Prenatal Radiation Exposure Program, is revised to include organizational title changes. This revision is an intent revision.

Attachment 03, Area TLD Monitoring Program, is revised to include organizational title changes. This revision is an intent revision.

SQN	PERSONNEL MONITORING	RCI-03 Revision 48 Page 2 of 16
------------	-----------------------------	-----------------------------------------------------

1.0 PURPOSE

The purpose of this Instruction is to provide guidelines for monitoring personnel external radiation exposures.

2.0 SCOPE

This Instruction establishes the requirements for personnel dose monitoring (PCs, EDs, TLDs), extremity dose monitoring, Administrative Dose Levels (ADLs), emergency exposure guidance, details the prenatal radiation exposure program, provides the requirements for calculation of skin doses, and details the area TLD monitoring program.

3.0 REFERENCES

- A. 10CFR19, Notices, Instructions, and Reports to Workers; Inspection and Investigations
- B. 10CFR20, Standards for Protection Against Radiation
- C. NUREG CR-4418, Dose Calculation for Contamination of the Skin Using the Computer Code VARSKIN
- D. NUREG CR-5569, Health Physics Position Database
- E. NUREG CR-5873, VARSKIN MOD2 and SADDE MOD2: Computer Codes for Assessing Skin Dose from Skin Contamination
- F. NUREG CR-6204, Questions and Answers Based on Revised 10CFR20
- G. Regulatory Guide 1.109, Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purposes of Evaluating Compliance with 10CFR50, Appendix I
- H. Regulatory Guide 8.13, Instruction Concerning Prenatal Radiation Exposure
- I. NRC IE Notice 86-23, Excessive Skin Exposures Due to Contamination of Hot Particles
- J. NRC IE Notice 90-48, Enforcement Policy for Hot Particle Exposures.
- K. SPP-2.3, Document Control
- L. SPP-5.1, Radiological Controls
- M. SPP-5.9, Radiological Control and Radioactive Material Shipment Augmented Quality Assurance Program
- N. ANSI N413-1974, Guidelines for the Documentation of Digital Computer Programs

SQN	PERSONNEL MONITORING	RCI-03 Revision 48 Page 3 of 16
-----	----------------------	---------------------------------------

3.0 REFERENCES (Continued)

- O. INPO 91-014, Guidelines for Radiological Protection at Nuclear Power Stations
- P. SQN Technical Specifications, Unit 1 and Unit 2
- Q. EPIP-15, Emergency Exposure Guidelines
- R. RCI-01, Radiation Protection Program
- S. RCI-04, Radiological Respiratory Protection Program
- T. RCI-11, Bioassay Program
- U. RCI-15, Radiological Postings
- V. RCI-23, Radiation Protection Records
- W. RCI-24, Control of Very High Radiation Areas
- X. RCI-27, External Dosimetry Services (TEDS) Laboratory Quality Manual
- Y. RCI-28, Control of Locked High Radiation Areas
- Z. RCI-29, Control of Radiation Protection Keys
- AA. ANI/MAELU Engineering Bulletin 88-3A, Nuclear Liability Guidance for Hot Particle Contamination
- AB. Revision of Technical Basis for Skin Dose Assessment Process, RIMS L78 880209 800
- AC. Counting Efficiencies for GM Detectors, RIMS L78 871105 800
- AD. Revision of Factors for Calculating Skin Dose from Contamination, RIMS L09 891215 800
- AE. Skin Dose Due to Beta Particles from Noble Gases, RIMS L78 870505 801
- AF. Validation Testing of the Lotus 1-2-3 Application SKINDOSE.WK1, RIMS L78 0606 801
- AG. Response of Panasonic Dosimeters to Submersion Exposure by Xe-133, Hoffman, J.M., and Catchen, G.L., Health Physics, Vol. 58, No. 1
- AH. A Method of Adding Nuclides to VARSKIN and QUINCE Skin Dose Calculation Software, Radiation Protection Management, Vol. 4, No. 6
- AI. RCTP-105, Personnel Inprocessing and Dosimetry Administrative Processes

SQN	PERSONNEL MONITORING	RCI-03 Revision 48 Page 4 of 16
-----	----------------------	---------------------------------------

4.0 DEFINITIONS/ABBREVIATIONS

Calendar Quarter - A calendar quarter is a normal 13 consecutive week period. The first calendar quarter of each year shall begin in the first week of January, and subsequent calendar quarters shall be such that no day is included in more than one calendar quarter, or omitted from inclusion within a calendar quarter.

Committed Effective Dose Equivalent (CEDE) - The sum of the products of the weighting factors applicable to each of the body organs or tissues that are irradiated and the committed dose equivalent to these organs or tissues.

Dose Equivalent - The product of the absorbed dose in tissue, quality factor, and all other necessary modifying factors at the location of interest. The units of dose equivalence are the rem and the Sievert (Sv).

Dose Estimate - (as used in this Instruction) A written estimate of dose received during the current calendar quarter and/or year.

Dosimetry - Personnel monitoring devices used to determine the occupational radiation dose received by an individual.

Electronic Dosimeter (ED) - An electronic dose measuring device worn next to the TLD that is used to give the wearer immediate dose readout. The ED dose is the actual deep dose received and may be used as official dose in the event the TLD is lost.

Form 4 - TVA Form 40763, Form-4 Equivalent, Cumulative Occupational Exposure History. This form is provided as **RCTP-105-7**.

Permanent TLD - A TLD assigned to personnel whose assignment on site is greater than 90 consecutive days. Each TLD (front insert) will contain the employee's name and TLD ID number.

Pocket Chamber (PC) - An ion chamber radiation detection device that is used to give the wearer an immediate dose readout. The pocket chamber dose may be used as official dose in the event that the TLD is lost. A pocket chamber is sometimes called a dosimeter or a DRD (direct reading dosimeter).

Radiation Operations (Rad Ops) - New organizational title for the previous Field Operations section.

Radiation Protection (Rad Protection) - New organizational title for the previous RADCON and Radwaste organizations.

Responsible Onsite Supervisor - A supervisor who is responsible for a specific individual and is on the list of site supervisors authorized to admit personnel to the site.

SQN	PERSONNEL MONITORING	RCI-03 Revision 48 Page 5 of 16
-----	----------------------	---------------------------------------

4.0 DEFINITIONS/ABBREVIATIONS (Continued)

Temporary TLD - A TLD assigned to personnel whose assignment on site is less than or equal to 90 consecutive days. Each TLD will contain an ID number. Upon issuance, the employee's name is recorded on the TLD.

Total Effective Dose Equivalent (TEDE) - The sum of the deep dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures).

Whole Body (or Total Body) - For the purposes of external exposure, the head, trunk (including male gonads), arms above the elbow, or legs above the knee.

5.0 RESPONSIBILITIES

5.1 Radiation Operations Manager

The Rad Ops Manager is responsible for ensuring that all requirements for personnel monitoring are maintained as specified in this Instruction.

5.2 Site Emergency Director (SED)

The SED is responsible for providing written authorization for all emergency radiation doses that may exceed the limits of **10CFR20** (Reference 3.B).

5.3 Site Section Supervisors

The site Section Supervisors are responsible for ensuring that personnel under their supervision comply with all TVAN procedures and instructions concerning radiation dose control.

5.4 Radiation Protection

Rad Protection is responsible for controlling, tracking, monitoring, reviewing, and reporting personnel radiation dose, to include:

- A. Maintenance of employee's radiation dose records. A Rad Protection computer-based records storage system shall be implemented to track and control worker radiation exposure.
- B. Issuance of periodic employee exposure summary reports.

SQN	PERSONNEL MONITORING	RCI-03 Revision 48 Page 6 of 16
-----	----------------------	---------------------------------------

5.0 RESPONSIBILITIES (Continued)

Note Written estimates of current year exposure are provided to an employee upon request. Estimates of dose in the absence of finally determined personnel monitoring results must be clearly indicated as such.

- C. Issuance of a written report regarding the radiation dose received at SQN during the current year to each individual having monitoring records at TVA. This report is provided annually to current employees and upon request to former employees.
- D. Investigation of TLD, PC and/or ED reading discrepancies.
- E. Ensuring that dosimetry values are correct, properly assigned, and entered into the appropriate Rad Protection records data base.
- F. Maintaining dosimetry processing accreditation under the National Voluntary Laboratory Accreditation Program (NVLAP).

5.5 Individual Employee

The individual employee is responsible for complying with all regulations concerning radiation exposure control. Employees are responsible for:

- A. Properly wearing prescribed dosimetry.
- B. Wearing only that dosimetry assigned to them.
- C. Proper care and handling of dosimetry, equipment, and instrumentation.
- D. Pick up and return of dosimetry.
- E. Notifying Rad Ops in the event of lost or damaged dosimetry.
- F. Processing through Rad Protection when arriving, transferring, or terminating at SQN.
- G. Informing Rad Protection whenever radiation exposure or medical radionuclide injections have been, or will be received, from a source other than TVA.

6.0 REQUIREMENTS

6.1 Precautions and Limitations

- A. During normal operations, no individual or group of individuals shall be permitted to receive a radiation dose that exceeds the limits specified in this Instruction.
- B. During emergency situations dose limitations will be as described in **EPIP-15** (Reference 3.Q).

SQN	PERSONNEL MONITORING	RCI-03 Revision 48 Page 7 of 16
-----	----------------------	---------------------------------------

6.0 REQUIREMENTS (Continued)

- C. TLD results are used as the official record of radiation exposure. In the event of a lost or damaged TLD, individuals are responsible for immediately reporting the condition to Rad Ops. In some instances, due to the loss or damage of a monitoring device or the inability of the monitoring device to measure certain types of radiation, it will be necessary to calculate an individual's dose. All calculations shall be documented and included in the employee's personal exposure history record.
- D. If a PC and/or ED is lost, damaged, or offscale, it shall be immediately reported to Rad Ops.
- E. When a PC reads in excess of 3/4 scale, or an ED alarms, the wearer shall report to Rad Ops to have the reading recorded and the PC and/or the ED reset.
- F. Individuals shall verify possession of their assigned dosimetry prior to entering a Radiologically Controlled Area (RCA).
- G. Any individual who enters an RCA shall sign in on an appropriate active **Radiation Work Permit (RWP)** and be monitored for radiation exposure with a TLD and a secondary dosimeter (e.g., ED), unless waived by Rad Protection management.
- H. Dosimetry shall normally be worn on the front of the person between the neck and belt line. It shall be in a clearly visible position. When worn in combination, a PC and/or ED should be located within six inches of the TLD. The beta window side of the TLD should normally face outward. When in a Radiation Area, High Radiation Area, or higher radiological zone classification, the PC and/or ED should be placed in a location that will allow the user to frequently read them.
- I. Extremity doses shall be measured when an individual's extremity dose exceeds, or is expected to exceed, 10% of the annual limit as indicated in **10CFR20**. Extremity dosimetry will be issued when the whole body TLD is not an appropriate monitor of extremity dose.
- J. If it is determined by Rad Ops that the portion of the body most likely to receive the greatest exposure is not in the area of the normal placement location of the TLD, the TLD will be moved to the more appropriate area, and/or multiple TLDs provided.

SQN	PERSONNEL MONITORING	RCI-03 Revision 48 Page 8 of 16
-----	----------------------	---------------------------------------

6.0 REQUIREMENTS (Continued)

- K. Any employee whose radiation dose exceeds any established limits shall not be permitted to enter any RCA for the remainder of the specific monitoring period.
- L. During a site emergency, storage, pickup and collection of dosimetry may be performed at alternate locations as conditions warrant.

6.2 General

- A. Dosimetry processing equipment, TLDs, PCs, and EDs shall be calibrated in accordance with approved procedures.
- B. As a minimum, all assigned TLDs are read at least **semi-annually**. Special TLD readouts are performed as necessary.
- C. TLDs are not required to be stored in their specified storage location upon exiting the plant site each day. However, each individual is responsible for maintaining possession of their TLD and ensuring that it is worn in accordance with the requirements of this Instruction.
- D. Any individual permitted to enter a posted High Radiation Area, Locked High Radiation Area, or Very High Radiation Area shall comply with the requirements of **RCI-15** (Reference 3.U), **RCI-24** (Reference 3.W), **RCI-28** (Reference 3.Y), and/or **RCI-29** (Reference 3.Z), as appropriate. [C.1][C.3]
- E. Dosimetry may be issued by Rad Protection after a **Form-4** (RCTP-105-7, TVA Form 40763), or equivalent, has been initiated and signed by the individual and all applicable requirements have been met (i.e., bioassay, training, etc.).
- F. For individuals requiring an Administrative Dose Level (**ADL**) of less than 500 mrem per year, current year and lifetime dose estimates must be provided and signed by the individual. A **Request for Dosimetry Issuance** (RCTP-105-10, TVA Form 40823), or equivalent, will be used to document current year and lifetime estimates.

SQN	PERSONNEL MONITORING	RCI-03 Revision 48 Page 9 of 16
-----	----------------------	---------------------------------------

6.0 REQUIREMENTS (Continued)

- G. All individuals who have a permanent/temporary TLD at SQN must checkout through Rad Protection prior to terminating work at SQN. In addition, individuals who will visit another licensee or TVA plant, and require a TLD, must checkout prior to leaving SQN.
- H. Area TLDs will be controlled in accordance with the requirements of **Attachment 03, Area TLD Monitoring Program**.

6.3 Assignment of Radiation Dose Limits

- A. If an employee is assigned to work at a non-TVA installation where an exposure to radiation is incurred, the employee shall inform Rad Protection of this assignment. The employee shall turn in their dosimetry, obtain any required bioassays, and complete any requested documentation. When the employee returns, they must report to Rad Protection to update their exposure records.
- B. When visitor or contract personnel have more restrictive dose limits than TVA, the more restrictive limits will be used. It is the responsibility of the contractor to provide written notification to Rad Protection of any company administrative limit.

6.4 Administrative Dose Levels (ADLs)

In addition to the limits of **10CFR20**, ADLs shall be used. The following ADLs shall be observed for routine work:

- A. To ensure that ADLs are not exceeded an administrative control system has been established (refer to **Section 6.7**).
- B. ADLs are based on dosimeters used in determining the reported dose. Results which exceed an ADL, based on other dosimeter data, do not violate the ADL.

6.0 REQUIREMENTS (Continued)

- C. An individual's dose shall be controlled by the ADLs listed in the following table:

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 the individual	Not applicable
Up to 1.0 TEDE (or 3.0 LDE or 10 SDE) all sources	Form 4 (or equivalent) to document current year and previous years dose equivalent	Not Applicable
To exceed 1.0 TEDE (or 3.0 LDE or 10 SDE) all sources	Same as above	Site Radiation Protection Manager / RSO
To exceed 5.0 ³ TEDE all sources	Form 4 information must be verified and a Planned Special Exposure initiated	Site Rad Protection Mgr / RSO, Plant Manager ¹ , and Site VP ² or SED, as appropriate
To exceed 1N ⁴ all sources	Form 4 must be verified	Site Rad Protection Mgr / RSO, Plant Manager ¹ , and Site VP ² or SED, as appropriate

Legend

- 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** TEDE should not exceed 1N rem, where N equals the individual's age in years at their last birthday, without authorization signatures delineated.

SQN	PERSONNEL MONITORING	RCI-03 Revision 48 Page 11 of 16
-----	----------------------	----------------------------------------

6.0 REQUIREMENTS (Continued)

- D. Individuals under the age of **18** shall not be granted RCA access.
- E. Individuals whose lifetime accumulated TEDE is \geq **1N rem** shall be limited to **1,000 mrem/yr**. The administrative controls of previous **Table 1** are applicable.
- F. To authorize ADL increases, an **Administrative Dose Level Extension** (RCTP-105-1, TVA Form 40757) must be completed in accordance with **SPP-5.1** (Reference 3.L). At the discretion of the **Rad Protection Manager**, other methods may be utilized (i.e., a memo covering a group of people). Alternate methods shall include the information required on the **RCTP-105-1**. ADL extensions are tracked on an **ADL Tracking Sheet** (RCTP-105-6, TVA Form 40762).
- G. Any personnel exposure received which is in excess of the limits of **10CFR20** shall be reported by the site **Rad Protection Manager** to the Radiation Effects Advisory Group (REAG) and the appropriate area chief physician for an examination. A medical examination and authorization from the Chief Nuclear Officer and Executive Vice President are required before resumption of duties in RCAs for individuals who have received five times the annual limit of **10CFR20**.
- H. Prenatal exposure will be controlled as described in **Attachment 01, Prenatal Radiation Exposure Program**.
- I. Employees shall be instructed during RADCON training to report to their local TVA medical facility and site Rad Protection whenever they receive medical external radiation therapy or internal radionuclides for diagnosis or treatment (routine diagnostic x-rays need not be reported). Rad Ops shall be contacted and requested to perform a radiation survey on the worker. Based on the results of this survey, the individual may be restricted from entry into the RCA. RCA access will be granted when it can be determined, through bioassay and direct surveys, that the medical treatment does not interfere with the ability to monitor the individual's occupational dose.

SQN	PERSONNEL MONITORING	RCI-03 Revision 48 Page 12 of 16
-----	----------------------	----------------------------------------

6.0 REQUIREMENTS (Continued)

- J. Individuals who have or are undergoing therapeutic radiation exposures can have their ADL lowered to **500 mrem**, absent other circumstances which warrant a higher or lower ADL, upon their written request. A **Therapeutic Medical Radiation Exposure** (RCTP-105-8, TVA Form 40764) can be used for this request. The ADLs for individuals receiving therapeutic medical radiation exposures and individuals with radiologically related medical restrictions should be evaluated on a case-by-case basis. It is recommended that the opinion and recommendations of the individual's treating specialist be solicited. The treating specialist would be most aware of the individual diagnosis, specific therapy, the attendant risks, as well as any unusual susceptibility or precautions necessary regarding workplace radiation exposure. The individual and their supervisor will be counseled by Medical Services. A written record of this counseling shall be made and maintained along with all other supporting documentation. It will be included in the individual's personal history file. For individuals receiving therapeutic medical radiation exposures the individual should have risks clearly explained and be encouraged, but not required, to be placed on a lower ADL.
1. If the individual chooses to be placed on a lower ADL, the individual shall be informed that reasonable accommodations will be made to retain their present job status; however, their present job status cannot be guaranteed.
 2. For individuals with radiologically related medical restrictions, Medical Services, in consultation with the Rad Protection Manager (or designee), will determine if occupational exposure should be administratively restricted.

6.5 Skin Dose From Contamination

Skin dose calculations shall be performed in accordance with the requirements of **RCTP-106, Special Dosimetry Operations**.

6.0 REQUIREMENTS (Continued)

6.6 Emergency Exposure Guidance

- A. It is consistent with the risk concept to accept exposures leading to doses in excess of those appropriate for routine operation when recovery from an accident or major operational difficulty is necessary. Saving of a life, measures to circumvent substantial exposure to the general public, or the preservation of valuable installations may be sufficient cause for accepting above normal exposures. Dose limits for an emergency cannot be specified, but they should be commensurate with the significance of the objective and held to the lowest practical level that the emergency permits.
- B. Any decision to embark on emergency operations which would result in exposures in excess of **10CFR20** should be done in consultation with the most senior member of Rad Protection who is available on a timely basis. The guidelines that should be utilized when assigning administrative exposure limits for emergency conditions are listed below. Actual guidance for emergency situations is described in **EPIP-15**.

Table 2	
Maximum Limiting Whole Body Dose Equivalent to Radiation Workers During Extreme Emergency	
Dose Equivalent	Remarks
10 rem	Taken only to prevent serious damage to the plant or hazard to personnel
25 rem	Taken to save a life

- C. Personnel must be made aware of possible consequences of such an exposure and selected on a voluntary basis. Emergency team members who are expected to respond to a radiological emergency must be aware of the consequences of such exposure.

SQN	PERSONNEL MONITORING	RCI-03 Revision 48 Page 14 of 16
-----	----------------------	----------------------------------------

6.0 REQUIREMENTS (Continued)

6.7 Administrative Control of Radiation Exposure

Note When deemed necessary, Rad Protection Support shall perform a special TLD analysis.

To minimize the potential for an overexposure, Rad Protection Support shall notify the responsible section supervisor in writing when an individual in that supervisor's section is approaching Action Level 1, or has exceeded Action Level 2:

A. Action Level 1

An individual has exceeded **80%** of the ADL. The responsible supervisor shall not use that individual in a posted Radiation Area, High Radiation Area, or higher radiological zone classification, unless no other qualified personnel with lower exposures are available.

B. Action Level 2

An individual has exceeded **90%** of the ADL. The individual shall be restricted from the RCA.

C. Removal of either Action Level restriction requires the completion of an **Administrative Dose Level Extension** and approval of the **Rad Protection Manager**.

SQN	PERSONNEL MONITORING	RCI-03 Revision 48 Page 15 of 16
-----	----------------------	----------------------------------------

7.0 QUALITY ASSURANCE (QA) RECORDS

7.1 QA Records

The following records are QA records and shall be completed, handled, and stored in accordance with **RCI-23** (Reference 3.V):

Request for Dosimetry Issuance
RCTP-105-10 TVA Form 40823

Form-4 Equivalent, Cumulative Occupational Exposure History
RCTP-105-7 TVA Form 40763

Administrative Dose Level Extension
RCTP-105-1 TVA Form 40757

Therapeutic Medical Radiation Exposure
RCTP-105-8 TVA Form 40764

7.2 Non-QA Records

The following records are non-QA records and shall be completed, handled, and stored in accordance with **RCI-23**:

ADL Tracking Sheet
RCTP-105-6 TVA Form 40762

8.0 APPENDICES/ATTACHMENTS

- Attachment 01 Prenatal Radiation Exposure Program
- Attachment 02 Calculation of Skin Dose
- Attachment 03 Area TLD Monitoring Program

SOURCE NOTES

SQN	PERSONNEL MONITORING	RCI-03 Revision 48 Page 16 of 16
-----	----------------------	----------------------------------------

Source Notes

Implementing Statement	Requirements Document	Requirements Statement
C.1	RIMS A02 871116 013 RIMS S53 880208 994	Revise RCI-3 to indicate that each individual entering a high radiation area shall be equipped with a survey meter or alarming dosimeter unless the work is continuously monitored by a RADCON representative with an appropriate survey meter.
C.2	IE Notice #88-063 NER 910813001	Annotated as a reference to indicate high radiation area controls are addressed in the implementation of this Instruction. Cancelled by Revision 44.
C.3	Self-Assessment #SQ-RP-00-002	Transfer information denoted in previous Source Note C.1 to RCI-15 per this Self-Assessment.

SEQUOYAH NUCLEAR PLANT JOB PERFORMANCE MEASURE

SRO ADMIN D

**Classify the Event per the REP
(SGTR with Failed S/G Safety)**

RO/SRO
JOB PERFORMANCE MEASURE

Task: Classify the Event per the REP (SGTR with Failed S/G Safety)

Task #: (SRO) 3440030302; (SRO) 3440190302

Task Standard: The event is classified as a GENERAL EMERGENCY based on Loss of 3 fission barriers (SGTR with Failed S/G Safety).

Time Critical Task: YES: NO:

K/A Reference/Ratings: 2.4.38 (2.4/4.4)
2.4.44 (2.4/4.4)

Method of Testing:

Simulated Performance: **Actual Performance:**

Evaluation Method:

Simulator **In-Plant** **Classroom**

Main Control Room **Mock-up**

Performer: _____
Trainee Name

Evaluator: _____ / _____
Name / Signature DATE

Performance Rating: SAT: UNSAT:

Validation Time: _____ **Total Time:** _____

Performance Time: **Start Time:** _____ **Finish Time:** _____

COMMENTS

SPECIAL INSTRUCTIONS TO EVALUATOR:

1. Critical steps are identified in step SAT/UNSAT column by bold print 'Critical Step'.
2. Any UNSAT requires comments
3. Ensure operator performs the following required actions for **SELF-CHECKING**;
 - a. Identifies the correct unit, train, component, etc.
 - b. Reviews the intended action and expected response.
 - c. Compares the actual response to the expected response.

Tools/Equipment/Procedures Needed:

- EPIP-1 thru EPIP-5, for each student in classroom
- FR Procedures
- Steam Tables, for each student in classroom
- Clock must be available in classroom that all examinees and evaluator can see

References:

	Reference	Title	Rev No.
1.	EPIP-1	Emergency Plan Classification Matrix	43
2.	EPIP-5	General Emergency	39

=====

READ TO OPERATOR

DIRECTIONS TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All steps of this JPM shall be performed in a classroom or in the simulator (simulator will not be set up to match the scenario). The evaluator will provide initiating cues and any other data that may be needed. Time begins when directed by evaluator. When the declaration has been made, raise your hand, the evaluator will record the time, and then you may continue the procedure. Raise your hand again when you have completed the TVA Initial Notification Form, to the point of notifying the ODS. If during the performance of the JPM you have any questions raise your hand and the evaluator will assist you.

INITIAL CONDITIONS:

1. Unit 2 is at 100% RTP and stable.
2. Unit 1 has experienced a Reactor trip and Safety Injection.
3. Security reports that at the time of the trip, steam started blowing from the roof of the Unit 1 east valve vault and steam flow is still in progress at this time.
4. Steam Generator #2 is isolated per E-3, Steam Generator Tube Rupture.
5. CRO has informed you that SG #2 pressure is slowly lowering.
6. Containment pressure is 0.1 psig and steady.
7. RCS pressure is 1500 psig.
8. Core exit TCs 532 °F and slowly rising.
9. The most recent Chem Lab sample of RCS indicates that RCS activity has risen to 345 µCi/gm Equivalent Iodine-131.
10. Emergency Paging System (EPS) is not available in MCR.
11. There are no indications of an Onsite Security Event.

INITIATING CUES:

1. You are the Unit 1 US and have assumed the duties of the SED position, until the TSC is staffed.
2. You are to perform each of the following:
 - a. Classify this event per EPIP-1 and
 - b. Fill out TVA Initial Notification Form and make Protective Action Recommendations, if any.

This is a time critical JPM, Time begins when directed by evaluator.

Job Performance Checklist:

STEP / STANDARD	SAT / UNSAT
<p><u>STEP 1.:</u> Refers to EPIP-1 to determine level of event.</p> <p>STANDARD: Operator refers to EPIP-1, Section 1, Fission Product Barrier Matrix. Operator determines that they have met the conditions of:</p> <p> 1.1.2 Loss, "Primary Coolant Activity Level" 1.2.3 Loss, "SGTR" 1.3.4 Loss, "Containment Bypass"</p> <p>Declaration of event must be made in 15 minutes from the time the task was accepted.</p> <p>Record Time of Declaration: _____</p> <p>Time from Task Acceptance to Declaration: _____</p> <p>Utilizing "Emergency Class Criteria," operator determines the need to declare a General Emergency based on Loss of all 3 barriers. Time of declaration is recorded when the operator raises his/her hand.</p> <p>This is a critical step to arrive at the correct classification within 15 minutes.</p> <p><u>COMMENT:</u></p>	<p>___ SAT</p> <p>___ UNSAT</p> <p>Critical Step</p> <p>Task Start Time</p> <p>_____</p>
<p><u>STEP 2.:</u> Implements EPIP-5 GENERAL EMERGENCY, section 3.1, [1] If TSC is OPERATIONAL, (SED transferred to TSC), THEN GO TO .</p> <p>STANDARD: Operator should recall that the TSC has not been manned per the initiating cues, N/As this step and moves on to the next step.</p> <p><u>COMMENT:</u></p>	<p>___ SAT</p> <p>___ UNSAT</p>
<p><u>STEP 3.</u> [2] RECORD time of declaration. TIME _____</p> <p>NOTE: Operator may have already stated time of declaration, but must enter time properly in EPIP-5 form.</p> <p>STANDARD: Operator enters proper time that declaration was made.</p> <p><u>COMMENT:</u></p>	<p>___ SAT</p> <p>___ UNSAT</p>

Job Performance Checklist:

STEP / STANDARD	SAT / UNSAT
<p>Evaluator Note: Candidate may choose to N/A entire step based on Initiating Cue of EPS not available, if so then N/A JPM step 4 and go to JPM step 9. If step 3 is completed then the following will apply.</p>	
<p><u>STEP 4.:</u> [3] Activate Emergency Paging System (EPS).</p> <p><u>STANDARD:</u> Operator recalls from initial conditions, that EPS is not available from the control room, N/As step and continues on to next step.</p> <p><u>COMMENT:</u></p>	<p>_____ SAT</p> <p>_____ UNSAT</p>
<p><u>STEP 5.:</u> [3] Activate Emergency Paging System (EPS). [a] IF EPS has already been activated, then go to step 4.</p> <p><u>STANDARD:</u> Operator recalls from initial conditions, that EPS has not been previously activated and N/As step.</p> <p><u>COMMENT:</u></p>	<p>_____ SAT</p> <p>_____ UNSAT</p>
<p><u>STEP 6.:</u> [3] Activate Emergency Paging System (EPS). [b] If ongoing onsite Security events may present risk to the emergency responders, Then consult with Security to determine if site access is dangerous to the life and health of emergency responders.</p> <p><u>STANDARD:</u> Operator recalls from initial conditions, that there are no site security threats present and N/As step.</p> <p><u>COMMENT:</u></p>	<p>_____ SAT</p> <p>_____ UNSAT</p>
<p><u>STEP 7.:</u> [3] Activate Emergency Paging System (EPS). [c] If ongoing events makes site access dangerous to the life and health of emergency responders, Then select staging area button on the EPS terminal INSTEAD of the EMERGENCY button.</p> <p><u>STANDARD:</u> Operator recalls from initial conditions, that EPS system not available, so N/As step.</p> <p><u>COMMENT:</u></p>	<p>_____ SAT</p> <p>_____ UNSAT</p>

Job Performance Checklist:

STEP / STANDARD	SAT / UNSAT
<p><u>STEP 8.:</u> [3] Activate Emergency Paging System (EPS). [c] ACTIVATE EPS using touch screen terminal. IF EPS fails to activate, THEN continue with step 4.</p> <p><u>STANDARD:</u> Operator recalls from initial conditions, that EPS system not available, so N/As step.</p> <p><u>COMMENT:</u></p>	<p>_____ SAT _____ UNSAT</p>
<p><u>STEP 9.:</u> [4] EVALUATE Protective Action Recommendations using (Appendix B)</p> <p><u>STANDARD:</u> Operator determines from Appendix B, logic chart in EPIP-5, Note 1: if conditions are unknown then answer is NO, then appropriate protective action recommendation is RECOMMENDATION 2.</p> <p>This step is critical to arrive at the correct PAR recommendation</p> <p><u>COMMENT:</u></p>	<p>___ SAT ___ UNSAT</p> <p>Critical Step</p>

Job Performance Checklist:

STEP / STANDARD	SAT / UNSAT
<p><u>Evaluator Note:</u> When candidate raises their hand to ask for MET tower data, hand them a copy of the MET TOWER LINK data sheet for their use in developing the affected sectors for Protective Action Recommendations.</p>	
<p><u>STEP 10.:</u> [5] Complete Appendix C, TVA Initial Notification of General Emergency.</p> <p><u>STANDARD:</u> Operator completes the Appendix C through step 8, using information from turnover sheet and EPIP-1, prior to Notifying the ODS.</p> <ol style="list-style-type: none"> 1. This is a Drill 2. This is [Their name, Shift Manager (SED) Sequoyah has declared a General Emergency affecting Unit 1. 3. EAL Designators: LOSS 1.1.2 and LOSS 1.2.3, and LOSS 1.3.4. 4. Brief description of incident: [Primary Coolant Activity Level, SGTR AND Containment Bypass]. 5. Radiological Conditions [Release information not known] 6. Event Declared: [Time and Date] 7. Meteorological Conditions are: Wind direction at 46 meters [Southwest at 237 degrees] AND wind speed at 46 meters [4.5 mph] (from Met Tower Data; 46 meters; Instantaneous readings) 8. Protective Action Recommendation: [2 - Evacuate 2 mile radius and 5 miles downwind A-1, B-1, C-1, D-1, B-2, B-5, and shelter remainder of 10 mile EPZ], consider issuance of POTASSIUM IODIDE in accordance with the State Plan. 9. Please repeat back the information you have received to ensure accuracy. 10. Fax information to ODS <p>This is a critical step to ensure information is accurate.</p> <p><u>COMMENT:</u></p>	<p>_____ SAT</p> <p>_____ UNSAT</p> <p>Critical Step</p>

Job Performance Checklist:

STEP / STANDARD	SAT / UNSAT
<p><u>STEP 11.:</u> [6] Notify ODS.</p> <p><u>NOTE:</u> Evaluator Enter time call is made to the ODS _____.</p> <p> Time from Declaration (step 1) to ODS Notification_____.</p> <p><u>STANDARD:</u> Candidate raises their hand as signal that they have completed TVA initial Notification form and are ready to Notify the ODS. ODS should be notified within 10 min after declaration of the event.</p> <p>This is a critical step to (complete forms within 10 min) to ensure required notifications are completed within required time.</p> <p><u>Cue:</u> This completes the JPM.</p> <p><u>COMMENTS:</u></p>	<p>_____ SAT</p> <p>_____ UNSAT</p> <p>Critical Step</p> <p>Stop Time:</p> <p>_____</p>

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT

Key

EMERGENCY PLAN IMPLEMENTING PROCEDURE

EPIP-5

GENERAL EMERGENCY

REVISION 39

PREPARED BY: BILL PEGGRAM

RESPONSIBLE
ORGANIZATION: EMERGENCY PREPAREDNESS

APPROVED BY: RUSSELL THOMPSON

EFFECTIVE DATE: 02/19/2010

LEVEL OF USE: REFERENCE USE

QUALITY-RELATED

Revision History

Rev	Date		Reason for Revision
30	04/01/2003		General Revision to restructure EPIP for better flow. Moved ODS notification earlier in procedures. Added evacuation sectors to Initial Notification Appendix and to consider issuance of KI in accordance with the State Plan in the PAR. Intent Change.
31	06/25/2003	9, 14	Non intent change. Phone number correction. Changed title of Appendix B and added note o match Figure 10-1 of the TVA REP.
32	10/23/2003	4, 8, 12, 13	Intent change. Added step to record time of declaration upon entry into the procedure. Steps concerning dose assessment, PAR, PAR changes, and announce GE to Plant Mgmt., NRC, SM/SED that notifications are complete. Split step that had two actions in one step. Specified Security implement EPIP-11
33	04/22/2004	3, 6, 13	Intent Change: Made corrections to the Table of Contents making sections titles consistent with EPIPs 3,4 and sections within the procedure. Added "SED's Initials" to Section 3.2. Clarified that MSS/WWM in the OSC is verifying ERO response and that SM is to ENSURE that this is in progress. Added guidance to utilize EPIP-6 Apdx B to initially brief NRC using ENS line.
34	09/23/2004	6, 7, 8, 9, 13	Intent Change: Changed Bradley County EMA phone number, removed TEMA satellite phone numbers, added classification validation to Sect.3.2, moved transfer of PARs from Sect. 3.3 to Sect. 3.2, App. A: added to GE announcement to staff TSC/OSC and corrected "SAE" to "General Emergency".
35	06/10/2005	4, 14, 15, 16	Revision Change: replaced SSI-1 with SSI-7.1. Replaced the PAR Chart with the new PAR Chart, NRC Regulatory Issue Summary 2004-13. which addressed the range of protective actions that includes sheltering for the public.
36	04/28/2006	15,16	Revision summary: Changed App.C, Step 10 from being the time and date info was provided to the ODS to faxing App. C to the ODS. Made App. D consistent with App. C by putting "THIS IS A DRILL" before "THIS IS A REAL EVENT".
37	01/23/2007	6, 7, 8, 12, 13	Plan effectiveness determinations reviews indicate the following revisions do not reduce the level of effectiveness of the procedure of REP: Changed the phone number for Bradley County. Changed the call to request a dose assessment from Chemistry to Radiation Protection and changed the corresponding phone numbers so that calls are directed to the RP Lab and no longer the Chem Lab. Changed to current organization titles, added to make Alert announcement on old plant PA and the x4800 bridge. Annual review. Revised responsibility of dose assessment from Chemistry to RP.
38	12/15/2008	8, 11, 15	Plan effectiveness determinations reviews indicate the following revisions do not reduce the level of effectiveness of the procedure of REP. Annual Review. Added a place keeping box in Sect. 3.3[1]. Clarified use of App C when a PAR is changed and CECC has not been activated. Changed SHELTER wording on App. C to match the PAR chart wording on recommendations 1&2.
39	02/19/2010	15	Revision Summary: Plan effectiveness determinations reviews indicate the following revisions do not reduce the level of effectiveness of the procedure of REP: Annual review. PER 162926-001, Revised Appendix C to change step 10 to say, "When completed, FAX this information to the ODS or TEMA as required by Sections 3.1 or 3.2".

Table of Contents

1.0 PURPOSE	4
2.0 REFERENCES	4
3.0 INSTRUCTIONS	4
3.1 General Emergency Declaration by the Main Control Room	4
3.2 General Emergency Declaration by the TSC	7
3.3 Monitor Conditions	8
3.4 Termination of the Event	10
4.0 RECORDS RETENTION	11
4.1 Records of Classified Emergencies	11
4.2 Drill and Exercise Records	11
5.0 ILLUSTRATIONS AND APPENDICES	11
5.1 Appendix A, Notifications and Announcements	12
5.2 Appendix B, Protective Action Recommendations	14
5.3 Appendix C, TVA Initial Notification of General Emergency	15
5.4 Appendix D, General Emergency Follow-up Information	16

1.0 PURPOSE

- 1.1 To provide a method for timely notifications of appropriate individuals or organizations when the Shift Manager (SM)/Site Emergency Director (SED) has determined by EPIP-1 that events have occurred that are classified as a **GENERAL EMERGENCY (GE)**.
- 1.2 To provide the SED/SM a method for periodic reanalysis of current conditions to determine whether the **GENERAL EMERGENCY** should be terminated or continued.

2.0 REFERENCES

2.1 Interface Documents

- [1] SPP-3.5 "Regulatory Reporting Requirements"
- [2] EPIP-6, "Activation and Operation of the Technical Support Center"
- [3] EPIP-7, "Activation and Operation of the Operations Support Center, OSC"
- [4] EPIP-8, "Personnel Accountability and Evacuation"
- [5] EPIP-10, "Emergency Medical Response"
- [6] EPIP-13, "Dose Assessment"
- [7] EPIP-14, "Radiation Protection Response"
- [8] EPIP-16, "Termination and Recovery"
- [9] CECC EPIP-9, "Emergency Environmental Radiological Monitoring Procedures"
- [10] SSI-7.1, "Post Requirements and Responsibilities, Central and Secondary Alarm Stations"

3.0 INSTRUCTIONS

NA **NOTE:** IF there are personnel injuries, **THEN IMPLEMENT** EPIP-10, "Emergency Medical Response" in parallel with this procedure.

NA **NOTE:** IF there are immediate hazards to plant personnel, **THEN** consider immediately implementing EPIP-8 "Personnel Accountability and Evacuation" in parallel with this procedure

3.1 GENERAL EMERGENCY DECLARATION BY THE MAIN CONTROL ROOM

Upon classifying events as a "**GENERAL EMERGENCY**", the SM/SED shall:

[1] IF TSC is OPERATIONAL, (SED transferred to TSC), **THEN GO TO** Section 3.2. *N/A*

[2] **RECORD** time of Declaration _____ Time

3.1 GENERAL EMERGENCY DECLARATION BY THE MAIN CONTROL ROOM (Continued)

- ~~[3]~~
ACTIVATE Emergency Paging System (EPS) as follows.
 - ~~[a]~~
 IF EPS has already been activated, **THEN GO TO Step 4.** ~~N/A~~
 - ~~[b]~~
 IF ongoing onsite Security events may present risk to the emergency responders, **THEN CONSULT** with Security to determine if site access is dangerous to the life and health of emergency responders. ~~N/A~~
 - ~~[c]~~
 IF ongoing events makes site access dangerous to the life and health of emergency responders, **THEN SELECT STAGING AREA** button on the EPS terminal **INSTEAD** of the EMERGENCY button. ~~N/A~~
 - ~~[d]~~
ACTIVATE EPS using touch screen terminal. IF EPS fails to activate, **THEN** continue with step 4. ~~N/A~~
- ~~[4]~~
EVALUATE Protective Action Recommendations (PARs) using Appendix B.
- ~~[5]~~
COMPLETE Appendix C (TVA Initial Notification for General Emergency).

NOTE: ODS should be notified within 5 minutes after declaration of the event.

[6] **NOTIFY ODS.** _____ Initial _____ Time

ODS: Ringdown Line or
5-751-1700 or 5-751-2495 or 9-785-1700

- [a]
 IF EPS failed to activate from SQN, **THEN DIRECT** ODS to activate SQN EPS. IF ODS is also unable to activate EPS, **THEN** continue with step [5] [b].
- [b]
READ completed Appendix C to ODS.
- [c]
FAX completed Appendix C to ODS.

5-751-8620 (Fax)

[d] **MONITOR** for confirmation call from ODS that State/Local notifications complete: **RECORD** time State notified. _____ Notification Time

3.1 GENERAL EMERGENCY DECLARATION BY THE MAIN CONTROL ROOM (Continued)

[7] IF ODS CANNOT be contacted within 10 minutes of the declaration, THEN

[a] CONTACT Hamilton County Emergency Management Agency (EMA) AND READ completed Appendix C.

Initial Time

9-209-6900 or 9-622-7777 or 9-622-0022

[b] CONTACT Bradley County EMA AND READ completed Appendix C.

Initial Time

9-728-7289 or 9-728-7290

[c] NOTIFY Tennessee Emergency Management Agency (TEMA) AND READ completed Appendix C.

Initial Time

9-1-800-262-3300 or 9-1-615-741-0001

[d] FAX completed Appendix C to TEMA.

9-1-615-242-9635 (Fax)

[8] ENSURE MSS/WWM in the OSC (x6427) is monitoring Emergency Response Organization (ERO) responses using printed report available in the OSC.

[a] IF any ERO positions are not responding, THEN DIRECT MSS to CALL personnel to staff TSC/OSC positions. (Use REP Duty Roster and Call List.)

[9] NOTIFY plant staff using Appendix A. (Delegate as needed.)

[10] GO TO Section 3.3

3.2 GENERAL EMERGENCY DECLARATION BY THE TSC

Upon classifying events as a "GENERAL EMERGENCY", the SED shall:

NOTE: CECC Director should be notified within 5 minutes after declaration of the event.

- [1] **RECORD** Time of Declaration _____
- [2] **RECORD** EAL(s) _____
- [3] **VALIDATE** time and EAL numbers with the Ops Mgr, Site VP or EP Mgr.
- [4] IF PAR responsibility has **NOT** been transferred to the CECC Director,
- [a] **THEN REFER** to Appendix B (Protective Action Recommendations)
- [b] **FAX** Appendix C (Notification of General Emergency) to CECC Director
- [5] **CALL** CECC Director and inform of escalation, time of declaration, EAL(s) declared, and description of events.

_____ SED's Initials Time

Ringdown Line or 5-751-1614 or 5-751-1680

- [6] IF CECC Director **CANNOT** be contacted within 10 minutes of the declaration, **THEN**
- [a] **COMPLETE** Appendix C (TVA Initial Notification for General Emergency) using Appendix B to evaluate Protective Actions.
- [b] **NOTIFY** Hamilton County EMA **AND READ** Appendix C. _____
SED's Initials Time

9-209-6900 or 9-622-7777 or 9-622-0022

- [c] **NOTIFY** Bradley County EMA **AND READ** Appendix C. _____
SED's Initials Time

9-728-7289 or 9-728-7290

3.2 GENERAL EMERGENCY DECLARATION BY THE TSC

[d] NOTIFY TEMA AND READ completed Appendix C. _____
 SED's Initials Time

9-1-800-262-3300 or 9-1-615-741-0001

[e] FAX completed Appendix C to TEMA.

9-1-615-242-9635 (Fax)

3.3 MONITOR CONDITIONS

[1] MONITOR radiation monitors.

[2] WHEN indication exists of an unplanned radiological release,
 THEN ENSURE Dose Assessment is performed.

[a] IF the CECC has not assumed Dose Assessment responsibility,
 THEN NOTIFY Radiation Protection to perform a dose
 assessment using EPIP-13, "Dose Assessment"

AND

PROVIDE the following information:

- 1. Type Of Event (SGTR/L, LOCA, WGDT, Cntmt Bypass)
- 2. Release Path (SG/PORV, Aux, Shld, Turb, Serv, Cond)
- 3. Expected Duration (If unknown assume 4 hour duration)

7865 (RP Lab) or 6417 (RP Lab) or
 Use Call List to Page RP Lead

[b] IF changes to PARs are necessary,
 THEN complete Appendix C (IF CECC has not assumed responsibility
 for PARs) and D.

CAUTION: Accountability should **NOT** be initiated at this time **IF** Assembly will
 present a danger to employees - For example:
 A severe weather condition exists or is imminent (such as a Tornado)
 An onsite Security risk condition exists (Consult with Nuclear Security)

[3] IF personnel accountability has not been previously initiated,
 THEN **ACTIVATE** assembly and accountability by using EPIP-8,
 Appendix C (may be delegated).

3.3 MONITOR CONDITIONS (Continued)

[4] MONITOR plant conditions:

[a] EVALUATE conditions using EPIP-1:

[1] IF additional conditions satisfy criteria of other GENERAL EMERGENCY(s) THEN complete Appendix D.

[2] IF conditions warrant a need for follow-up information, THEN complete Appendix D.

[b] IF Appendix D completed, THEN

[1] REPORT to CECC for State notification. _____
Initial Time

CECC Director: Ringdown Line or
 5-751-1614 or 5-751-1680
 OR
 ODS: Ringdown Line or 5-751-1700 or
 5-751-2495 or 9-785-1700

[2] FAX completed Appendix D to CECC.

CECC: 5-751-1682 (Fax) OR ODS: 5-751-8620 (Fax)

[3] IF neither the CECC or ODS can be reached, THEN

[a] NOTIFY TEMA AND READ completed Appendix D. _____
Initial Time

9-1-800-262-3300 or 9-1-615-741-0001

[b] FAX completed Appendix D to TEMA.

9-1-615-242-9635 (Fax)

3.4 TERMINATION OF THE EVENT

[1] IF the situation no longer exists, THEN

- [a] **TERMINATE** emergency per EPIP-16, "Termination and Recovery".
- [b] **COMPLETE** Appendix D including Time and Date Event Terminated.
- [c] **FAX** completed Appendix D to CECC Director.

ODS: 5-751-8620 (Fax) OR
CECC: 5-751-1682 (Fax)

[2] **COLLECT** documentation and **FORWARD** to Emergency Preparedness.

END OF SECTION

4.0 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 and 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 ILLUSTRATIONS AND APPENDICES

5.1 Appendix A - Notifications and Announcements

Appendix A provides guidance for security threats, and for prompt notification of the NRC Resident and plant personnel.

5.2 Appendix B - Protective Action Recommendation Logic Diagram

Appendix B, Protective Action Recommendation Logic Diagram, is used to determine the Protective Action Recommendation which is made to the State and is part of the initial notification made to the State. Protective Action Recommendations are the responsibility of the CECC Director after assuming the responsibility from the SED.

5.3 Appendix C - TVA Initial Notification of General Emergency

Appendix C, TVA Initial Notification of General Emergency, is the form used to initially notify the Operations Duty Specialist who notifies the Tennessee Emergency Management Agency.

5.4 Appendix D - General Emergency Follow-up Information

Appendix D, General Emergency Follow-up Information is the form used to provide additional information concerning other General Emergencies or other information concerning additional conditions to the ODS for State notification and event termination.

Appendix A
 NOTIFICATIONS AND ANNOUNCEMENTS
 (Page 1 of 2)

[1] IF there is a security threat, THEN

[a] NOTIFY Security Shift Supervisor to implement SSI-1, "Security Instructions For Members Of The Security Force" and EPIP-11 "Security and Access Control".

6144 or 6568

Initial Time

[b] DETERMINE if Security recommends implementing the "Two Person Line of Sight" Rule.

[c] IF Nuclear Security recommends establishing the "Two Person Line of Sight" Rule, THEN INFORM the SM/SED. ("Two Person Line of Sight" requires use of EPIP-8.)

Initial Time

[2] NOTIFY Radiation Protection Lead:

[a] STATE: "A GENERAL EMERGENCY HAS BEEN DECLARED, BASED UPON (*Describe the conditions*), AFFECTING UNIT(s) _____."

7865 (RP Lab) or 6417, (RP Lab)
 Use Call List to Page RP Lead

Initial Time

[b] DIRECT Radiation Protection to implement EPIP-14, "Radiation Protection Response".

[c] DIRECT Radiation Protection to implement CECC EPIP-9, "Emergency Environmental Radiological Monitoring Procedures" which includes activation of the radiological monitoring van.

[3] NOTIFY personnel in the Chemistry Lab:

[a] STATE: "A GENERAL EMERGENCY HAS BEEN DECLARED, BASED UPON (*Describe the conditions*), AFFECTING UNIT(s) _____."

7285 (Lab) or 6348 (Lab) or 20126 (Pager)

Initial Time

[b] DIRECT Chemistry to implement EPIP-14, "Radiation Protection Response".

Appendix A
NOTIFICATIONS AND ANNOUNCEMENTS
 (Page 2 of 2)

- [4] **ANNOUNCE** to plant personnel on old plant PA and x4800:
 - [a] "ATTENTION PLANT PERSONNEL. ATTENTION PLANT PERSONNEL. A **GENERAL EMERGENCY** HAS BEEN DECLARED BASED ON (Describe the condition), AFFECTING UNIT(s) _____. (if not already staffed, add) STAFF THE TSC AND OSC."
 - [b] **REPEAT** Announcement.

- [5] **NOTIFY** Plant Management in accordance with SPP-3.5 **AND PROVIDE** General Emergency Information. _____
Initial Time

- [6] **NOTIFY** the "On Call" NRC Resident **AND PROVIDE** General Emergency Information. _____
Initial Time

NOTE: NRC ENS notification should be made as soon as practicable, but within 1 hour of "**GENERAL EMERGENCY**" declaration. Whenever NRC requests, a qualified person must provide a continuous update to NRC Operations Center. Use EPIP-6, Appendix B as a briefing guide.

- [7] **NOTIFY** NRC of plan activation via ENS phone _____
Initial Time
 - 9-1-(301) 816-5100 (Main)
 - 9-1-(301) 951-0550 (Backup)
 - 9-1-(301) 816-5151 (Fax)

- [8] **NOTIFY** the SM/SED that notifications are complete. _____
Initial Time

Appendix B

PROTECTIVE ACTION RECOMMENDATIONS

Note 1: If conditions are unknown utilizing the flowchart, then answer is NO.

Note 2: A short term release is defined as "a release that does not exceed a 15 minute duration".

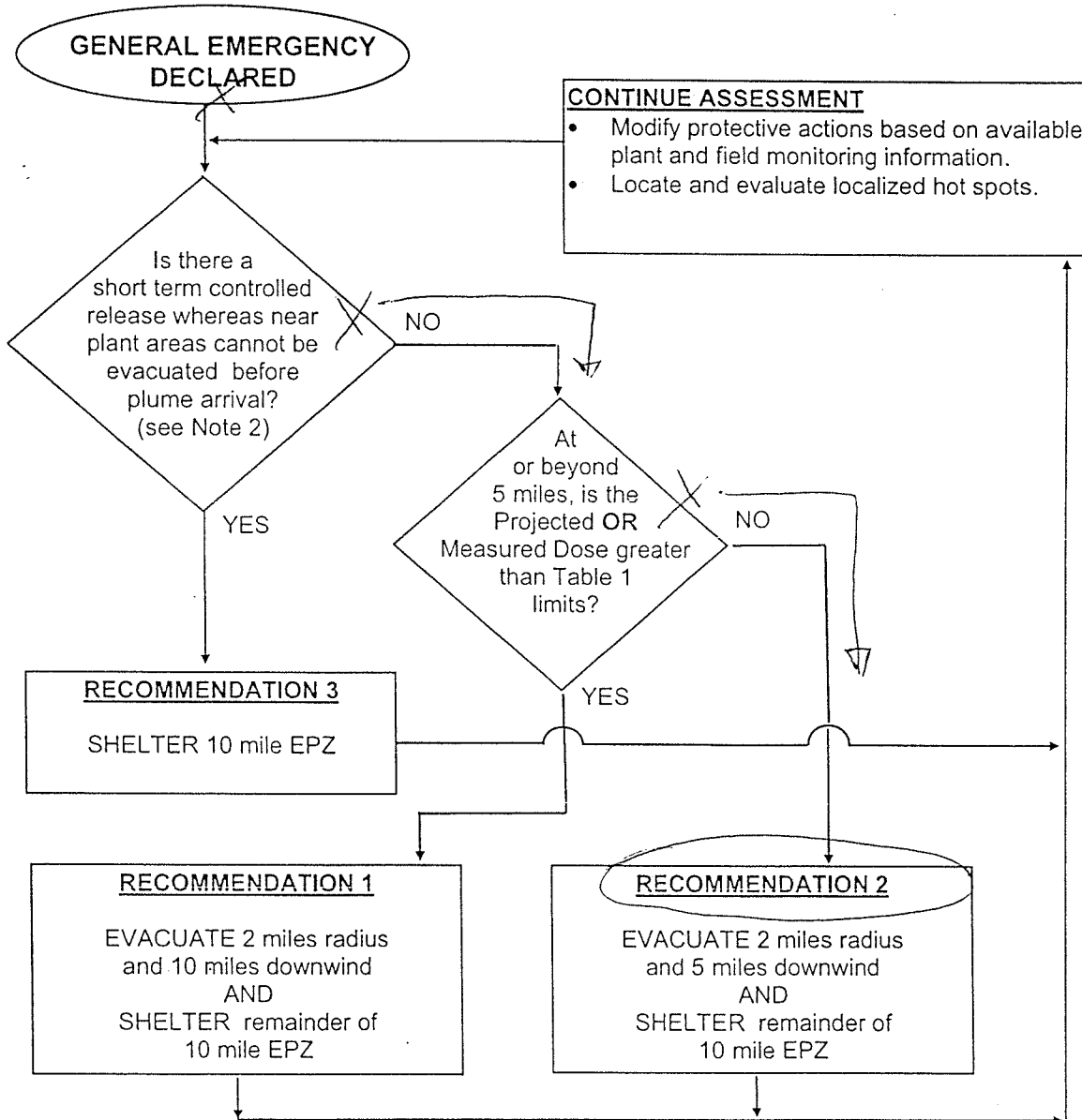


TABLE 1 Protective Action Guides (PAG)	
TYPE	LIMIT
Measured	3.9 E-6 micro Ci/cc of Iodine 131 or 1 REM per hour External Dose
Projected	1 REM TEDE or 5 REM Thyroid CDE

Note: Unknown conditions are assumed less than listed conditions.

Appendix C
TVA INITIAL NOTIFICATION OF GENERAL EMERGENCY

1. This is a Drill This is an Actual Event - Repeat - This is an Actual Event

2. This is (Name), Sequoyah has declared a **GENERAL EMERGENCY**
affecting: Unit 1 Unit 2 Both Unit 1 and Unit 2

3. EAL Designator(s): 1.1, 2L + 1.2, 3L + 1.3, 4L (or Loss 1.1, 2 and Loss 1.2, 3 and Loss 1.3, 4)

4. Brief Description of the Event: Primary Coolant Acidity level, SGT R, and Containment Bypass

5. Radiological Conditions: (Check one under both Airborne and Liquid column.)

<u>Airborne Releases Offsite</u>	<u>Liquid Releases Offsite</u>
<input type="checkbox"/> Minor releases within federally approved limits ¹	<input type="checkbox"/> Minor releases within federally approved limits ¹
<input type="checkbox"/> Releases above federally approved limits ¹	<input type="checkbox"/> Releases above federally approved limits ¹
<input checked="" type="checkbox"/> Release information not known (¹ Tech Specs)	<input type="checkbox"/> Release information not known (¹ Tech Specs)

6. Event Declared: Time: _____ Date: _____

7. The Meteorological Conditions are: (Use 46 meter data from the Met Tower)

Wind Direction is FROM: 237 degrees Wind Speed: 4.5 m.p.h

8. Provide Protective Action Recommendation: (Check either 1 or 2 or 3.)

	R E C	WIND FROM *DEGREES DIRECTION (item 7) (Mark)	R E C	
<input type="checkbox"/> Recommendation 1 • EVACUATE LISTED SECTORS (2 mile Radius and 10 miles downwind) • SHELTER remainder of 10 mile EPZ • CONSIDER issuance of POTASSIUM IODIDE in accordance with the State Plan.	1		2	<input checked="" type="checkbox"/> Recommendation 2 • EVACUATE LISTED SECTORS (2 mile Radius and 5 mile downwind) • SHELTER remainder of 10 mile EPZ • CONSIDER issuance of POTASSIUM IODIDE in accordance with the State Plan.
A-1, B-1, C-1, D-1, C-2, -6, -7, -8, D-2, -3, -5, -6		12 - 49		A-1, B-1, C-1, D-1, C-2, D-2
A-1, B-1, C-1, D-1, D-2, -3, -4, -5, -6		50 - 70		A-1, B-1, C-1, D-1, D-2
A-1, B-1, C-1, D-1, A-3, -4, D-2, -3, -4, -5		71 - 112		A-1, B-1, C-1, D-1, A-3, D-2
A-1, B-1, C-1, D-1, A-2, -3, -4, -5, -6, D-4		113 - 146		A-1, B-1, C-1, D-1, A-2, A-3,
A-1, B-1, C-1, D-1, A-2, -3, -4, -5, -6, B-2		147 - 173		A-1, B-1, C-1, D-1, A-2, A-3, B-2
A-1, B-1, C-1, D-1, A-2, -5, -6, B-2, -3, -4		174 - 214		A-1, B-1, C-1, D-1, A-2, B-2,
A-1, B-1, C-1, D-1, B-2, -3, -4, -5, -6, -7, -8		215 - 258	<input checked="" type="checkbox"/>	A-1, B-1, C-1, D-1, B-2, B-5,
A-1, B-1, C-1, D-1, B-2, -3, -5, -6, -7, -8, C-2, -3, -4, -5, -6		259 - 331		A-1, B-1, C-1, D-1, B-2, B-5, C-2
A-1, B-1, C-1, D-1, B-5, C-2, -3, -4, -5, -6, -7, -8		332 - 11		A-1, B-1, C-1, D-1, B-5, C-2

Recommendation 3
▶ SHELTER all sectors.
▶ CONSIDER issuance of Potassium Iodide in accordance with the State Plan.

9. Please repeat back the information you have received to ensure accuracy.

10. When completed, FAX this information to the ODS or TEMA as required by Sections 3.1 or 3.2.

**Appendix D
GENERAL EMERGENCY FOLLOW-UP INFORMATION**

1. THIS IS A DRILL THIS IS A REAL EVENT

2. There has been a **GENERAL EMERGENCY** declared at Sequoyah affecting:
 Unit 1 Unit 2 Both Unit 1 and Unit 2

3. **Reactor Status:** Unit 1: Shut Down At Power Refueling N/A
 Unit 2: Shut Down At Power Refueling N/A

4. **Additional EAL Designators** _____

5. **Significant Changes in Plant Conditions:** _____

6. **Significant Changes in Radiological Conditions:** _____

7. **Offsite Protective Action Recommendation:** (CECC to provide detailed PAR Sector Recommendations)

Recommendation 1 -
 Evacuate 2 mile radius and 10 miles downwind and shelter remainder of the 10 mile EPZ

Recommendation 2 -
 Evacuate 2 mile radius and 5 miles downwind and shelter remainder of the 10 mile EPZ

Recommendation 3 - Shelter all sectors.

8. **Onsite Protective Actions:** Assembly and Accountability No Initiated Completed
 Site Evacuation No Initiated Completed

9. **The Meteorological Conditions are:** Wind Speed: _____ m.p.h.
 (Use 46 meter data on the Met Tower) Wind Direction is from: _____ degrees

10. **Event Terminated:** Date/Time _____

11. **Please repeat the information you have received to ensure accuracy.**

12. **FAX to ODS at 5-751-8620 or CECC Director at 5-751-1682 after completing the notification.**

Completed by: _____, Date/Time _____

READ TO OPERATOR

DIRECTIONS TO TRAINEE:

I will explain the initial conditions, and state the task to be performed. All steps of this JPM shall be performed in a classroom or in the simulator (simulator will not be set up to match the scenario). The evaluator will provide initiating cues and any other data that may be needed. Time begins when directed by evaluator. When the declaration has been made, raise your hand, the evaluator will record the time, and then you may continue the procedure. Raise your hand again when you have completed the TVA Initial Notification Form, to the point of notifying the ODS. If during the performance of the JPM you have any questions raise your hand and the evaluator will assist you.

INITIAL CONDITIONS:

1. Unit 2 is at 100% RTP and stable.
2. Unit 1 has experienced a Reactor trip and Safety Injection.
3. Security reports that at the time of the trip, steam started blowing from the roof of the east valve vault and steam flow is still in progress at this time.
4. Steam Generator #2 is isolated per E-3, Steam Generator Tube Rupture.
5. CRO has informed you that SG #2 pressure is slowly lowering.
6. Containment pressure is 0.1 psig and steady.
7. RCS pressure is 1500 psig.
8. Core exit TCs 532 °F and slowly lowering.
9. The most recent Chem Lab sample of RCS indicates that RCS activity has risen to 345 µCi/gm Equivalent Iodine-131.
10. Emergency Paging System (EPS) is not available in MCR.
11. There are no indications of an Onsite Security Event.

INITIATING CUES:

1. You are the Unit 1 US and have assumed the duties of the SED position, until the TSC is staffed.
2. You are to perform each of the following:

- a. Classify this event per EPIP-1

And

- b. Fill out TVA Initial Notification Form and make Protective Action Recommendations, if any.

This is a time critical JPM, Time begins when directed by evaluator

MET-TOWER LINK

DOSE
CALC
NET DATA

WIND
DIRECTION

INSTANTANEOUS 15 MIN AVG

AIR TEMPERATURE: 91.70 DEG F 91.99
 WIND SPEED: 5.7 MPH 6.2
 WIND DIRECTION FROM: 226 DEGREES 225

RAINFALL
 LAST HOUR: 0.00 INCHES
 LAST 15 MIN: 0.00 INCHES

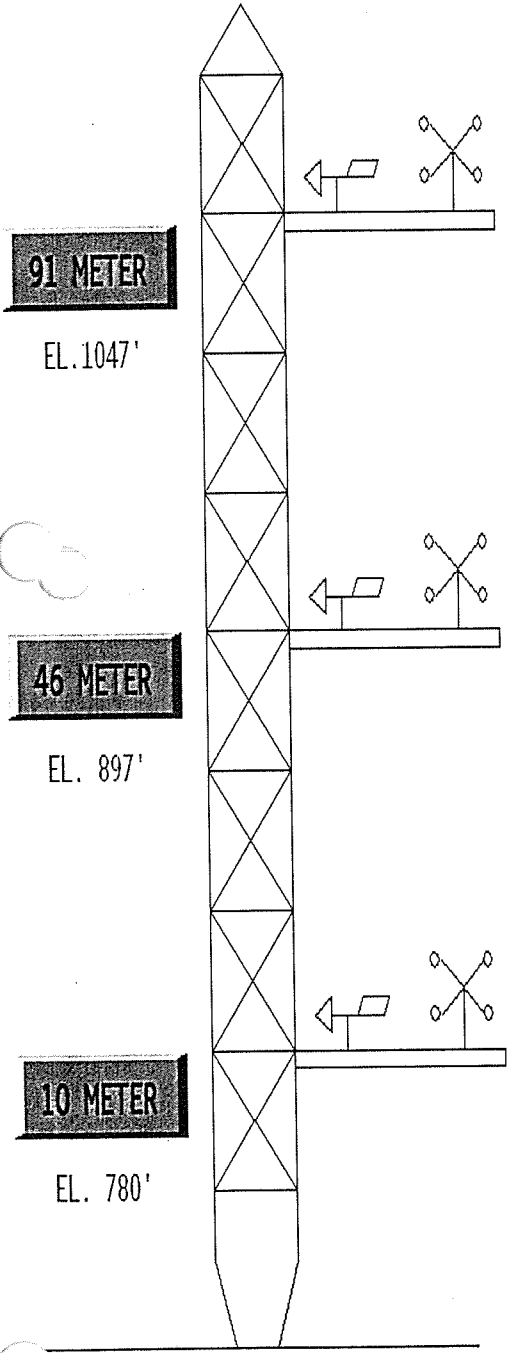
AIR TEMPERATURE: 92.23 DEG F 92.58
 WIND SPEED: 4.5 MPH 6.3
 WIND DIRECTION FROM: 237 DEGREES 231

10 METER DEW POINT
 (HOURLY AVG): 70.04 DEG F

AIR TEMPERATURE: 93.15 DEG F 93.69
 WIND SPEED: 5.0 MPH 6.3
 WIND DIRECTION FROM: 270 DEGREES 235

RIVER

STABILITY
DELTA-T'S



91 METER

EL. 1047'

46 METER

EL. 897'

10 METER

EL. 780'

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT
EMERGENCY PLAN IMPLEMENTING PROCEDURE

EPIP - 1
EMERGENCY PLAN
CLASSIFICATION MATRIX

Revision 44

QUALITY RELATED

PREPARED BY: Bill Peggram

RESPONSIBLE ORGANIZATION: Emergency Preparedness

APPROVED BY: Russell Thompson

EFFECTIVE DATE: 03/31/2010

Level of Use: Reference

Revision History

Rev	Date	Revised Pages	Reason for Revision
39	01/23/2007	46	Plan effectiveness determination reviews indicate the following revisions do not reduce the level of effectiveness of the procedure or REP. Revised effluent radiation monitor EAL values for RM-90-212.
40	12/19/2007	9, 10, 35	Annual Review, Plan Effectiveness Determination reviews indicate the following revisions do not reduce the level of effectiveness of the procedure or REP. Added note to 1.1.5L and 1.3.5P directing users to Fission Product Barrier Matrix Instruction Number 4. Changed 5.5 NOUE from 673 to 674 feet.
41	03/30/2009	9, 10, 14, 15, 33	Annual Review, Plan Effectiveness Determination reviews indicate the following revisions do not reduce the level of effectiveness of the procedure or REP. Added "A" to RM-90-271 and 273. Changed 1.1.5L and 1.3.5P to reflect DCN 21643A Changed wording on SED Judgment section of the Fission Product Barriers to read more accurately by changing the word "of" to "or". Corrected EAL 2.3 Alert to match the REP, App B by changing the "1 or 2" in the opening statement to say "1 and 2". Revised NOUE EALs 2.5 RCS Unidentified Leakage & 2.6 RCS Identified Leakage to include RCS Flow Balance Calculation. RCS leakage through a SG to the secondary system (primary to secondary leakage) was also added to NOUE 2.6 EAL. Clarified alternate met data note in EAL 5.5 Alert.
42	05/15/2009	2, 46	Annual Review, Plan Effectiveness Determination reviews indicate the following revisions do not reduce the level of effectiveness of the procedure or REP. D21643A: added "A" to 1-RM-90-256A - U-1 CVE. Corrected the note in Revision 41 revision log: "Clarified alternate met data note in EAL 5.2 [instead of 5.5] Alert".
43	02/09/2010	9, 44, 46	Annual Review, Plan Effectiveness Determination reviews indicate the following revisions do not reduce the level of effectiveness of the procedure or REP. Revised to show the complete or accurate radiation monitor UNIDs in EAL 1.1.5 due to DCNs 21643 and 21644 and correct EAL 7.4 and correct units on Table 7-1, Effluent Radiation Monitor EALs.
44	03/31/2010	8,12, 20, 24,27,29,32, 38,42, 46	Annual Review, Plan Effectiveness Determination reviews indicate the following revisions do not reduce the level of effectiveness of the procedure or REP. Revised EAL 4.6 due to NEI99-01 R5 and NEI 03-12 R6, Appendix C. Appendix C details Security Regulations. Added definition for SECURITY CONDITION. Removed TABLE 4-3 SECURITY EVENTS. DCN 21644-add "A" to 2-RM-90-256.

TABLE OF CONTENTS

1.0 Purpose4

2.0 References4

2.1 Developmental References.....4

3.0 Instructions4

3.1 REP Activation.....4

3.2 EAL Interpretation5

3.3 Validation of Information5

3.4 Classification Determination5

4.0 Records Retention6

4.1 Records of Classified Emergencies6

4.2 Drill and Exercise Records.....6

EAL Section 1 - FISSION PRODUCT BARRIER MATRIX.....7

EAL Section 2 - SYSTEM DEGRADATION11

EAL Section 3 - LOSS OF POWER19

EAL Section 4 - HAZARDS and SED JUDGMENT23

EAL Section 5 - DESTRUCTIVE PHENOMENON31

EAL Section 6 - SHUTDOWN SYSTEM DEGRADATION37

EAL Section 7 - RADIOLOGICAL EFFLUENTS41

1.0 PURPOSE

This procedure provides criteria to the Shift Manager (SM) or Site Emergency Director (SED) to be used in classifying and declaring an emergency based on plant conditions. The responsibility for declaring an emergency, based on the criteria in this procedure, belongs to the SM or SED, the designated Unit Supervisor when acting as the SM, or the TSC SED. This responsibility **cannot** be delegated.

2.0 REFERENCES

2.1 Developmental Documents

- A. 10 CFR 50, Domestic Licensing of Production and Utilization Facilities.
- B. Reg Guide 1.101, Emergency Planning and Preparedness For Nuclear Power Reactors endorsing NEI 99-01 Methodology For Development Of Emergency Action Levels - Revision 4, January 2003.
- C. Sequoyah Technical Specifications (Tech Specs), Abnormal Operating Procedures (AOPs), Emergency Operating Procedures (EOPs), Functional Restoration Guidelines (FRGs), Technical Instructions (TI), Surveillance Instructions (SI), and the Updated Final Safety Analysis Report (UFSAR) are also referenced in Appendix B of the Radiological Emergency Plan.
- D. Letter to Bruce A. Boger, Director of Inspection Program Management, USNRC, December 8, 2001 from Lynnette Hendricks, NEI, Recommended Actions in Response to a Site Specific Credible Threat at a Nuclear Power Plant (1A-01-1).

3.0 INSTRUCTIONS

3.1 REP Activation

The Nuclear Power (NP) Radiological Emergency Plan (REP) will be activated when any one of the conditions listed in this matrix is detected and declared. The REP is not activated based on a reporting of past conditions. This procedure will be used in conjunction with the REP Appendix B.

If the event is determined to be one of the four emergency classifications then implement EPIP-2, -3, -4, or -5 as applicable.

3.0 INSTRUCTIONS (Continued)

3.2 EAL Interpretation

The criteria in SQN EPIP-1 are given for reference: knowledge of actual plant conditions or the extent of the emergency may require that additional steps be taken. In all cases, this logic procedure should be combined with the sound judgment of the SM or SED to arrive at an appropriate classification for a particular set of circumstances. These criteria apply to both Unit 1 and Unit 2. The SED must be aware of the affects of simultaneous events on both units.

3.3 Validation of Information

If there is a reason to doubt if a given initiating condition has actually occurred, the SM or SED shall follow indications provided. Unless a suspected spurious or otherwise false alarm can be substantiated within an acceptable timeframe (based on potential severity of the event), the SM or SED is to proceed with actions as required by this procedure until such time as the alarm is verified to be false.

3.4 Classification Determination

- 3.4.1 To determine the classification of the emergency, review the Initiating Conditions of the respective status tree criteria that will be monitored and used to determine the event classification for the modes listed on the classification matrix.
- 3.4.2 If a Critical Safety Function (CSF) is listed as an Initiating Condition the respective status tree criteria will be monitored and used to determine the event classification for the modes listed on the classification matrix.
- 3.4.3 Declare the highest emergency class based on **events that are in progress** at the time that the classification is made.
- 3.4.4 **If, during an ongoing event,** investigation shows that a higher classification was previously met, then report that, as information only, to the Operation Duty Specialist (ODS) and the NRC. Do not declare or upgrade to a higher emergency class if the conditions do not exist unless it is a noted exception (i.e., EAL 2.3).
- 3.4.5 **If, following termination of an emergency declaration,** investigation shows that a higher classification was met, then report that, as information only, to the ODS and the NRC. Do not declare or upgrade to a higher emergency class if the conditions do not exist.
- 3.4.6 **If conditions have returned to a non-emergency state** before any emergency can be classified, then the highest emergency class that was appropriate shall be reported, as information only, to the ODS and NRC and shall not be declared unless it is a noted exception (i.e., EAL 2.3).

3.0 INSTRUCTIONS (Continued)

- 3.4.7 The NRC shall be notified within one hour of all classifications. Once made and reported, a declaration cannot be canceled or rescinded even if it is later determined to be invalid. If there is reason to doubt that a given condition has occurred, the SM or SED shall follow indications and proceed with classification, as required by this procedure, until otherwise proven false.
- 3.4.8 The State shall be notified by the ODS within 15 minutes of any declaration and notified, for information only, within one hour of any classification that was met but not declared as allowed above. If the State is notified of a declaration that is **invalidated before the NRC is notified**, terminate the classification, if not already done, and report the declaration to the NRC.
- 3.4.9 The **ACCEPTABLE** timeframe for initiating notification to the ODS of an emergency declaration is considered to be five (5) minutes. This is the time period between declaration of the emergency and contacting the ODS.

4.0 RECORDS 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 and 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.

1	FISSION PRODUCT BARRIER MATRIX (Modes 1-4) 1.1 Fuel Clad Barrier 1.2 RCS Barrier 1.3 Containment Barrier
----------	--------------------------------------------------------------------------------------------------------------------------

2

SYSTEM DEGRADATION

- | | |
|----------------------------------------------------------------------------------------------------------------------------|-------------------------------------------------------------------------------------------------------------------------------------|
| 2.1 Loss of Instrumentation
2.2 Loss of Communication
2.3 Failure of Reactor Protection
2.4 Fuel Clad Degradation | 2.5 RCS Unidentified Leakage
2.6 RCS Identified Leakage
2.7 Uncontrolled Cool Down
2.8 Turbine Failure
2.9 Safety Limit |
|----------------------------------------------------------------------------------------------------------------------------|-------------------------------------------------------------------------------------------------------------------------------------|

3

LOSS OF POWER

- 3.1 Loss of AC (Power Ops)
- 3.2 Loss of AC (Shutdown)
- 3.3 Loss of DC

4

HAZARDS and SED JUDGMENT

- | | |
|---------------------------------------------------------------------------------------------------------------------------------------------|----------------------------------------------------|
| 4.1 Fire
4.2 Explosion
4.3 Flammable Gas
4.4 Toxic Gas or Smoke
4.5 Control Room Evacuation
4.6 Security
4.7 SED Judgment | Table 4-1
Table 4-2
Figure 4-A
Figure 4-B |
|---------------------------------------------------------------------------------------------------------------------------------------------|----------------------------------------------------|

5

DESTRUCTIVE PHENOMENON

- | | |
|----------------------------------------------------------------------------------|------------------------------------------------------------------------|
| 5.1 Earthquake
5.2 Tornado
5.3 Aircraft/Projectile
5.4 River Level High | 5.5 River Level Low
5.6 Watercraft Crash
Table 5-1
Figure 5-A |
|----------------------------------------------------------------------------------|------------------------------------------------------------------------|

6

SHUTDOWN SYSTEM DEGRADATION

- 6.1 Loss of Shutdown Systems
- 6.2 Loss of Shutdown Capability
- 6.3 Loss of RCS Inventory

7

RADIOLOGICAL EFFLUENTS

- | | |
|--------------------------------------------------------------------------------------------------------------------|--------------------------------------|
| 7.1 Gaseous Effluent
7.2 Liquid Effluent
7.3 Radiation Levels
7.4 Fuel Handling
7.5 Spent Fuel Storage | Table 7-1
Table 7-2
Figure 7-A |
|--------------------------------------------------------------------------------------------------------------------|--------------------------------------|

Definitions and Abbreviations:

BOMB: An explosive device. (See EXPLOSION)

CIVIL DISTURBANCE: A group of twenty (20) or more persons within the EAB violently protesting onsite operations or activities at the site.

CONFINEMENT BOUNDARY: Spent Fuel Storage Cask CONFINEMENT BOUNDARY consists of MPC shell, bottom baseplate, MPC lid (including the vent and drain port cover plates), MPC closure ring, and associated welds.

CRITICAL-SAFETY FUNCTION (CSFs): A plant safety function required to prevent significant release of core radioactivity to the environment. There are six CSFs: Subcriticality, Core Cooling, Heat Sink, Pressurized Thermal Shock, Integrity (Containment) and Inventory (RCS).

EVENT: Assessment of an EVENT commences when recognition is made that one or more of the initiating conditions associated with the event exist. Implicit in this definition is the need for timely assessment within 15 minutes.

EXCLUSION AREA BOUNDARY (EAB): That area surrounding the reactor, in which the reactor licenses has the authority to determine all activities including exclusion or removal of personnel and property from the area. For purposes of Emergency Action Levels, based on radiological field measurements and dose assessments, and for design calculations, the Site Boundary shall be defined as the EAB.

EXPLOSION: Rapid, violent, unconfined combustion, or a catastrophic failure of pressurized or electrical equipment that imparts energy of sufficient force to potentially damage permanent structures or equipment.

EXTORTION: An attempt to cause an action at the site by threat or force.

FAULTED: (Steam Generator) Existence of secondary side leakage (e.g., steam or feed line break) that results in an uncontrolled decrease in steam generator pressure or the steam generator being completely depressurized.

FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical components do not constitute a fire. Observation of flame is preferred but is NOT required if large quantities of smoke and/or heat are observed.

FLAMMABLE GAS: Combustible gases at concentrations > than the LOWER EXPLOSIVE LIMIT (LEL).

HOSTAGE: A person(s) held as leverage against the site to ensure that demands will be met by the site.

HOSTILE ACTION: An act toward a nuclear plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land or water; using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. **HOSTILE ACTION** should NOT be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant.

HOSTILE FORCE: One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

IMMINENT: Within two hours.

INEFFECTIVE: When the specified restoration action(s) does not result in a reduction in the level of severity of the RED or ORANGE PATH condition within 15 minutes from identification of the CSF Status Tree RED or ORANGE PATH.

INITIATING CONDITIONS: Plant Parameters, radiation monitor readings or personnel observations that identify an Event for purposes of Emergency Plan Classification.

INTRUSION/INTRUDER: Suspected hostile individual present in the protected area without authorization.

ISFSI: Independent Spent Fuel Storage Installation.

ODCM: Offsite Dose Calculation Manual is a supporting document to the Tech Specs. that contain Rad Effluent Controls, Environs Monitoring controls, and methodology for calculating routine gaseous and liquid effluent offsite doses and monitor alarm/trip setpoints.

ORANGE PATH: Monitoring of one or more CSFs by FR-0 which indicates that the CSF(s) is under severe challenge; prompt operator action is required.

PROJECTILE: An object ejected, thrown or launched towards a plant structure resulting in damage sufficient to cause concern regarding the integrity of the affected structure or the operability or reliability of safety equipment contained therein. The source of the projectile may be onsite or offsite.

PROTECTED AREA: The area encompassed by the security fence and to which access is controlled.

RCS: The RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary and secondary isolation valves.

RED PATH: Monitoring of one or more CSFs by FR-0 which indicates that the CSF(s) is under extreme challenge; prompt operator action is required.

RUPTURED: (Steam Generator) Existence of primary to secondary leakage of a magnitude greater than the capacity one charging pump.

SABOTAGE: Deliberate damage, misalignment, or misoperation of plant equipment with the intent to render the equipment inoperable.

SECURITY CONDITION: Any Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.

SIGNIFICANT TRANSIENT: An UNPLANNED event involving one or more of the following: (1) An automatic turbine runback >15% thermal reactor power; (2) Electrical load rejection >25% full electrical load; (3) Reactor Trip; (4) Safety Injection System Activation; (5) Thermal Power Oscillations $\geq 10\%$.

STRIKE ACTION: A work stoppage within the PROTECTED AREA by a body of workers to enforce compliance with demands made on TVA. The STRIKE ACTION must threaten to interrupt normal plant operations.

TOXIC GAS: A gas that is dangerous to life or limb by reason of inhalation or skin contact (e.g., chlorine, CO₂, etc.)

UNPLANNED: An event or action that is not the expected result of normal operations, testing or maintenance. Events that result in corrective or mitigative actions being taken in accordance with abnormal or emergency procedures are UNPLANNED.

UNPLANNED RELEASE: A release of radioactivity is UNPLANNED if the release has not been authorized by a Discharge Permit (DP). Implicit in this definition are unintentional releases, unmonitored releases, or planned releases that exceed a condition specified on the DP, (e.g., alarm setpoints, minimum dilution flow, minimum release times, maximum release rates, and/or discharge of incorrect tank).

VALID: An indication, report or condition is considered to be VALID when it is conclusively verified by (1) an instrument channel check, or (2) indication on related or redundant indicators, or (3) by direct observation by plant personnel. Implicit in this definition is the need for timely assessment within 15 minutes.

VISIBLE DAMAGE: Damage to equipment that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes deformation due to heat or impact, denting, penetration, rupture, cracking, or paint blistering. Surface blemishes (e.g., paint chipping, scratches, etc.) should NOT be included as visible damage.

VITAL AREA: Any area within the PROTECTED AREA which contains equipment, systems, devices, or material which the failure, destruction, or release of, could directly or indirectly endanger the public health and safety by exposure to radiation.

1.1 Fuel Clad Barrier

1. Critical Safety Function Status	
LOSS	Potential LOSS
Core Cooling Red (FR-C.1)	Core Cooling Orange (FR-C.2) <u>OR</u> Heat Sink RED (FR-H.1) and RHR Shutdown Cooling not in service

- OR -

2. Primary Coolant Activity Level	
LOSS	Potential LOSS
RCS sample activity is greater than 300 uCi/gm dose equivalent I131	Not Applicable

- OR -

3. Incore Thermocouple Hi Quad Average	
LOSS	Potential LOSS
Greater than 1200 °F on XI-94-101 or 102 (EXOSENSOR)	Greater than or equal to 700 °F on XI-94-101 or 102 (EXOSENSOR)

- OR -

4. Reactor Vessel Water Level	
LOSS	Potential LOSS
Not Applicable	VALID RVLIS level <42% on LI-68-368 or LI-68-371 with no RCP running

- OR -

5. Containment Radiation Monitor	
LOSS	Potential LOSS
VALID reading of greater than: 2.8E+01 Rem/hr on RM-90-271A or -272A <u>OR</u> 2.9E+01 Rem/hr on RM-90-273A or -274A (see instruction note 4)	Not Applicable

- OR -

6. SED Judgment	
Any condition that, in the judgment of the SM or SED, indicates loss or potential loss of the Fuel Clad Barrier comparable to the conditions listed above.	

1.2 RCS Barrier

1. Critical Safety Function Status	
LOSS	Potential LOSS
Not Applicable	Pressurized Thermal Shock Red (FR-P.1) <u>OR</u> Heat Sink RED (FR-H.1) and RHR Shutdown Cooling not in service

- OR -

2. RCS Leakage / LOCA	
LOSS	Potential LOSS
RCS leak results in subcooling <40 °F as indicated on XI-94-101 or 102 (EXOSENSOR)	Non Isolatable RCS leak exceeding the capacity of one charging pump in the normal charging alignment <u>OR</u> RCS leakage results in entry into E-1

- OR -

3. Steam Generator Tube Rupture	
LOSS	Potential LOSS
SGTR that results in a Safety Injection actuation <u>OR</u> Entry into E-3	Not Applicable

- OR -

4.	
LOSS	Potential LOSS
VALID RVLIS level <42% on LI-68-368 or LI-68-371 with no RCP running	Not Applicable

- OR -

5. SED Judgment	
Any condition that, in the judgment of the SM or SED, indicates loss or potential loss of the RCS Barrier comparable to the conditions listed above.	

1.3 Containment Barrier

1. Critical Safety Function Status

LOSS	Potential LOSS
Not Applicable	Containment Red (FR-Z.1) OR Actions of FR-C.1 (Red Path) are INEFFECTIVE (i.e.: core TCs trending up)

- OR -

2. Containment Pressure / Hydrogen

LOSS	Potential LOSS
Rapid unexplained pressure decrease following initial increase on PDI-30-44 or 45 OR Containment pressure or sump level not increasing on LI-63-178 and 179 with a LOCA in progress	Containment Hydrogen increases to >4% by volume on H2I-43-200 or 210 OR Pressure >2.8 PSIG (Phase B) with < one full train of containment spray

- OR -

3. Containment Isolation Status

LOSS	Potential LOSS
Containment isolation, when required is incomplete and a release path to the environment exists.	Not Applicable

- OR -

4. Containment Bypass

LOSS	Potential LOSS
RUPTURED S/G that is also faulted outside containment (E2 and E3) OR >4 hour secondary side release outside containment from a S/G with a S/G tube leak >T/S limits (AOP R.01 App A)	Unexpected VALID increase in area or ventilation RAD monitors adjacent to containment (with LOCA in progress).

- OR -

5. Significant Radiation in Containment

LOSS	Potential LOSS
Not Applicable	VALID reading of greater than: 3.6E+02 Rem/hr on RM-90-271A and 272 OR 2.8E+02 Rem/hr on RM-90-273A and 274 (see instruction note 4)

- OR -

6. SED Judgment

Any condition that, in the judgment of the SM or SED, indicates loss or potential loss of the Containment Barrier comparable to the conditions listed above.

INSTRUCTIONS

Note: A condition is considered to be **MET** if, in the judgment of the SED, the condition will be **MET IMMEDIATELY** (i.e.: with two hours). The classification shall be made as soon as this determination is made.

1. In the matrix to the left, REVIEW the initiating conditions in all three barrier columns and circle the conditions that are MET.
2. In each of the three barrier columns, IDENTIFY if any Loss or Potential Loss **INITIATING CONDITIONS** have been MET.
3. COMPARE the number of barrier Losses and Potential losses to the criteria below and make the appropriate declaration.
4. Containment Radiation Monitors are temperature sensitive and can be affected by temperature-induced currents. These monitors should be used for trending only until containment temperature has been stable for approximately 5 minutes after a Steam Line Break or LOCA.

Note: MONITOR the respective status tree criteria if a CSF is listed as an **INITIATING CONDITION**.

Emergency Class Criteria

General Emergency

LOSS of any two barriers and Potential LOSS of third barrier

Site Area Emergency

LOSS or Potential LOSS of any two barriers

Alert

Any LOSS or Potential LOSS of Fuel Clad barrier

OR

Any LOSS or Potential LOSS of RCS barrier

Unusual Event

LOSS or Potential LOSS of Containment barrier

FISSION PRODUCT BARRIER MATRIX

(Modes 1-4)

1

- 1.1 Fuel Clad Barrier
- 1.2 RCS Barrier
- 1.3 Containment Barrier

2	SYSTEM DEGRADATION	
	2.1 Loss of Instrumentation	2.5 RCS Unidentified Leakage
	2.2 Loss of Communication	2.6 RCS Identified Leakage
	2.3 Failure of Reactor Protection	2.7 Uncontrolled Cool Down
	2.4 Fuel Clad Degradation	2.8 Turbine Failure
		2.9 Safety Limit

LOSS OF POWER

3

- 3.1 Loss of AC (Power Ops)
- 3.2 Loss of AC (Shutdown)
- 3.3 Loss of DC

HAZARDS and SED JUDGMENT

4

- 4.1 Fire Table 4-1
- 4.2 Explosion Table 4-2
- 4.3 Flammable Gas Figure 4-A
- 4.4 Toxic Gas or Smoke Figure 4-B
- 4.5 Control Room Evacuation
- 4.6 Security
- 4.7 SED Judgment

DESTRUCTIVE PHENOMENON

5

- 5.1 Earthquake 5.5 River Level Low
- 5.2 Tornado 5.6 Watercraft Crash
- 5.3 Aircraft/Projectile Table 5-1
- 5.4 River Level High Figure 5-A

SHUTDOWN SYSTEM DEGRADATION

6

- 6.1 Loss of Shutdown Systems
- 6.2 Loss of Shutdown Capability
- 6.3 Loss of RCS Inventory

RADIOLOGICAL EFFLUENTS

7

- 7.1 Gaseous Effluent Table 7-1
- 7.2 Liquid Effluent Table 7-2
- 7.3 Radiation Levels Figure 7-A
- 7.4 Fuel Handling
- 7.5 Spent Fuel Storage

Definitions and Abbreviations:

BOMB: An explosive device. (See EXPLOSION)

CIVIL DISTURBANCE: A group of twenty (20) or more persons within the EAB violently protesting onsite operations or activities at the site.

CONFINEMENT BOUNDARY: Spent Fuel Storage Cask CONFINEMENT BOUNDARY consists of MPC shell, bottom baseplate, MPC lid (including the vent and drain port cover plates), MPC closure ring, and associated welds.

CRITICAL-SAFETY FUNCTION (CSFs): A plant safety function required to prevent significant release of core radioactivity to the environment. There are six CSFs; Subcriticality, Core Cooling, Heat Sink, Pressurized Thermal Shock, Integrity (Containment) and Inventory (RCS).

EVENT: Assessment of an EVENT commences when recognition is made that one or more of the initiating conditions associated with the event exist. Implicit in this definition is the need for timely assessment within 15 minutes.

EXCLUSION AREA BOUNDARY (EAB): That area surrounding the reactor, in which the reactor licenses has the authority to determine all activities including exclusion or removal of personnel and property from the area. For purposes of Emergency Action Levels, based on radiological field measurements and dose assessments, and for design calculations, the Site Boundary shall be defined as the EAB.

EXPLOSION: Rapid, violent, unconfined combustion, or a catastrophic failure of pressurized or electrical equipment that imparts energy of sufficient force to potentially damage permanent structures or equipment.

EXTORTION: An attempt to cause an action at the site by threat or force.

FAULTED: (Steam Generator) Existence of secondary side leakage (e.g., steam or feed line break) that results in an uncontrolled decrease in steam generator pressure or the steam generator being completely depressurized.

FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical components do not constitute a fire. Observation of flame is preferred but is NOT required if large quantities of smoke and/or heat are observed.

FLAMMABLE GAS: Combustible gases at concentrations > than the LOWER EXPLOSIVE LIMIT (LEL).

HOSTAGE: A person(s) held as leverage against the site to ensure that demands will be met by the site.

HOSTILE ACTION: An act toward a nuclear plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land or water; using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. **HOSTILE ACTION** should NOT be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant.

HOSTILE FORCE: One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

IMMINENT: Within two hours.

INEFFECTIVE: When the specified restoration action(s) does not result in a reduction in the level of severity of the RED or ORANGE PATH condition within 15 minutes from identification of the CSF Status Tree RED or ORANGE PATH.

INITIATING CONDITIONS: Plant Parameters, radiation monitor readings or personnel observations that identify an Event for purposes of Emergency Plan Classification.

INTRUSION/INTRUDER: Suspected hostile individual present in the protected area without authorization.

ISFSI: Independent Spent Fuel Storage Installation.

ODCM: Offsite Dose Calculation Manual is a supporting document to the Tech Specs. that contain Rad Effluent Controls, Environs Monitoring controls, and methodology for calculating routine gaseous and liquid effluent offsite doses and monitor alarm/trip setpoints.

ORANGE PATH: Monitoring of one or more CSFs by FR-0 which indicates that the CSF(s) is under severe challenge; prompt operator action is required.

PROJECTILE: An object ejected, thrown or launched towards a plant structure resulting in damage sufficient to cause concern regarding the integrity of the affected structure or the operability or reliability of safety equipment contained therein. The source of the projectile may be onsite or offsite.

PROTECTED AREA: The area encompassed by the security fence and to which access is controlled.

RCS: The RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary and secondary isolation valves.

RED PATH: Monitoring of one or more CSFs by FR-0 which indicates that the CSF(s) is under extreme challenge; prompt operator action is required.

RUPTURED: (Steam Generator) Existence of primary to secondary leakage of a magnitude greater than the capacity one charging pump.

SABOTAGE: Deliberate damage, misalignment, or misoperation of plant equipment with the intent to render the equipment inoperable.

SECURITY CONDITION: Any Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.

SIGNIFICANT TRANSIENT: An UNPLANNED event involving one or more of the following: (1) An automatic turbine runback >15% thermal reactor power; (2) Electrical load rejection >25% full electrical load; (3) Reactor Trip; (4) Safety Injection System Activation; (5) Thermal Power Oscillations \geq 10%.

STRIKE ACTION: A work stoppage within the PROTECTED AREA by a body of workers to enforce compliance with demands made on TVA. The STRIKE ACTION must threaten to interrupt normal plant operations.

TOXIC GAS: A gas that is dangerous to life or limb by reason of inhalation or skin contact (e.g., chlorine, CO₂, etc.)

UNPLANNED: An event or action that is not the expected result of normal operations, testing or maintenance. Events that result in corrective or mitigative actions being taken in accordance with abnormal or emergency procedures are UNPLANNED.

UNPLANNED RELEASE: A release of radioactivity is UNPLANNED if the release has not been authorized by a Discharge Permit (DP). Implicit in this definition are unintentional releases, unmonitored releases, or planned releases that exceed a condition specified on the DP, (e.g., alarm setpoints, minimum dilution flow, minimum release times, maximum release rates, and/or discharge of incorrect tank).

VALID: An indication, report or condition is considered to be VALID when it is conclusively verified by (1) an instrument channel check, or (2) indication on related or redundant indicators, or (3) by direct observation by plant personnel. Implicit in this definition is the need for timely assessment within 15 minutes.

VISIBLE DAMAGE: Damage to equipment that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes deformation due to heat or impact, denting, penetration, rupture, cracking, or paint blistering. Surface blemishes (e.g., paint chipping, scratches, etc.) should NOT be included as visible damage.

VITAL AREA: Any area within the PROTECTED AREA which contains equipment, systems, devices, or material which the failure, destruction, or release of, could directly or indirectly endanger the public health and safety by exposure to radiation.

2.1 Loss of Instrumentation		2.2 Loss of Communications	
Mode	Initiating / Condition	Mode	Initiating / Condition
	Refer to "Fission Product Barrier Matrix" (Section 1) and "Radiological Effluents" (Section 7) and Continue in This Column.		Not Applicable.
1, 2, 3, 4	On either unit an inability to monitor a SIGNIFICANT TRANSIENT in progress (1 and 2 and 3 and 4): 1. Loss of > 75% of MCR annunciator windows AND the annunciator printer AND the annunciator CRT in the horseshoe OR > 75% of safety system indications. 2. Loss of ICS. 3. Inability to directly monitor any of the following CSFs: Subcriticality PTS Core Cooling Containment Heat Sink Inventory 4. SIGNIFICANT TRANSIENT in progress.		Not Applicable.
1, 2, 3, 4	On either unit an UNPLANNED loss of >75% of the MCR annunciators and annunciator printer or > 75% of safety system indications for > 15 minutes with a SIGNIFICANT TRANSIENT in progress or ICS unavailable. <i>(1 and 2 and 3):</i> 1. UNPLANNED loss of >75% of both channels of MCR annunciator windows AND the annunciator printer AND the annunciator CRT in the horseshoe for > 15 minutes OR > 75% of safety system indicators for > 15 minutes. 2. SM/SED judgment that increased surveillance is required (> shift complement) to safely operate the unit. 3. (a or b) a. SIGNIFICANT TRANSIENT in progress. OR b. Loss of ICS.		Not Applicable.
1, 2, 3, 4	On either unit an UNPLANNED loss of > 75% of the MCR annunciators or > 75% of safety system indications for > 15 minutes and ICS available. <i>(1 and 2 and 3):</i> 1. UNPLANNED loss of >75% of both channels of MCR annunciator windows AND the annunciator printer AND the annunciator CRT in the horseshoe for > 15 minutes OR > 75% of safety system indicators for > 15 minutes. 2. SM/SED judgment that increased surveillance is required (> shift complement) to safely operate the unit. 3. The ICS is capable of displaying requested data.		Significant Loss of Communications (1 or 2) 1. UNPLANNED loss of all in-plant communication capabilities listed below (a and b and c): a. UNPLANNED loss of EPABX phones. b. UNPLANNED loss of all sound powered phones. c. UNPLANNED loss of all in-plant radio frequencies. OR 2. UNPLANNED loss of all offsite communication capabilities listed below: (a and b and c and d and e and f) a. UNPLANNED loss of all EPABX phones b. UNPLANNED loss of all offsite radio frequencies c. UNPLANNED loss of all OPX (Microwave) system d. UNPLANNED loss of all 1-FB-Bell lines e. UNPLANNED loss of all NRC ENS and HPN lines f. UNPLANNED loss of all satellite phones

GENERAL

SITE AREA

ALERT

NOUVEAU

2.3 Failure of Rx Protection Sys		2.4 Fuel Clad Degredation	
Mode	Initiating / Condition	Mode	Initiating / Condition
1	<p>Reactor power > 5% and not decreasing after VALID trip signals and loss of core cooling capability. (1 and 2):</p> <ol style="list-style-type: none"> 1. FR-S.1 entered and immediate operator actions did not result in a reactor power of ≤ 5% and decreasing. 2. (a or b) <ol style="list-style-type: none"> a. CSF status tree indicates Core Cooling Red (FR-C.1). <p style="text-align: center;"><u>OR</u></p> <ol style="list-style-type: none"> b. CSF status tree indicates Heat Sink Red (FR-H.1) 	GENERAL	<p>Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.</p>
1	<p>Reactor power > 5% and not decreasing after VALID auto and manual trip signals.</p> <p><i>NOTE: Although a mode change may occur before classification this event will still be classified and declared as SAE.</i></p>	SITE AREA	<p>Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.</p>
1, 2	<p>Reactor power > 5% and not decreasing after VALID auto trip signal but a manual trip from the Control Room is successful. (1 and 2)</p> <ol style="list-style-type: none"> 1. Reactor power > 5% and not decreasing following auto trip signal. 2. Manual trip in the Main Control Room successfully reduces reactor power ≤ 5%. <p><i>NOTE: Although a mode change will occur, this event will still be classified and declared as an ALERT.</i></p>	ALERT	<p>Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.</p>
	<p>Refer to "Fission Product Barrier Matrix" (Section 1).</p>	NOU E	<p>Reactor coolant system specific activity exceeds LCO (Refer to SQN Tech. Spec. 3.4.8):</p> <ol style="list-style-type: none"> 1. Radiochemistry analysis indicates (a or b): <ol style="list-style-type: none"> a. Dose equivalent Iodine (I-131) > 0.35 μCi/gm for > 48 hours or in excess of T/S Figure 3.4-1 with Tave ≥ 500 °F. <p style="text-align: center;"><u>OR</u></p> <ol style="list-style-type: none"> b. Specific activity > 100/É μCi/gm with Tave ≥ 500 °F.

2.5 RCS Unidentified Leakage		2.6 RCS Identified Leakage	
Mode	Initiating / Condition	Mode	Initiating / Condition
	Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.		Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.
	Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.		Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.
	Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.		Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.
1, 2, 3, 4	<p>RCS unidentified or pressure boundary leakage > 10 GPM.</p> <p>1. Unidentified or pressure boundary leakage (as defined by Tech. Spec.) > 10 GPM as indicated by (a or b):</p> <p>a. SI-OPS-068-137.0 results or RCS Flow Balance Calculation</p> <p style="text-align: center;"><u>OR</u></p> <p>b. With RCS temperature and PZR level stable, the VCT level on LI-62-129 or LI-62-130 is dropping at a rate > 10 GPM.</p> <p>Refer to "Shutdown Systems Degradation" (Section 6.3).</p>	1, 2, 3, 4	<p>RCS Identified leakage > 25 GPM.</p> <p>1. Identified RCS leakage (as defined by Tech. Spec.) > 25 GPM as indicated by (a or b or c):</p> <p>a. SI-OPS-068-137.0 results or RCS Flow Balance Calculation</p> <p style="text-align: center;"><u>OR</u></p> <p>b. Level rise in excess of 25 GPM into PRT, RCDT or CVCS holdup tank (Refer to TI-28).</p> <p style="text-align: center;"><u>OR</u></p> <p>c. RCS leakage through a steam generator to the secondary system (primary to secondary leakage).</p> <p>Refer to "Shutdown Systems Degradation" (Section 6.3).</p>

GENERAL

SITE AREA

ALERT

NOUE

2.7 Uncontrolled Cooldown		2.8 Turbine Failure													
Mode	Initiating / Condition	Mode	Initiating / Condition												
	Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.		Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.												
	Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.		Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.												
	Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.	1, 2, 3	<p>Turbine failure has generated projectiles that cause visible damage to any area containing safety related equipment.</p> <p>1. Turbine generated PROJECTILES have resulted in VISIBLE DAMAGE to any of the following areas:</p> <table border="0"> <tr> <td>Control Building</td> <td>Diesel Generator Bldg.</td> </tr> <tr> <td>Auxiliary Building</td> <td>RWST</td> </tr> <tr> <td>Unit #1 Containment</td> <td>Intake Pumping Station</td> </tr> <tr> <td>Unit #2 Containment</td> <td>Common Sta. Serv. Xfmr's</td> </tr> <tr> <td>ERCW Pumping Station</td> <td>Condensate Storage Tanks</td> </tr> <tr> <td>Add Equipment Bldgs.</td> <td></td> </tr> </table>	Control Building	Diesel Generator Bldg.	Auxiliary Building	RWST	Unit #1 Containment	Intake Pumping Station	Unit #2 Containment	Common Sta. Serv. Xfmr's	ERCW Pumping Station	Condensate Storage Tanks	Add Equipment Bldgs.	
Control Building	Diesel Generator Bldg.														
Auxiliary Building	RWST														
Unit #1 Containment	Intake Pumping Station														
Unit #2 Containment	Common Sta. Serv. Xfmr's														
ERCW Pumping Station	Condensate Storage Tanks														
Add Equipment Bldgs.															
1, 2, 3	<p>UNPLANNED rapid depressurization of the main steam system resulting in a rapid RCS cooldown and safety injection initiation. (1 and 2):</p> <p>1. Rapid depressurization of any or all steam generators or the main steam system to < 600 psig on PI-1-2A, 2B or 9A, 9B or 20A, 20B or 27A, 27B.</p> <p>2. Safety injection has initiated or is required.</p>	1, 2, 3	<p>Turbine failure results in casing penetration or main generator seal damage.</p> <p>1. Turbine failure which results in penetration of the turbine casing or damage to main generator seals.</p> <p>Refer to "Hazards and SED Judgment" (Section 4.3)</p>												

GENERAL

SITE AREA

ALERT

NOU

2.9 Safety Limit	
Mode	Initiating / Condition
	Not Applicable.
	Not Applicable.
	Not Applicable.
1, 2, 3, 4	Safety Limits have been exceeded. (1 or 2): 1. The combination of thermal power, RCS temperature and RCS pressure > safety limit indicated by SQN Tech. Spec. Figure 2.1-1 "Reactor Core Safety Limit". <u>OR</u> 2. RCS/Pressurizer pressure exceeds safety limit (> 2735 psig).

G
E
N
E
R
A
L

S
I
T
E
A
R
E
A

A
L
E
R
T

N
O
U
E

This Page Intentionally Blank

- 1** **FISSION PRODUCT BARRIER MATRIX** (Modes 1-4)
- 1.1 Fuel Clad Barrier
 - 1.2 RCS Barrier
 - 1.3 Containment Barrier
- 2** **SYSTEM DEGRADATION**
- 2.1 Loss of Instrumentation
 - 2.2 Loss of Communication
 - 2.3 Failure of Reactor Protection
 - 2.4 Fuel Clad Degradation
 - 2.5 RCS Unidentified Leakage
 - 2.6 RCS Identified Leakage
 - 2.7 Uncontrolled Cool Down
 - 2.8 Turbine Failure
 - 2.9 Safety Limit

3	<p>LOSS OF POWER</p> <ul style="list-style-type: none"> 3.1 Loss of AC (Power Ops) 3.2 Loss of AC (Shutdown) 3.3 Loss of DC
----------	---------------------------------------------------------------------------------------------------------------------------------------------------------------------

- 4** **HAZARDS and SED JUDGMENT**
- 4.1 Fire Table 4-1
 - 4.2 Explosion Table 4-2
 - 4.3 Flammable Gas Figure 4-A
 - 4.4 Toxic Gas or Smoke Figure 4-B
 - 4.5 Control Room Evacuation
 - 4.6 Security
 - 4.7 SED Judgment

- 5** **DESTRUCTIVE PHENOMENON**
- 5.1 Earthquake
 - 5.2 Tornado
 - 5.3 Aircraft/Projectile
 - 5.4 River Level High
 - 5.5 River Level Low
 - 5.6 Watercraft Crash

- 6** **SHUTDOWN SYSTEM DEGRADATION**
- 6.1 Loss of Shutdown Systems
 - 6.2 Loss of Shutdown Capability
 - 6.3 Loss of RCS Inventory

- 7** **RADIOLOGICAL EFFLUENTS**
- 7.1 Gaseous Effluent Table 7-1
 - 7.2 Liquid Effluent Table 7-2
 - 7.3 Radiation Levels Figure 7-A
 - 7.4 Fuel Handling
 - 7.5 Spent Fuel Storage

Definitions and Abbreviations:

BOMB: An explosive device. (See EXPLOSION)

CIVIL DISTURBANCE: A group of twenty (20) or more persons within the EAB violently protesting onsite operations or activities at the site.

CONFINEMENT BOUNDARY: Spent Fuel Storage Cask CONFINEMENT BOUNDARY consists of MPC shell, bottom baseplate, MPC lid (including the vent and drain port cover plates), MPC closure ring, and associated welds.

CRITICAL-SAFETY FUNCTION (CSFs): A plant safety function required to prevent significant release of core radioactivity to the environment. There are six CSFs: Subcriticality, Core Cooling, Heat Sink, Pressurized Thermal Shock, Integrity (Containment) and Inventory (RCS).

EVENT: Assessment of an EVENT commences when recognition is made that one or more of the initiating conditions associated with the event exist. Implicit in this definition is the need for timely assessment within 15 minutes.

EXCLUSION AREA BOUNDARY (EAB): That area surrounding the reactor, in which the reactor licenses has the authority to determine all activities including exclusion or removal of personnel and property from the area. For purposes of Emergency Action Levels, based on radiological field measurements and dose assessments, and for design calculations, the Site Boundary shall be defined as the EAB.

EXPLOSION: Rapid, violent, unconfined combustion, or a catastrophic failure of pressurized or electrical equipment that imparts energy of sufficient force to potentially damage permanent structures or equipment.

EXTORTION: An attempt to cause an action at the site by threat or force.

FAULTED: (Steam Generator) Existence of secondary side leakage (e.g., steam or feed line break) that results in an uncontrolled decrease in steam generator pressure or the steam generator being completely depressurized.

FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical components do not constitute a fire. Observation of flame is preferred but is NOT required if large quantities of smoke and/or heat are observed.

FLAMMABLE GAS: Combustible gases at concentrations > than the LOWER EXPLOSIVE LIMIT (LEL).

HOSTAGE: A person(s) held as leverage against the site to ensure that demands will be met by the site.

HOSTILE ACTION: An act toward a nuclear plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land or water; using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. **HOSTILE ACTION** should NOT be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant.

HOSTILE FORCE: One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

IMMINENT: Within two hours.

INEFFECTIVE: When the specified restoration action(s) does not result in a reduction in the level of severity of the RED or ORANGE PATH condition within 15 minutes from identification of the CSF Status Tree RED or ORANGE PATH.

INITIATING CONDITIONS: Plant Parameters, radiation monitor readings or personnel observations that identify an Event for purposes of Emergency Plan Classification.

INTRUSION/INTRUDER: Suspected hostile individual present in the protected area without authorization.

ISFSI: Independent Spent Fuel Storage Installation

ODCM: Offsite Dose Calculation Manual is a supporting document to the Tech Specs. that contain Rad Effluent Controls, Environs Monitoring controls, and methodology for calculating routine gaseous and liquid effluent offsite doses and monitor alarm/trip setpoints.

ORANGE PATH: Monitoring of one or more CSFs by FR-0 which indicates that the CSF(s) is under severe challenge; prompt operator action is required.

PROJECTILE: An object ejected, thrown or launched towards a plant structure resulting in damage sufficient to cause concern regarding the integrity of the affected structure or the operability or reliability of safety equipment contained therein. The source of the projectile may be onsite or offsite.

PROTECTED AREA: The area encompassed by the security fence and to which access is controlled.

RCS: The RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary and secondary isolation valves.

RED PATH: Monitoring of one or more CSFs by FR-0 which indicates that the CSF(s) is under extreme challenge; prompt operator action is required.

RUPTURED: (Steam Generator) Existence of primary to secondary leakage of a magnitude greater than the capacity one charging pump.

SABOTAGE: Deliberate damage, misalignment, or misoperation of plant equipment with the intent to render the equipment inoperable.

SECURITY CONDITION: Any Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.

SIGNIFICANT TRANSIENT: An UNPLANNED event involving one or more of the following: (1) An automatic turbine runback >15% thermal reactor power; (2) Electrical load rejection >25% full electrical load; (3) Reactor Trip; (4) Safety Injection System Activation; (5) Thermal Power Oscillations $\geq 10\%$.

STRIKE ACTION: A work stoppage within the PROTECTED AREA by a body of workers to enforce compliance with demands made on TVA. The STRIKE ACTION must threaten to interrupt normal plant operations.

TOXIC GAS: A gas that is dangerous to life or limb by reason of inhalation or skin contact (e.g., chlorine, CO₂, etc.)

UNPLANNED: An event or action that is not the expected result of normal operations, testing or maintenance. Events that result in corrective or mitigative actions being taken in accordance with abnormal or emergency procedures are UNPLANNED.

UNPLANNED RELEASE: A release of radioactivity is UNPLANNED if the release has not been authorized by a Discharge Permit (DP). Implicit in this definition are unintentional releases, unmonitored releases, or planned releases that exceed a condition specified on the DP, (e.g., alarm setpoints, minimum dilution flow, minimum release times, maximum release rates, and/or discharge of incorrect tank).

VALID: An indication, report or condition is considered to be VALID when it is conclusively verified by (1) an instrument channel check, or (2) indication on related or redundant indicators, or (3) by direct observation by plant personnel. Implicit in this definition is the need for timely assessment within 15 minutes.

VISIBLE DAMAGE: Damage to equipment that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes deformation due to heat or impact, denting, penetration, rupture, cracking, or paint blistering. Surface blemishes (e.g., paint chipping, scratches, etc.) should NOT be included as visible damage.

VITAL AREA: Any area within the PROTECTED AREA which contains equipment, systems, devices, or material which the failure, destruction, or release of, could directly or indirectly endanger the public health and safety by exposure to radiation.

3.1 Loss of AC (Power Ops)		3.2 Loss of AC (Shutdown)	
Mode	Initiating / Condition	Mode	Initiating / Condition
1, 2, 3, 4	<p>Prolonged loss of all offsite and all onsite AC power to either unit. (1 and 2):</p> <ol style="list-style-type: none"> 1. Both unit related 6.9 KV shutdown boards de-energized for > 15 minutes. 2. (a or b) <ol style="list-style-type: none"> a. Core Cooling Status Tree Red or Orange Path. <p style="text-align: center;"><u>OR</u></p> <ol style="list-style-type: none"> b. Restoration of either a 6.9 KV shutdown board or a 6.9 KV unit board is not likely within 4 hours of the loss. 		Not Applicable.
1, 2, 3, 4	<p>Loss of all offsite and all onsite AC power to either unit for > 15 Minutes.</p> <ol style="list-style-type: none"> 1. Both unit related 6.9 KV shutdown boards de-energized for > 15 minutes. 		Not Applicable.
1, 2, 3, 4	<p>Loss of offsite power to either unit with degraded onsite AC power for > 15 minutes. ([1a and 1b] or 2):</p> <ol style="list-style-type: none"> 1a. All four (4) 6.9KV unit boards de-energized for > 15 minutes. 1b. One (1) unit related 6.9 KV shutdown board de-energized for > 15 minutes. <p style="text-align: center;"><u>OR</u></p> <ol style="list-style-type: none"> 2. Any AC power condition lasting > 15 minutes where a single additional failure will result in a unit blackout. 	5, 6	<p>UNPLANNED loss of all offsite and all onsite AC power to either unit for > 15 minutes.</p> <ol style="list-style-type: none"> 1. Both unit related 6.9KV shutdown boards de-energized for > 15 minutes. <p><i>Also Refer to "Loss of Shutdown Systems" (6.1) and continue in this column.</i></p>
1, 2, 3, 4	<p>Loss of offsite power to either unit for > 15 minutes. (1 and 2):</p> <ol style="list-style-type: none"> 1. All four (4) 6.9KV unit boards de-energized for > 15 minutes. 2. Both unit related 6.9KV shutdown boards are energized. 	5, 6	<p>UNPLANNED loss of all offsite power to either unit for > 15 minutes. (1 and 2):</p> <ol style="list-style-type: none"> 1. All four (4) 6.9KV unit boards de-energized for > 15 minutes. 2. One (1) unit related 6.9KV shutdown board de-energized for > 15 minutes.

GENERAL

SITE AREA

ALERT

NOUVE

DEFUELED

DEFUELED

3.3 Loss of DC Power	
Mode	Initiating / Condition
	Refer to "Fission Product Barrier Matrix" (Section 1) and "Loss of Communication" (2.2) and Continue in This Column.
1, 2, 3, 4	<p>Loss of all vital DC power for > 15 minutes.</p> <p>1. Voltage < 105 V DC on 125V DC vital battery board buses I <u>and</u> II <u>and</u> III <u>and</u> IV for > 15 minutes.</p> <p>Also Refer to "Fission Product Barrier Matrix" (Section 1), "Loss of Communication" (2.2) and, "Loss of Instrumentation" (2.1) and Continue in This Column.</p>
	Refer to "Fission Product Barrier Matrix" (Section 1), "Loss of Communication" (2.2), and "Loss of Instrumentation" (2.1).
5, 6	<p>UNPLANNED loss of a required train of DC power for > 15 minutes: (1 or 2).</p> <p>1. Voltage < 105 V DC on 125V dc vital battery board buses I and III for > 15 minutes.</p> <p style="text-align: center;"><u>OR</u></p> <p>2. Voltage < 105 V DC on 125V dc vital battery board busses II and IV for > 15 minutes.</p>

GENERAL

SITE AREA

ALERT

NOUE

- 1**
- FISSION PRODUCT BARRIER MATRIX (Modes 1-4)**
- 1.1 Fuel Clad Barrier
 - 1.2 RCS Barrier
 - 1.3 Containment Barrier

- 2**
- SYSTEM DEGRADATION**
- 2.1 Loss of Instrumentation
 - 2.2 Loss of Communication
 - 2.3 Failure of Reactor Protection
 - 2.4 Fuel Clad Degradation
 - 2.5 RCS Unidentified Leakage
 - 2.6 RCS Identified Leakage
 - 2.7 Uncontrolled Cool Down
 - 2.8 Turbine Failure
 - 2.9 Safety Limit

- 3**
- LOSS OF POWER**
- 3.1 Loss of AC (Power Ops)
 - 3.2 Loss of AC (Shutdown)
 - 3.3 Loss of DC

4	HAZARDS and SED JUDGMENT	
	4.1 Fire	Table 4-1
	4.2 Explosion	Table 4-2
	4.3 Flammable Gas	Figure 4-A
	4.4 Toxic Gas or Smoke	Figure 4-B
	4.5 Control Room Evacuation	
	4.6 Security	
	4.7 SED Judgment	

- 5**
- DESTRUCTIVE PHENOMENON**
- 5.1 Earthquake
 - 5.2 Tornado
 - 5.3 Aircraft/Projectile
 - 5.4 River Level High
 - 5.5 River Level Low
 - 5.6 Watercraft Crash

- 6**
- SHUTDOWN SYSTEM DEGRADATION**
- 6.1 Loss of Shutdown Systems
 - 6.2 Loss of Shutdown Capability
 - 6.3 Loss of RCS Inventory

- 7**
- RADIOLOGICAL EFFLUENTS**
- 7.1 Gaseous Effluent
 - 7.2 Liquid Effluent
 - 7.3 Radiation Levels
 - 7.4 Fuel Handling
 - 7.5 Spent Fuel Storage

Definitions and Abbreviations:

BOMB: An explosive device. (See EXPLOSION)

CIVIL DISTURBANCE: A group of twenty (20) or more persons within the EAB violently protesting onsite operations or activities at the site.

CONFINEMENT BOUNDARY: Spent Fuel Storage Cask CONFINEMENT BOUNDARY consists of MPC shell, bottom baseplate, MPC lid (including the vent and drain port cover plates), MPC closure ring, and associated welds.

CRITICAL-SAFETY FUNCTION (CSFs): A plant safety function required to prevent significant release of core radioactivity to the environment. There are six CSFs: Subcriticality, Core Cooling, Heat Sink, Pressurized Thermal Shock, Integrity (Containment) and Inventory (RCS).

EVENT: Assessment of an EVENT commences when recognition is made that one or more of the initiating conditions associated with the event exist. Implicit in this definition is the need for timely assessment within 15 minutes.

EXCLUSION AREA BOUNDARY (EAB): That area surrounding the reactor, in which the reactor licenses has the authority to determine all activities including exclusion or removal of personnel and property from the area. For purposes of Emergency Action Levels, based on radiological field measurements and dose assessments, and for design calculations, the Site Boundary shall be defined as the EAB.

EXPLOSION: Rapid, violent, unconfined combustion, or a catastrophic failure of pressurized or electrical equipment that imparts energy of sufficient force to potentially damage permanent structures or equipment.

EXTORTION: An attempt to cause an action at the site by threat or force.

FAULTED: (Steam Generator) Existence of secondary side leakage (e.g., steam or feed line break) that results in an uncontrolled decrease in steam generator pressure or the steam generator being completely depressurized.

FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical components do not constitute a fire. Observation of flame is preferred but is NOT required if large quantities of smoke and/or heat are observed.

FLAMMABLE GAS: Combustible gases at concentrations > than the LOWER EXPLOSIVE LIMIT (LEL).

HOSTAGE: A person(s) held as leverage against the site to ensure that demands will be met by the site.

HOSTILE ACTION: An act toward a nuclear plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land or water; using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. **HOSTILE ACTION** should NOT be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant.

HOSTILE FORCE: One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

IMMINENT: Within two hours.

INEFFECTIVE: When the specified restoration action(s) does not result in a reduction in the level of severity of the RED or ORANGE PATH condition within 15 minutes from identification of the CSF Status Tree RED or ORANGE PATH.

INITIATING CONDITIONS: Plant Parameters, radiation monitor readings or personnel observations that identify an Event for purposes of Emergency Plan Classification.

INTRUSION/INTRUDER: Suspected hostile individual present in the protected area without authorization.

ISFSI: Independent Spent Fuel Storage Installation.

ODCM: Offsite Dose Calculation Manual is a supporting document to the Tech Specs. that contain Rad Effluent Controls, Environs Monitoring controls, and methodology for calculating routine gaseous and liquid effluent offsite doses and monitor alarm/trip setpoints.

ORANGE PATH: Monitoring of one or more CSFs by FR-0 which indicates that the CSF(s) is under severe challenge; prompt operator action is required.

PROJECTILE: An object ejected, thrown or launched towards a plant structure resulting in damage sufficient to cause concern regarding the integrity of the affected structure or the operability or reliability of safety equipment contained therein. The source of the projectile may be onsite or offsite.

PROTECTED AREA: The area encompassed by the security fence and to which access is controlled.

RCS: The RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary and secondary isolation valves.

RED PATH: Monitoring of one or more CSFs by FR-0 which indicates that the CSF(s) is under extreme challenge; prompt operator action is required.

RUPTURED: (Steam Generator) Existence of primary to secondary leakage of a magnitude greater than the capacity one charging pump.

SABOTAGE: Deliberate damage, misalignment, or misoperation of plant equipment with the intent to render the equipment inoperable.

SECURITY CONDITION: Any Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.

SIGNIFICANT TRANSIENT: An UNPLANNED event involving one or more of the following: (1) An automatic turbine runback >15% thermal reactor power; (2) Electrical load rejection >25% full electrical load; (3) Reactor Trip; (4) Safety Injection System Activation; (5) Thermal Power Oscillations $\geq 10\%$.

STRIKE ACTION: A work stoppage within the PROTECTED AREA by a body of workers to enforce compliance with demands made on TVA. The STRIKE ACTION must threaten to interrupt normal plant operations.

TOXIC GAS: A gas that is dangerous to life or limb by reason of inhalation or skin contact (e.g., chlorine, CO₂, etc.)

UNPLANNED: An event or action that is not the expected result of normal operations, testing or maintenance. Events that result in corrective or mitigative actions being taken in accordance with abnormal or emergency procedures are UNPLANNED.

UNPLANNED RELEASE: A release of radioactivity is UNPLANNED if the release has not been authorized by a Discharge Permit (DP). Implicit in this definition are unintentional releases, unmonitored releases, or planned releases that exceed a condition specified on the DP, (e.g., alarm setpoints, minimum dilution flow, minimum release times, maximum release rates, and/or discharge of incorrect tank).

VALID: An indication, report or condition is considered to be VALID when it is conclusively verified by (1) an instrument channel check, or (2) indication on related or redundant indicators, or (3) by direct observation by plant personnel. Implicit in this definition is the need for timely assessment within 15 minutes.

VISIBLE DAMAGE: Damage to equipment that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes deformation due to heat or impact, denting, penetration, rupture, cracking, or paint blistering. Surface blemishes (e.g., paint chipping, scratches, etc.) should NOT be included as visible damage.

VITAL AREA: Any area within the PROTECTED AREA which contains equipment, systems, devices, or material which the failure, destruction, or release of, could directly or indirectly endanger the public health and safety by exposure to radiation.

4.1 Fire		4.2 Explosion	
Mode	Initiating / Condition	Mode	Initiating / Condition
	Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.		Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.
	Refer to "Control Room Evacuation," (4.5) and Fission Product Barrier Matrix" (Section 1) and Continue in This Column.		Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.
A L L	<p>FIRE in any of the areas listed in Table 4-1 that is affecting safety related equipment required to establish or maintain safe shutdown. (1 and 2):</p> <ol style="list-style-type: none"> 1. FIRE in any of the areas listed in Table 4-1. 2. (a or b) <ol style="list-style-type: none"> a. VISIBLE DAMAGE to permanent structure or safety related equipment in the specified area is observed due to the FIRE. <p style="text-align: center;"><u>OR</u></p> <ol style="list-style-type: none"> b. Control room indication of degraded safety system or component response due to the FIRE. 	A L L	<p>EXPLOSION in any of the areas listed in Table 4-1 that is affecting safety related equipment required to establish or maintain safe shutdown. (1 and 2):</p> <ol style="list-style-type: none"> 1. EXPLOSION in any of the areas listed in Table 4-1. 2. (a or b) <ol style="list-style-type: none"> a. VISIBLE DAMAGE to permanent structures or to safety related equipment in the specified area is due to the EXPLOSION. <p style="text-align: center;"><u>OR</u></p> <ol style="list-style-type: none"> b. Control room indication of degraded safety system or component response due to the EXPLOSION. <p>Refer to "Security" (Section 4.6).</p>
	<p>FIRE within the PROTECTED AREA (Figure 4-A) threatening any of the areas listed in Table 4-1 that is not extinguished within 15 minutes from the time of control room notification or verification of control room alarm.</p>		<p>UNPLANNED EXPLOSION within the PROTECTED AREA (Figure 4-A) resulting in VISIBLE DAMAGE to any permanent structure <u>or</u> equipment.</p> <p>Refer to "Security" (Section 4.6).</p>

GENERAL

SITE AREA

ALERT

NOUE

4.3 Flammable Gas		4.4 Toxic Gas or Smoke	
Mode	Initiating / Condition	Mode	Initiating / Condition
	Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.		Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.
	Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.		Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.
ALL	<p>UNPLANNED release of FLAMMABLE GAS within a facility structure containing safety related equipment or associated with safe operation of the plant.</p> <p>1. Plant personnel report the average of three (3) readings taken in an ~10 ft. Triangular Area is > 25% Lower Explosive Limit, as indicated on the monitoring instrument within any building listed in Table 4-2.</p> <p>Refer to the MSDS for the LEL.</p>	ALL	<p>Release of TOXIC GAS or smoke within a facility structure which prohibits safe operation of systems required to establish or maintain Cold S/D. (1 and 2 and 3):</p> <p>1. Plant personnel report TOXIC GAS or smoke within any building listed in Table 4-2.</p> <p>2. (a or b)</p> <p>a. Plant personnel report severe adverse health reactions due to TOXIC GAS or smoke (i.e., burning eyes, nose, throat, dizziness).</p> <p style="text-align: center;">OR</p> <p>b. Sampling indication > Permissible Exposure Limit (PEL).</p> <p>3. Plant personnel unable to perform actions to establish and maintain Cold Shutdown while utilizing appropriate personnel protection equipment.</p> <p>Refer to the MSDS for the PEL.</p>
	<p>A. UNPLANNED release of FLAMMABLE GAS within the EXCLUSION AREA BOUNDARY that may affect normal operations.</p> <p>1. Plant personnel report the average of three readings taken in an ~10 ft. Triangular Area is > 25% of the Lower Explosive Limit, as indicated on the monitoring instrument within the EXCLUSION AREA BOUNDARY (Figure 4-B).</p> <p style="text-align: center;">OR</p> <p>B. Confirmed report by Local, County, or State officials that a large offsite FLAMMABLE GAS release has occurred within one (1) mile of the site (Figure 4-B) with potential to enter the EXCLUSION AREA BOUNDARY (Figure 4-B) in concentrations > 25% of Lower Explosive Limit.</p> <p>Refer to the MSDS for the LEL.</p>		<p>A. Safe operations impeded due to access restrictions caused by TOXIC GAS or smoke concentrations within a facility structure listed in Table 4-2.</p> <p style="text-align: center;">OR</p> <p>B. Confirmed report by Local, County, or State officials that an offsite TOXIC GAS release has occurred within one (1) mile of the site (Figure 4-B) with potential to enter the EXCLUSION AREA BOUNDARY (Figure 4-B) in concentrations > the Permissible Exposure Limit (PEL) causing a site evacuation.</p> <p>Refer to the MSDS for the PEL.</p>

GENERAL SITE AREA

ALERT

NOUVE

4.5 Control Room Evacuation		4.6 Security	
Mode	Initiating / Condition	Mode	Initiating / Condition
	Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.	A L L	HOSTILE ACTION Resulting in Loss of Physical Control of the Facility: (1 or 2) 1. A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain CRITICAL SAFETY FUNCTIONS . 2. A HOSTILE ACTION has caused failure of Spent Fuel Cooling Systems and IMMINENT fuel damage is likely for a freshly off-loaded reactor core in pool.
A L L	Evacuation of the control room has been initiated and control of all necessary equipment has not been established within 15 minutes of staffing the auxiliary control room. (1 and 2): 1. AOP-C.04 "Shutdown from Aux Control Room" entered. 2. Control has not been established within 15 minutes of staffing the auxiliary control room and completing transfer of switches on panels L11A and L11B to the AUX position.	A L L	HOSTILE ACTION within the PROTECTED AREA . A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor. Refer to Figure 4-A for a drawing of PROTECTED AREA .
A L L	Evacuation of the Control Room is Required. 1. AOP-C.04 "Shutdown from Aux Control Room" has been entered.	A L L	HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat: (1 or 2) 1. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervisor. 2. A validated notification from NRC of an airliner attack threat within 30 minutes of the site. Note: The Owner Controlled Area is defined by the Physical Security Plan. Refer to Figure 4-A for a drawing of PROTECTED AREA .
	Not Applicable.	A L L	Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant (1 or 2 or 3) 1. A SECURITY CONDITION that does NOT involve a HOSTILE ACTION as reported by the Security Shift Supervisor. 2. A credible SQN security threat notification. 3. A validated notification from NRC providing information of an aircraft threat.

**G
E
N
E
R
A
L**

**S
I
T
E
A
R
E
A**

**A
L
E
R
T**

**N
O
U
E**

4.7 SED Judgment

Mode	Initiating / Condition
ALL	Events are in process <u>or</u> have occurred which involve Actual <u>or</u> Imminent Substantial Core Degradation <u>or</u> Melting With Potential for Loss of Containment Integrity <u>or</u> HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Plume Protective Action Guidelines Exposure Levels outside the EXCLUSION AREA BOUNDARY , refer to Figure 4-B.
ALL	Events are in process <u>or</u> have occurred which involve Actual <u>or</u> Likely Major Failures of Plant Functions needed for the Protection of the Public <u>or</u> HOSTILE ACTION that result in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevents effective access to equipment needed for the Protection of the Public. Any releases are not expected to result in Exposure Levels which Exceed EPA Plume Protective Action Guidelines Exposure Levels beyond the EXCLUSION AREA BOUNDARY , Refer to Figure 4-B.
ALL	Events are in process <u>or</u> have occurred which involve Actual <u>or</u> Potential Substantial Degradation of the Level of Safety of the Plant or a Security Event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION . Any releases are expected to be limited to small fractions of the EPA Plume Protective Action Guidelines Exposure Levels.
ALL	Events are in process <u>or</u> have occurred which indicate a Potential Degradation of the Level of Safety of the Plant or indicate a Security Threat to facility protection has been initiated. No releases of Radioactive Material requiring Offsite Response <u>or</u> Monitoring are expected unless further degradation of Safety Systems occurs.

GENERAL

SITE AREA

ALERT

NOUE

**TABLE 4-1
PLANT AREAS ASSOCIATED WITH
FIRE AND EXPLOSION EALS**

Unit #1 Containment
Unit #2 Containment
Auxiliary Building
Diesel Generator Building
Intake Pumping Station
ERCW Pumping Station
Control Building
Additional Equipment Buildings
CSST's
RWST
Condensate Storage Tanks

**TABLE 4-2
PLANT AREAS ASSOCIATED WITH
TOXIC OR FLAMMABLE GAS OR SMOKE EALS**

Unit #1 Containment
Unit #2 Containment
Auxiliary Building
Turbine Building
Diesel Generator Building
Intake Pumping Station
ERCW Pumping Station
Control Building
Additional Equipment Buildings
CDWE Building

Figure 4-A
Protected Area

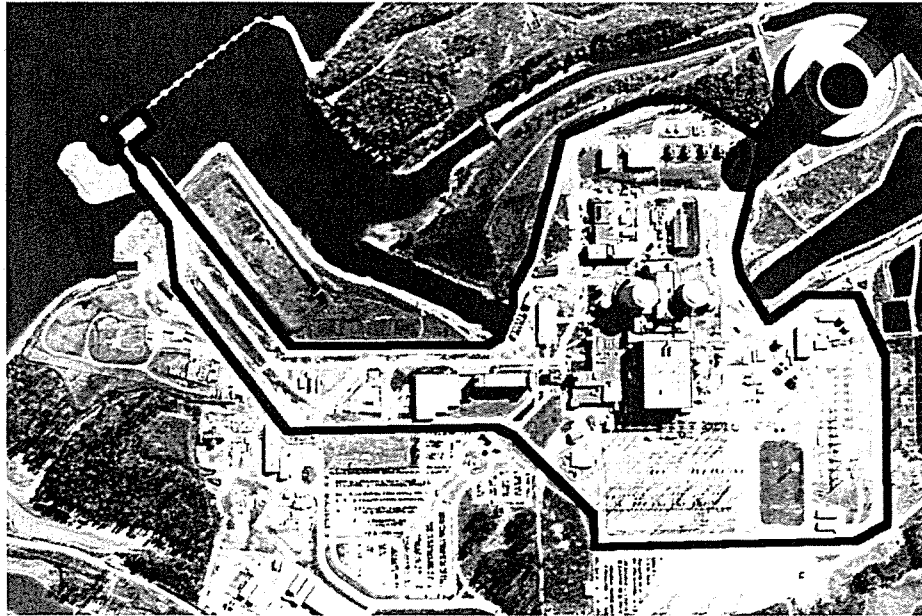
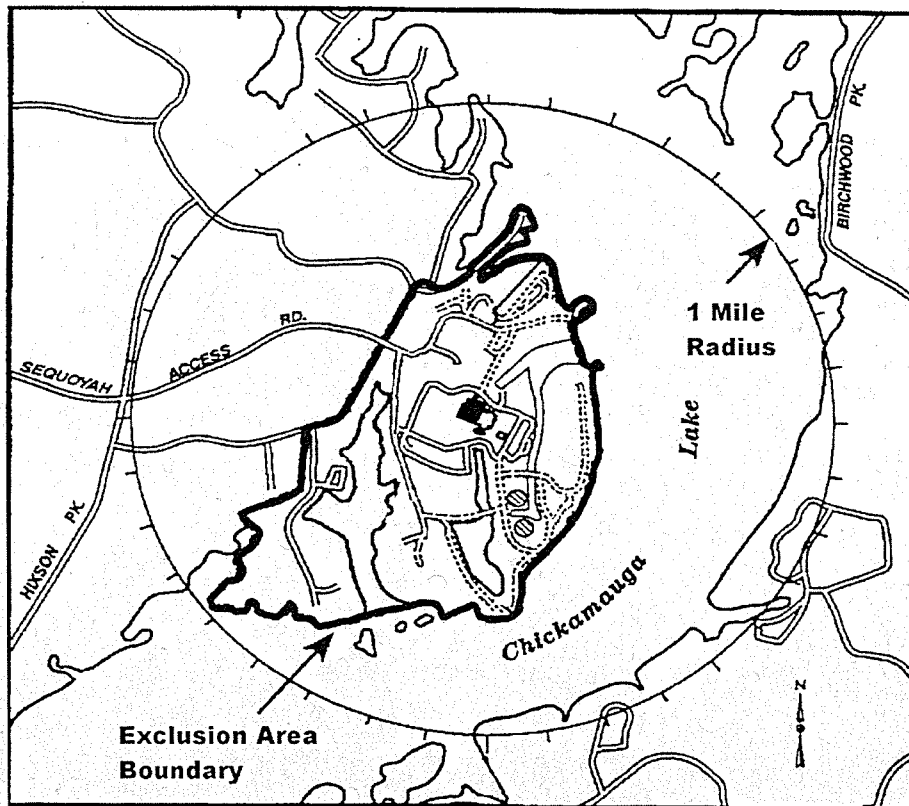


Figure 4-B
Exclusion Area Boundary



1

FISSION PRODUCT BARRIER MATRIX (Modes 1-4)

- 1.1 Fuel Clad Barrier
- 1.2 RCS Barrier
- 1.3 Containment Barrier

2

SYSTEM DEGRADATION

- 2.1 Loss of Instrumentation
- 2.2 Loss of Communication
- 2.3 Failure of Reactor Protection
- 2.4 Fuel Clad Degradation
- 2.5 RCS Unidentified Leakage
- 2.6 RCS Identified Leakage
- 2.7 Uncontrolled Cool Down
- 2.8 Turbine Failure
- 2.9 Safety Limit

3

LOSS OF POWER

- 3.1 Loss of AC (Power Ops)
- 3.2 Loss of AC (Shutdown)
- 3.3 Loss of DC

4

HAZARDS and SED JUDGMENT

- 4.1 Fire Table 4-1
- 4.2 Explosion Table 4-2
- 4.3 Flammable Gas Figure 4-A
- 4.4 Toxic Gas or Smoke Figure 4-B
- 4.5 Control Room Evacuation
- 4.6 Security
- 4.7 SED Judgment

5

DESTRUCTIVE PHENOMENON

- 5.1 Earthquake
- 5.2 Tornado
- 5.3 Aircraft/Projectile
- 5.4 River Level High
- 5.5 River Level Low
- 5.6 Watercraft Crash
- Table 5-1
- Figure 5-A

6

SHUTDOWN SYSTEM DEGRADATION

- 6.1 Loss of Shutdown Systems
- 6.2 Loss of Shutdown Capability
- 6.3 Loss of RCS Inventory

7

RADIOLOGICAL EFFLUENTS

- 7.1 Gaseous Effluent Table 7-1
- 7.2 Liquid Effluent Table 7-2
- 7.3 Radiation Levels Figure 7-A
- 7.4 Fuel Handling
- 7.5 Spent Fuel Storage

Definitions and Abbreviations:

BOMB: An explosive device. (See EXPLOSION)

CIVIL DISTURBANCE: A group of twenty (20) or more persons within the EAB violently protesting onsite operations or activities at the site.

CONFINEMENT BOUNDARY: Spent Fuel Storage Cask CONFINEMENT BOUNDARY consists of MPC shell, bottom baseplate, MPC lid (including the vent and drain port cover plates), MPC closure ring, and associated welds.

CRITICAL-SAFETY FUNCTION (CSFs): A plant safety function required to prevent significant release of core radioactivity to the environment. There are six CSFs; Subcriticality, Core Cooling, Heat Sink, Pressurized Thermal Shock, Integrity (Containment) and Inventory (RCS).

EVENT: Assessment of an EVENT commences when recognition is made that one or more of the initiating conditions associated with the event exist. Implicit in this definition is the need for timely assessment within 15 minutes.

EXCLUSION AREA BOUNDARY (EAB): That area surrounding the reactor, in which the reactor licenses has the authority to determine all activities including exclusion or removal of personnel and property from the area. For purposes of Emergency Action Levels, based on radiological field measurements and dose assessments, and for design calculations, the Site Boundary shall be defined as the EAB.

EXPLOSION: Rapid, violent, unconfined combustion, or a catastrophic failure of pressurized or electrical equipment that imparts energy of sufficient force to potentially damage permanent structures or equipment.

EXTORTION: An attempt to cause an action at the site by threat or force.

FAULTED: (Steam Generator) Existence of secondary side leakage (e.g., steam or feed line break) that results in an uncontrolled decrease in steam generator pressure or the steam generator being completely depressurized.

FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical components do not constitute a fire. Observation of flame is preferred but is NOT required if large quantities of smoke and/or heat are observed.

FLAMMABLE GAS: Combustible gases at concentrations > than the LOWER EXPLOSIVE LIMIT (LEL).

HOSTAGE: A person(s) held as leverage against the site to ensure that demands will be met by the site.

HOSTILE ACTION: An act toward a nuclear plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land or water; using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. **HOSTILE ACTION** should NOT be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant.

HOSTILE FORCE: One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

IMMINENT: Within two hours.

INEFFECTIVE: When the specified restoration action(s) does not result in a reduction in the level of severity of the RED or ORANGE PATH condition within 15 minutes from identification of the CSF Status Tree RED or ORANGE PATH.

INITIATING CONDITIONS: Plant Parameters, radiation monitor readings or personnel observations that identify an Event for purposes of Emergency Plan Classification.

INTRUSION/INTRUDER: Suspected hostile individual present in the protected area without authorization.

ISFSI: Independent Spent Fuel Storage Installation

ODCM: Offsite Dose Calculation Manual is a supporting document to the Tech Specs. that contain Rad Effluent Controls, Environs Monitoring controls, and methodology for calculating routine gaseous and liquid effluent offsite doses and monitor alarm/trip setpoints.

ORANGE PATH: Monitoring of one or more CSFs by FR-0 which indicates that the CSF(s) is under severe challenge; prompt operator action is required.

PROJECTILE: An object ejected, thrown or launched towards a plant structure resulting in damage sufficient to cause concern regarding the integrity of the affected structure or the operability or reliability of safety equipment contained therein. The source of the projectile may be onsite or offsite.

PROTECTED AREA: The area encompassed by the security fence and to which access is controlled.

RCS: The RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary and secondary isolation valves.

RED PATH: Monitoring of one or more CSFs by FR-0 which indicates that the CSF(s) is under extreme challenge; prompt operator action is required.

RUPTURED: (Steam Generator) Existence of primary to secondary leakage of a magnitude greater than the capacity one charging pump.

SABOTAGE: Deliberate damage, misalignment, or misoperation of plant equipment with the intent to render the equipment inoperable.

SECURITY CONDITION: Any Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.

SIGNIFICANT TRANSIENT: An UNPLANNED event involving one or more of the following: (1) An automatic turbine runback >15% thermal reactor power; (2) Electrical load rejection >25% full electrical load; (3) Reactor Trip; (4) Safety Injection System Activation; (5) Thermal Power Oscillations $\geq 10\%$.

STRIKE ACTION: A work stoppage within the PROTECTED AREA by a body of workers to enforce compliance with demands made on TVA. The STRIKE ACTION must threaten to interrupt normal plant operations.

TOXIC GAS: A gas that is dangerous to life or limb by reason of inhalation or skin contact (e.g., chlorine, CO₂, etc.)

UNPLANNED: An event or action that is not the expected result of normal operations, testing or maintenance. Events that result in corrective or mitigative actions being taken in accordance with abnormal or emergency procedures are UNPLANNED.

UNPLANNED RELEASE: A release of radioactivity is UNPLANNED if the release has not been authorized by a Discharge Permit (DP). Implicit in this definition are unintentional releases, unmonitored releases, or planned releases that exceed a condition specified on the DP, (e.g., alarm setpoints, minimum dilution flow, minimum release times, maximum release rates, and/or discharge of incorrect tank).

VALID: An indication, report or condition is considered to be VALID when it is conclusively verified by (1) an instrument channel check, or (2) indication on related or redundant indicators, or (3) by direct observation by plant personnel. Implicit in this definition is the need for timely assessment within 15 minutes.

VISIBLE DAMAGE: Damage to equipment that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes deformation due to heat or impact, denting, penetration, rupture, cracking, or paint blistering. Surface blemishes (e.g., paint chipping, scratches, etc.) should NOT be included as visible damage.

VITAL AREA: Any area within the PROTECTED AREA which contains equipment, systems, devices, or material which the failure, destruction, or release of, could directly or indirectly endanger the public health and safety by exposure to radiation.

5.1 Earthquake		5.2 Tornado	
Mode	Initiating / Condition	Mode	Initiating / Condition
	Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.		Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.
	Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.		Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.
A L L	<p>Earthquake detected by site seismic instrumentation. (1 and 2):</p> <ol style="list-style-type: none"> 1. Panel XA-55-15B alarm window 30 (E-2) plus window 22 (D-1) activated. 2. (a or b) <ol style="list-style-type: none"> a. Ground motion sensed by plant personnel. <p style="text-align: center;"><u>OR</u></p> <ol style="list-style-type: none"> b. National Earthquake Information Center at 1-(303) 273-8500 can confirm the event. 	A L L	<p>Tornado or high winds strikes any structure listed in Table 5-1 and results in VISIBLE DAMAGE. (1 and 2):</p> <ol style="list-style-type: none"> 1. Tornado or high winds (sustained >80 m.p.h. > one minute on the plant computer) strikes any structure listed in Table 5-1. 2. (a or b) <ol style="list-style-type: none"> a. Confirmed report of any VISIBLE DAMAGE. <p style="text-align: center;"><u>OR</u></p> <ol style="list-style-type: none"> b. Control room indications of degraded safety system or component response due to event. <p><i>Note: If site met data is unavailable, National Weather Service Morristown 1-(423)-586-8400, can provide additional information if needed.</i></p>
A L L	<p>Earthquake detected by site seismic instruments. (1 and 2):</p> <ol style="list-style-type: none"> 1. Panel XA-55-15B alarm window 22 (D-1) activated. 2. (a or b) <ol style="list-style-type: none"> a. Ground motion sensed by plant personnel. <p style="text-align: center;"><u>OR</u></p> <ol style="list-style-type: none"> b. National Earthquake Information Center at 1-(303) 273-8500 can confirm the event. 	A L L	<p>Tornado within the EXCLUSION AREA BOUNDARY.</p> <ol style="list-style-type: none"> 1. Plant personnel report a tornado has been sighted within the EXCLUSION AREA BOUNDARY (Figure 5-A)

5.3 Aircraft/Projectile Impact		5.4 River Level High	
Mode	Initiating / Condition	Mode	Initiating / Condition
	Refer to the "Fission Product Barrier Matrix" (Section 1).		Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.
	Refer to the "Fission Product Barrier Matrix" (Section 1).		Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.
A L L	<p>Aircraft or PROJECTILE impacts (strikes) any plant structure listed in Table 5-1 resulting in VISIBLE DAMAGE. (1 and 2):</p> <ol style="list-style-type: none"> 1. Plant personnel report aircraft or PROJECTILE has impacted any structure listed in Table 5-1. 2. (a or b) <ol style="list-style-type: none"> a. Confirmed report of VISIBLE DAMAGE. <p style="text-align: center;">OR</p> <ol style="list-style-type: none"> b. Control Room indications of degraded safety system or component response due to the event within any structure listed in Table 5-1. 	A L L	River reservoir level is at Stage II Flood Warning as reported by River Operations.
A L L	<p>Aircraft crash or projectile impact (strikes) within the EXCLUSION AREA BOUNDARY.</p> <ol style="list-style-type: none"> 1. Plant personnel report aircraft crash or PROJECTILE impact within the EXCLUSION AREA BOUNDARY (Figure 5-A). 	A L L	River reservoir level is at Stage I Flood Warning as reported by River Operations.

GENERAL

SITE AREA

ALERT

NOUE

5.5 River Level Low		5.6 WaterCraft Crash	
Mode	Initiating / Condition	Mode	Initiating / Condition
	Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.		Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.
	Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.		Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.
A L L	River reservoir level is < 670 Feet as reported by River Operations.		Refer to "Fission Product Barrier Matrix" (Section 1) and Continue in This Column.
A L L	River reservoir level is < 674 Feet as reported by River Operations.	A L L	<p>Watercraft strikes the ERCW pumping station resulting in a reduction of Essential Raw Cooling Water (ERCW). (1 and 2):</p> <ol style="list-style-type: none"> 1. Plant personnel report a watercraft has struck the ERCW pumping station. 2. (a or b) <ol style="list-style-type: none"> a. ERCW supply header pressure Train A 1(2)-PI-67-493A is < 15 psig. <p style="text-align: center;"><u>OR</u></p> <ol style="list-style-type: none"> b. ERCW supply header pressure Train B 1(2)-PI-67-488A is < 15 psig.

**G
E
N
E
R
A
L**

**S
I
T
E
A
R
E
A**

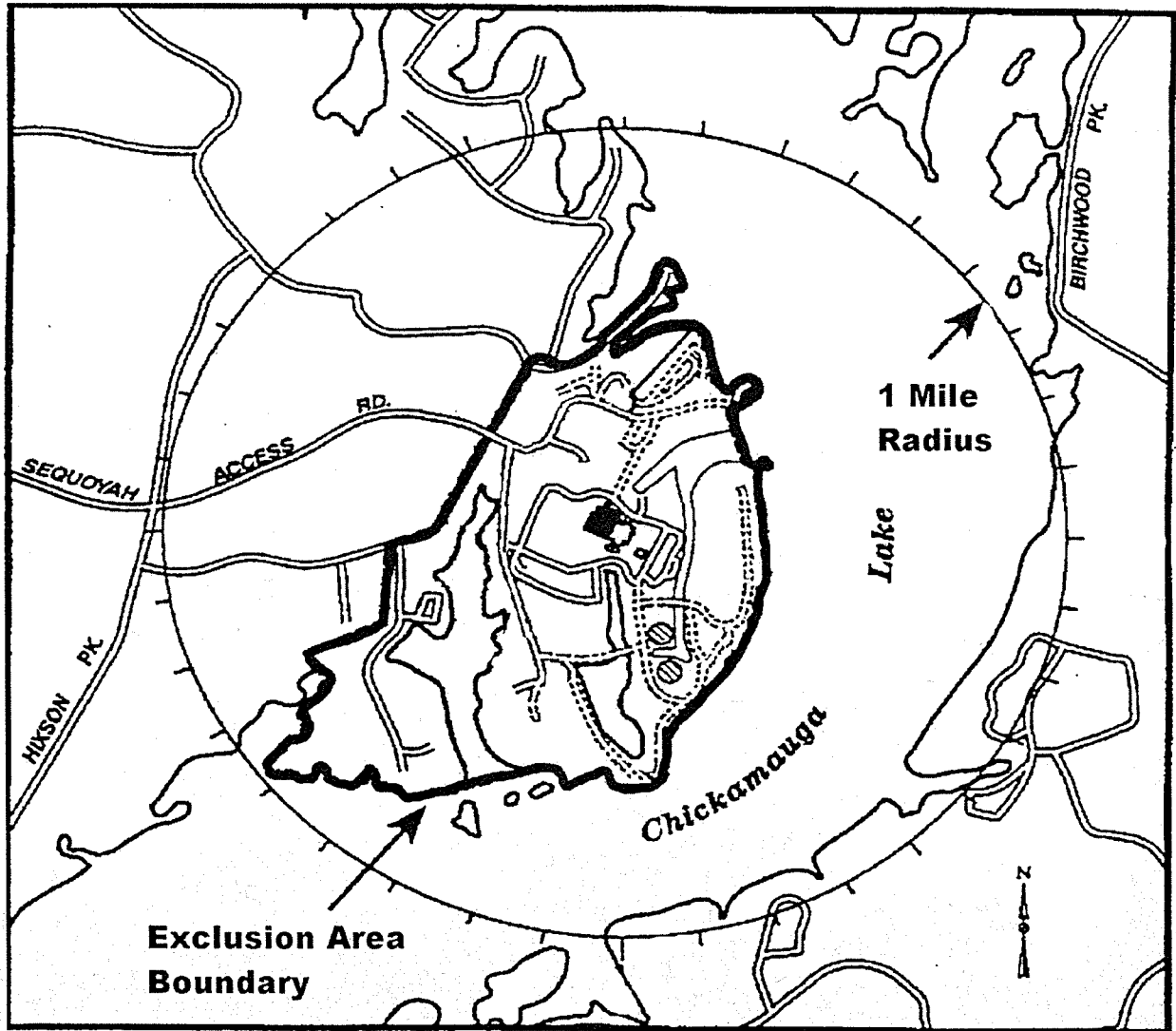
**A
L
E
R
T**

**N
O
U
E**

TABLE 5-1
 Plant Structure Associated With
 Tornado/High Wind and Aircraft EALs

Unit #1 Containment	Auxiliary Building
Turbine Building	RWST
Unit #2 Containment	Diesel Generator Bldg.
CDWE Building	Condensate Storage Tanks
Control Building	ERCW Pumping Station
Additional Equipment Bldgs	Intake Pumping Station
Common Station Service Transformer's	

Figure 5-A
 SEQUOYAH EXCLUSION AREA BOUNDARY



1 **FISSION PRODUCT BARRIER MATRIX** (Modes 1-4)
 1.1 Fuel Clad Barrier
 1.2 RCS Barrier
 1.3 Containment Barrier

2 **SYSTEM DEGRADATION**
 2.1 Loss of Instrumentation 2.5 RCS Unidentified Leakage
 2.2 Loss of Communication 2.6 RCS Identified Leakage
 2.3 Failure of Reactor Protection 2.7 Uncontrolled Cool Down
 2.4 Fuel Clad Degradation 2.8 Turbine Failure
 2.9 Safety Limit

3 **LOSS OF POWER**
 3.1 Loss of AC (Power Ops)
 3.2 Loss of AC (Shutdown)
 3.3 Loss of DC

4 **HAZARDS and SED JUDGMENT**
 4.1 Fire Table 4-1
 4.2 Explosion Table 4-2
 4.3 Flammable Gas Figure 4-A
 4.4 Toxic Gas or Smoke Figure 4-B
 4.5 Control Room Evacuation
 4.6 Security
 4.7 SED Judgment

5 **DESTRUCTIVE PHENOMENON**
 5.1 Earthquake 5.5 River Level Low
 5.2 Tornado 5.6 Watercraft Crash
 5.3 Aircraft/Projectile Table 5-1
 5.4 River Level High Figure 5-A

6 **SHUTDOWN SYSTEM DEGRADATION**
 6.1 Loss of Shutdown Systems
 6.2 Loss of Shutdown Capability
 6.3 Loss of RCS Inventory

7 **RADIOLOGICAL EFFLUENTS**
 7.1 Gaseous Effluent Table 7-1
 7.2 Liquid Effluent Table 7-2
 7.3 Radiation Levels Figure 7-A
 7.4 Fuel Handling
 7.5 Spent Fuel Storage

Definitions and Abbreviations:

BOMB: An explosive device. (See EXPLOSION)

CIVIL DISTURBANCE: A group of twenty (20) or more persons within the EAB violently protesting onsite operations or activities at the site.

CONFINEMENT BOUNDARY: Spent Fuel Storage Cask CONFINEMENT BOUNDARY consists of MPC shell, bottom baseplate, MPC lid (including the vent and drain port cover plates), MPC closure ring, and associated welds.

CRITICAL-SAFETY FUNCTION (CSFs): A plant safety function required to prevent significant release of core radioactivity to the environment. There are six CSFs; Subcriticality, Core Cooling, Heat Sink, Pressurized Thermal Shock, Integrity (Containment) and Inventory (RCS).

EVENT: Assessment of an EVENT commences when recognition is made that one or more of the initiating conditions associated with the event exist. Implicit in this definition is the need for timely assessment within 15 minutes.

EXCLUSION AREA BOUNDARY (EAB): That area surrounding the reactor, in which the reactor licensee has the authority to determine all activities including exclusion or removal of personnel and property from the area. For purposes of Emergency Action Levels, based on radiological field measurements and dose assessments, and for design calculations, the Site Boundary shall be defined as the EAB.

EXPLOSION: Rapid, violent, unconfined combustion, or a catastrophic failure of pressurized or electrical equipment that imparts energy of sufficient force to potentially damage permanent structures or equipment.

EXTORTION: An attempt to cause an action at the site by threat or force.

FAULTED: (Steam Generator) Existence of secondary side leakage (e.g., steam or feed line break) that results in an uncontrolled decrease in steam generator pressure or the steam generator being completely depressurized.

FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical components do not constitute a fire. Observation of flame is preferred but is NOT required if large quantities of smoke and/or heat are observed.

FLAMMABLE GAS: Combustible gases at concentrations > than the LOWER EXPLOSIVE LIMIT (LEL).

HOSTAGE: A person(s) held as leverage against the site to ensure that demands will be met by the site.

HOSTILE ACTION: An act toward a nuclear plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land or water; using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. **HOSTILE ACTION** should NOT be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant.

HOSTILE FORCE: One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

IMMINENT: Within two hours.

INEFFECTIVE: When the specified restoration action(s) does not result in a reduction in the level of severity of the RED or ORANGE PATH condition within 15 minutes from identification of the CSF Status Tree RED or ORANGE PATH.

INITIATING CONDITIONS: Plant Parameters, radiation monitor readings or personnel observations that identify an Event for purposes of Emergency Plan Classification.

INTRUSION/INTRUDER: Suspected hostile individual present in the protected area without authorization.

ISFSI: Independent Spent Fuel Storage Installation.

ODCM: Offsite Dose Calculation Manual is a supporting document to the Tech Specs. that contain Rad Effluent Controls, Environs Monitoring controls, and methodology for calculating routine gaseous and liquid effluent offsite doses and monitor alarm/trip setpoints.

ORANGE PATH: Monitoring of one or more CSFs by FR-0 which indicates that the CSF(s) is under severe challenge; prompt operator action is required.

PROJECTILE: An object ejected, thrown or launched towards a plant structure resulting in damage sufficient to cause concern regarding the integrity of the affected structure or the operability or reliability of safety equipment contained therein. The source of the projectile may be onsite or offsite.

PROTECTED AREA: The area encompassed by the security fence and to which access is controlled.

RCS: The RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary and secondary isolation valves.

RED PATH: Monitoring of one or more CSFs by FR-0 which indicates that the CSF(s) is under extreme challenge; prompt operator action is required.

RUPTURED: (Steam Generator) Existence of primary to secondary leakage of a magnitude greater than the capacity one charging pump.

SABOTAGE: Deliberate damage, misalignment, or misoperation of plant equipment with the intent to render the equipment inoperable.

SECURITY CONDITION: Any Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.

SIGNIFICANT TRANSIENT: An UNPLANNED event involving one or more of the following: (1) An automatic turbine runback >15% thermal reactor power; (2) Electrical load rejection >25% full electrical load; (3) Reactor Trip; (4) Safety Injection System Activation; (5) Thermal Power Oscillations $\geq 10\%$.

STRIKE ACTION: A work stoppage within the PROTECTED AREA by a body of workers to enforce compliance with demands made on TVA. The STRIKE ACTION must threaten to interrupt normal plant operations.

TOXIC GAS: A gas that is dangerous to life or limb by reason of inhalation or skin contact (e.g., chlorine, CO₂, etc.)

UNPLANNED: An event or action that is not the expected result of normal operations, testing or maintenance. Events that result in corrective or mitigative actions being taken in accordance with abnormal or emergency procedures are UNPLANNED.

UNPLANNED RELEASE: A release of radioactivity is UNPLANNED if the release has not been authorized by a Discharge Permit (DP). Implicit in this definition are unintentional releases, unmonitored releases, or planned releases that exceed a condition specified on the DP, (e.g., alarm setpoints, minimum dilution flow, minimum release times, maximum release rates, and/or discharge of incorrect tank).

VALID: An indication, report or condition is considered to be VALID when it is conclusively verified by (1) an instrument channel check, or (2) indication on related or redundant indicators, or (3) by direct observation by plant personnel. Implicit in this definition is the need for timely assessment within 15 minutes.

VISIBLE DAMAGE: Damage to equipment that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes deformation due to heat or impact, denting, penetration, rupture, cracking, or paint blistering. Surface blemishes (e.g., paint chipping, scratches, etc.) should NOT be included as visible damage.

VITAL AREA: Any area within the PROTECTED AREA which contains equipment, systems, devices, or material which the failure, destruction, or release of, could directly or indirectly endanger the public health and safety by exposure to radiation.

6.1 Loss of Shutdown Systems		6.2 Loss of S/D Capability	
Mode	Initiating / Condition	Mode	Initiating / Condition
	Refer to "Gaseous Effluents" (Section 7.1) and Continue in This Column.		Not Applicable.
5, 6	<p>Loss of water level in the reactor vessel that has or will uncover active fuel in the reactor vessel. (1 and 2 and 3):</p> <ol style="list-style-type: none"> 1. Loss of RHR capability. 2. VALID indication that reactor vessel water level < el. 695'. 3. Incore TCs (if available) indicate RCS temperature > 200 °F. <p>Note: If containment is open refer to "Gaseous Effluents" (Section 7.1) and continue in this column.</p>	1, 2, 3, 4	<p>Complete loss of function needed to achieve or maintain hot shutdown. (1 and [2a or 2b]):</p> <ol style="list-style-type: none"> 1. Hot shutdown required. 2a. CSF status tree indicated Core Cooling Red (FR-C.1). <p style="text-align: center;"><u>OR</u></p> <ol style="list-style-type: none"> 2b. CSF status tree indicates Heat Sink Red (FR-H.1) (RHR shutdown cooling not in service). <p>Note: Refer to "Reactor Protection System Failure" (Section 2.3) and Continue in This Column.</p>
5, 6	<p>Inability to maintain unit in cold shutdown when required (1 and 2):</p> <ol style="list-style-type: none"> 1. Cold shutdown required by Technical Specs. 2. Incore TCs (if available) indicate core exit temperature > 200 °F. <p>Note: If containment is open refer to "Gaseous Effluents" (Section 7.1) and continue in this column.</p>	1, 2, 3, 4	<p>Complete loss of function needed to achieve cold shutdown when cold shutdown required by Tech. Specs. (1 and 2 and 3):</p> <ol style="list-style-type: none"> 1. Cold shutdown required by Tech. Specs. 2. Loss of RHR shutdown cooling capability. 3. Loss of secondary heat sink and main condenser <p>Note: Also refer to "Failure of Rx Protection" (Section 2.3) and Continue in This Column.</p>
	Not Applicable.	1, 2, 3, 4	<p>Inability to reach required shutdown within Tech. Spec. limits.</p> <ol style="list-style-type: none"> 1. The unit has not been placed in the required mode within the time prescribed by the LCO action statement.

GENERAL

SITE AREA

ALERT

NOUE

6.3 loss of RCS Inventory	
Mode	Initiating / Condition
	Refer to "Gaseous Effluents" (Section 7.1) and Continue in This Column.
	Refer to "Gaseous Effluents" (Section 7.1) and Continue in This Column.
	Refer to "Gaseous Effluents" (Section 7.1) and Continue in This Column.
5, 6	<p>Loss of REACTOR COOLANT SYSTEM inventory with inadequate makeup. (1 and 2 and 3):</p> <ol style="list-style-type: none"> 1. Reactor coolant system is pressurized above atmospheric pressure. 2. Unplanned decrease in RCS or pressurizer level requiring initiation of makeup to the RCS. 3. With reactor coolant system temperature stable, the pressurizer level continues to decrease following initiation of RCS makeup.

GENERAL

SITE AREA

ALERT

NOUVE

FISSION PRODUCT BARRIER MATRIX

(Modes 1-4)

1

- 1.1 Fuel Clad Barrier
- 1.2 RCS Barrier
- 1.3 Containment Barrier

SYSTEM DEGRADATION

2

- 2.1 Loss of Instrumentation
- 2.2 Loss of Communication
- 2.3 Failure of Reactor Protection
- 2.4 Fuel Clad Degradation
- 2.5 RCS Unidentified Leakage
- 2.6 RCS Identified Leakage
- 2.7 Uncontrolled Cool Down
- 2.8 Turbine Failure
- 2.9 Safety Limit

LOSS OF POWER

3

- 3.1 Loss of AC (Power Ops)
- 3.2 Loss of AC (Shutdown)
- 3.3 Loss of DC

HAZARDS and SED JUDGMENT

4

- 4.1 Fire
 - 4.2 Explosion
 - 4.3 Flammable Gas
 - 4.4 Toxic Gas or Smoke
 - 4.5 Control Room Evacuation
 - 4.6 Security
 - 4.7 SED Judgment
- Table 4-1
Table 4-2
Figure 4-A
Figure 4-B

DESTRUCTIVE PHENOMENON

5

- 5.1 Earthquake
 - 5.2 Tornado
 - 5.3 Aircraft/Projectile
 - 5.4 River Level High
 - 5.5 River Level Low
 - 5.6 Watercraft Crash
- Table 5-1
Figure 5-A

SHUTDOWN SYSTEM DEGRADATION

6

- 6.1 Loss of Shutdown Systems
- 6.2 Loss of Shutdown Capability
- 6.3 Loss of RCS Inventory

RADIOLOGICAL EFFLUENTS

7

- 7.1 Gaseous Effluent
 - 7.2 Liquid Effluent
 - 7.3 Radiation Levels
 - 7.4 Fuel Handling
 - 7.5 Spent Fuel Storage
- Table 7-1
Table 7-2
Figure 7-A

Definitions and Abbreviations:

BOMB: An explosive device. (See EXPLOSION)

CIVIL DISTURBANCE: A group of twenty (20) or more persons within the EAB violently protesting onsite operations or activities at the site.

CONFINEMENT BOUNDARY: Spent Fuel Storage Cask CONFINEMENT BOUNDARY consists of MPC shell, bottom baseplate, MPC lid (including the vent and drain port cover plates), MPC closure ring, and associated welds.

CRITICAL-SAFETY FUNCTION (CSFs): A plant safety function required to prevent significant release of core radioactivity to the environment. There are six CSFs; Subcriticality, Core Cooling, Heat Sink, Pressurized Thermal Shock, Integrity (Containment) and Inventory (RCS).

EVENT: Assessment of an EVENT commences when recognition is made that one or more of the initiating conditions associated with the event exist. Implicit in this definition is the need for timely assessment within 15 minutes.

EXCLUSION AREA BOUNDARY (EAB): That area surrounding the reactor, in which the reactor licenses has the authority to determine all activities including exclusion or removal of personnel and property from the area. For purposes of Emergency Action Levels, based on radiological field measurements and dose assessments, and for design calculations, the Site Boundary shall be defined as the EAB.

EXPLOSION: Rapid, violent, unconfined combustion, or a catastrophic failure of pressurized or electrical equipment that imparts energy of sufficient force to potentially damage permanent structures or equipment.

EXTORTION: An attempt to cause an action at the site by threat or force.

FAULTED: (Steam Generator) Existence of secondary side leakage (e.g., steam or feed line break) that results in an uncontrolled decrease in steam generator pressure or the steam generator being completely depressurized.

FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical components do not constitute a fire. Observation of flame is preferred but is NOT required if large quantities of smoke and/or heat are observed.

FLAMMABLE GAS: Combustible gases at concentrations > than the LOWER EXPLOSIVE LIMIT (LEL).

HOSTAGE: A person(s) held as leverage against the site to ensure that demands will be met by the site.

HOSTILE ACTION: An act toward a nuclear plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land or water; using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. **HOSTILE ACTION** should NOT be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant.

HOSTILE FORCE: One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

IMMINENT: Within two hours.

INEFFECTIVE: When the specified restoration action(s) does not result in a reduction in the level of severity of the RED or ORANGE PATH condition within 15 minutes from identification of the CSF Status Tree RED or ORANGE PATH.

INITIATING CONDITIONS: Plant Parameters, radiation monitor readings or personnel observations that identify an Event for purposes of Emergency Plan Classification.

INTRUSION/INTRUDER: Suspected hostile individual present in the protected area without authorization.

ISFSI: Independent Spent Fuel Storage Installation

ODCM: Offsite Dose Calculation Manual is a supporting document to the Tech Specs. that contain Rad Effluent Controls, Environs Monitoring controls, and methodology for calculating routine gaseous and liquid effluent offsite doses and monitor alarm/trip setpoints.

ORANGE PATH: Monitoring of one or more CSFs by FR-0 which indicates that the CSF(s) is under severe challenge; prompt operator action is required.

PROJECTILE: An object ejected, thrown or launched towards a plant structure resulting in damage sufficient to cause concern regarding the integrity of the affected structure or the operability or reliability of safety equipment contained therein. The source of the projectile may be onsite or offsite.

PROTECTED AREA: The area encompassed by the security fence and to which access is controlled.

RCS: The RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary and secondary isolation valves.

RED PATH: Monitoring of one or more CSFs by FR-0 which indicates that the CSF(s) is under extreme challenge; prompt operator action is required.

RUPTURED: (Steam Generator) Existence of primary to secondary leakage of a magnitude greater than the capacity one charging pump.

SABOTAGE: Deliberate damage, misalignment, or misoperation of plant equipment with the intent to render the equipment inoperable.

SECURITY CONDITION: Any Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.

SIGNIFICANT TRANSIENT: An UNPLANNED event involving one or more of the following: (1) An automatic turbine runback >15% thermal reactor power; (2) Electrical load rejection >25% full electrical load; (3) Reactor Trip; (4) Safety Injection System Activation; (5) Thermal Power Oscillations $\geq 10\%$.

STRIKE ACTION: A work stoppage within the PROTECTED AREA by a body of workers to enforce compliance with demands made on TVA. The STRIKE ACTION must threaten to interrupt normal plant operations.

TOXIC GAS: A gas that is dangerous to life or limb by reason of inhalation or skin contact (e.g., chlorine, CO₂, etc.)

UNPLANNED: An event or action that is not the expected result of normal operations, testing or maintenance. Events that result in corrective or mitigative actions being taken in accordance with abnormal or emergency procedures are UNPLANNED.

UNPLANNED RELEASE: A release of radioactivity is UNPLANNED if the release has not been authorized by a Discharge Permit (DP). Implicit in this definition are unintentional releases, unmonitored releases, or planned releases that exceed a condition specified on the DP, (e.g., alarm setpoints, minimum dilution flow, minimum release times, maximum release rates, and/or discharge of incorrect tank).

VALID: An indication, report or condition is considered to be VALID when it is conclusively verified by (1) an instrument channel check, or (2) indication on related or redundant indicators, or (3) by direct observation by plant personnel. Implicit in this definition is the need for timely assessment within 15 minutes.

VISIBLE DAMAGE: Damage to equipment that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes deformation due to heat or impact, denting, penetration, rupture, cracking, or paint blistering. Surface blemishes (e.g., paint chipping, scratches, etc.) should NOT be included as visible damage.

VITAL AREA: Any area within the PROTECTED AREA which contains equipment, systems, devices, or material which the failure, destruction, or release of, could directly or indirectly endanger the public health and safety by exposure to radiation.

7.1 Gaseous Effluents		7.2 Liquid Effluents	
Mode	Initiating / Condition	Mode	Initiating / Condition
GENERAL SITE AREA ALERT NOUVE	<p>ALL</p> <p>EAB dose, resulting from an actual or imminent release of gaseous radioactivity > 1 Rem TEDE or > 5 Rem thyroid CDE for the actual or projected duration of release. (1 or 2 or 3):</p> <ol style="list-style-type: none"> 1. A VALID rad monitor reading exceeds the values under General Emergency in Table 7-1 for >15 min, unless assessment within that 15 min confirms that the criterion is not exceeded. 2. Field surveys indicate >1Rem/hr gamma or an I-131 concentration of 3.9E-06 $\mu\text{Ci}/\text{cm}^3$ at the EAB (Fig. 7-A) 3. Dose assessment results indicate EAB dose >1 Rem TEDE or >5 Rem thyroid CDE for the actual or projected duration of the release (Fig. 7-A). 		Not Applicable.
	<p>ALL</p> <p>EAB dose resulting from an actual or imminent release of gaseous radioactivity >100 mrem TEDE or >500 mrem thyroid CDE for actual or projected duration of release. (1 or 2 or 3):</p> <ol style="list-style-type: none"> 1. A VALID rad monitor reading > Table 7-1 values under Site Area for > 15 min, unless assessment within that 15 min confirms that the criterion is not exceeded. 2. Field surveys indicate >100 mrem/hr gamma or an I-131 conc of 3.9E-07 $\mu\text{Ci}/\text{cm}^3$ at the EAB (Fig. 7-A). 3. Dose assessment results indicate EAB dose >100 mrem TEDE or >500 mrem thyroid CDE for actual or projected duration of the release (Fig. 7-A). 		Not Applicable.
	<p>ALL</p> <p>Any UNPLANNED release of gaseous radioactivity that exceeds 200 times the ODCM Section 1.2.2.1 Limit for >15 minutes. (1 or 2 or 3 or 4)</p> <ol style="list-style-type: none"> 1. A VALID rad monitor reading > Table 7-1 values under Alert for >15 minutes, unless assessment within that 15 minutes confirms that the criterion is not exceeded. 2. Field surveys indicate >10 mrem/hr gamma at the EAB for >15 minutes (Fig 7-A). 3. Dose assessment results indicate EAB dose >10 mrem TEDE for the duration of the release (Fig. 7-A). 4. Sample results exceed 200 times the ODCM limit value for an unmonitored release of gaseous radioactivity >15 minutes in duration. 	<p>ALL</p> <p>Any UNPLANNED release of liquid radioactivity that exceeds 200 times the ODCM Section 1.2.1.1 Limit for >15 minutes. (1 or 2)</p> <ol style="list-style-type: none"> 1. A VALID rad monitor reading > Table 7-1 values under Alert for >15 minutes, unless assessment within this time period confirms that the criterion is not exceeded. 2. Sample results indicate an ECL >200 times the ODCM limit value for an unmonitored release of liquid radioactivity >15 minutes in duration 	
	<p>ALL</p> <p>Any UNPLANNED release of gaseous radioactivity that exceeds 2 times the ODCM Section 1.2.2.1 Limit for >60 minutes. (1 or 2 or 3 or 4)</p> <ol style="list-style-type: none"> 1. A VALID rad monitor reading > Table 7-1 values under UE for >60 minutes, unless assessment within that 60 minutes confirms that the criterion is not exceeded. 2. Field surveys indicate >0.1 mrem/hr gamma at the EAB for >60 minutes (Fig 7-A) 3. Dose assessment results indicate EAB dose >0.1 mrem TEDE for the duration of the release (Fig. 7-A). 4. Sample results exceed 2 times the ODCM limit value for an unmonitored release of gaseous radioactivity >60 minutes in duration 	<p>ALL</p> <p>Any UNPLANNED release of liquid radioactivity to the environment that exceeds 2 times the ODCM Section 1.2.1.1 Limit for >60 minutes. (1 or 2)</p> <ol style="list-style-type: none"> 1. A VALID rad monitor reading > Table 7-1 values under UE for >60 minutes, unless assessment within this time period confirms that the criterion is not exceeded. 2. Sample results indicate an ECL >2 times the ODCM limit value for an unmonitored release of liquid radioactivity >60 minutes in duration. 	

7.3 Radiation Levels		7.4 Fuel Handling	
Mode	Initiating / Condition	Mode	Initiating / Condition
	Refer to "Fission Product Barrier Matrix" (Section 1) or "Gaseous Effluents" (Section 7.1) and Continue in This Section.		Refer to "Gaseous Effluents" (Section 7.1) and Continue in This Section.
	Refer to "Fission Product Barrier Matrix" (Section 1) or "Gaseous Effluents" (Section 7.1) and Continue in This Section.		Refer to "Gaseous Effluents" (Section 7.1) and Continue in This Section.
A L L	<p>UNPLANNED increases in radiation levels within the facility that impedes safe operations or establishment or maintenance of cold shutdown. (1 or 2):</p> <p>1. VALID area radiation monitor readings or survey results exceed 15 mrem/hr in the control room or CAS.</p> <p style="text-align: center;"><u>OR</u></p> <p>2. (a and b):</p> <p style="padding-left: 20px;">a. VALID area radiation monitor readings exceed values listed in Table 7-2.</p> <p style="padding-left: 20px;">b. Access restrictions impede operation of systems necessary for safe operation or the ability to establish cold shutdown (See Note Below).</p>	A L L	<p>Major damage to irradiated fuel or loss of water level that has or will uncover irradiated fuel outside the reactor vessel. (1 and 2):</p> <p>1. VALID alarm on 0-RM-90-101B or 0-RM-90-102 or 0-RM-90-103 or 1-RM-90-130 or 2-RM-90-130 or 2-RM-90-131 or 2-RM-90-131 1-RM-90-112A or 1-RM-90-112B or 2-RM-90-112A or 2-RM-90-112B.</p> <p>2. (a or b):</p> <p style="padding-left: 20px;">a. Plant personnel report damage to irradiated fuel sufficient to rupture fuel rods.</p> <p style="text-align: center;"><u>OR</u></p> <p style="padding-left: 20px;">b. Plant personnel report water level drop has or will exceed makeup capacity such that irradiated fuel will be uncovered in the spent fuel pool or transfer canal.</p>
A L L	<p>UNPLANNED increase in radiation levels within the facility.</p> <p>1. A VALID area radiation monitor reading increases by 1000 mrem/hr over the highest reading in the past 24 hours excluding the current peak value.</p> <p><i>Note: In either the UE or ALERT EAL, the SED must determine the cause of increase in radiation levels and review other initiating conditions for applicability (e.g., a dose rate of 15 mrem/hr in the control room could be caused by a release associated with a DBA).</i></p>	A L L	<p>UNPLANNED loss of water level in spent fuel pool or reactor cavity or transfer canal with fuel remaining covered. (1 and 2 and 3):</p> <p>1. Plant personnel report water level drop in spent fuel pool or reactor cavity, or transfer canal.</p> <p>2. VALID alarm on 0-RM-90-101B or 0-RM-90-102 or 0-RM-90-103.</p> <p>3. Fuel remains covered with water.</p>

GENERAL

SITE AREA

ALERT

NOUE

7.5 Spent Fuel Storage	
Mode	Initiating / Condition
	Not Applicable.
	Not Applicable.
	Not Applicable.
ALL	<p>Damage to a loaded cask CONFINEMENT BOUNDARY from: (1 or 2 or 3)</p> <ol style="list-style-type: none"> 1. Natural phenomena (e.g., seismic event, tornado, flood, lightning, snow/ice accumulation, etc.). <p style="text-align: center;"><u>OR</u></p> <ol style="list-style-type: none"> 2. Accident (e.g: dropped cask, tipped over cask, explosion, missile damage, fire damage, burial under debris, etc). <p style="text-align: center;"><u>OR</u></p> <ol style="list-style-type: none"> 3. Judgment of the Site Emergency Director that the CONFINEMENT BOUNDARY damage is a degradation in the level of safety of the ISFSI

GENERAL

SITE AREA

ALERT

NOUE

**TABLE 7-1
EFFLUENT RADIATION MONITOR EALS**

NOTE: The monitor values below, if met or exceeded, indicate the need to perform the required assessment. If the assessment can not be completed within 15 minutes (60 minutes for UE), the appropriate emergency classification shall be made based on the **VALID** reading.

GASEOUS MONITORS	Units ⁽²⁾	UE	Alert	SAE	General
<i>Site Total Release Limit</i>	μCi/s	4.90E+05	4.90E+07	1.31E+08	1.31E+09
1-RI-90-400 (EFF LEVEL) - U-1 Shield Bldg	μCi/s	4.90E+05	4.90E+07	1.31E+08	1.31E+09
2-RI-90-400 (EFF LEVEL) - U-2 Shield Bldg	μCi/s	4.90E+05	4.90E+07	1.31E+08	1.31E+09
0-RM-90-101B - Auxiliary Bldg	cpm	1.03E+05	Offscale ⁽¹⁾	Offscale ⁽¹⁾	Offscale ⁽¹⁾
0-RM-90-132B - Service Bldg	cpm	2.62E+06	Offscale ⁽¹⁾	Offscale ⁽¹⁾	Offscale ⁽¹⁾
1-RI-90-421 thru 424 - U-1 MSL Monitors⁽²⁾	μCi/cc	1.71 E-01	1.71E+01	4.58E+01	4.58E+02
2-RI-90-421 thru 424 - U-2 MSL Monitors⁽²⁾	μCi/cc	1.71 E-01	1.71E+01	4.58E+01	4.58E+02
1-RM-90-255 or 256A - U-1 CVE	mR/h	4.10E+02	4.10E+04	1.09E+05	1.09E+06
2-RM-90-255 or 256A - U-2 CVE	mR/h	4.10E+02	4.10E+04	1.09E+05	1.09E+06
RELEASE DURATION	<i>minutes</i>	>60	>15	>15	>15
LIQUID MONITORS	Units	UE	Alert	Site Area	General
<i>Site Total Release Limit</i>	μCi/ml	6.50E-03	6.50E-01	N/A	N/A
0-RM-90-122 - RadWaste	cpm	1.45E+06	Offscale ⁽¹⁾	N/A	N/A
1-RM-90-120,121 - S/G Bldn	cpm	1.07E+06	Offscale ⁽¹⁾	N/A	N/A
2-RM-90-120,121 - S/G Bldn	cpm	1.07E+06	Offscale ⁽¹⁾	N/A	N/A
0-RM-90-225 - Condensate Demin	cpm	1.90E+06	Offscale ⁽¹⁾	N/A	N/A
0-RM-90-212 - Turbine Building Sump	cpm	3.28E+03	3.28E+05	N/A	N/A
RELEASE DURATION	<i>minutes</i>	>60	>15	>15	>15

ASSESSMENT METHODS:

- ◆ Airborne Dose Assessment per SQN EPIP-13 "Dose Assessment"
- ◆ ODCM Liquid Release Rate assessment per SQN 0-TI-CEM-030.030.0
- ◆ Integrated Airborne Release Rate assessment per SQN 0-TI-CEM-030.030.0

- (1) The calculated value is outside of the upper range for this detector. The maximum monitor output which can be read is 1.0E+07 cpm. Releases in excess of monitor capacity should be evaluated for proper classification by use of Dose Assessment.
- (2) These unit values are based on flow rates through one PORV of 890,000 lb/hr at 1078.7 psia with 0.25% carry over (0.9975 quality). Before using these values, ensure a release to the environment is ongoing, (e.g., PORV).

NOTE 1: These EALs are based on the assumption that an emergency release is restricted to one pathway from the plant. In all cases, the total site EAL is the limiting value. Therefore, in the case where there are multiple release paths from the plant, it is the total release EAL (obtained from ICS and/or SQN 0-TI-CEM-030-030, "Manual Calculation of Plant Gas, Iodine, and Particulate Release Rates for Offsite Dose Calculation Manual (ODCM) Compliance") that will determine whether an emergency classification is warranted.

NOTE 2: In the case when there is no CECC dose assessment available, the length and relative magnitude of the release is the key in determining the classification. For example, in the case of the NOUE EAL of 2 times the Tech Spec limit, the classification is based more on the fact that a release above the limit has continued unabated for more than 60 minutes, than on the projected offsite dose.

NOTE 3: See REP Appendix B for basis information.

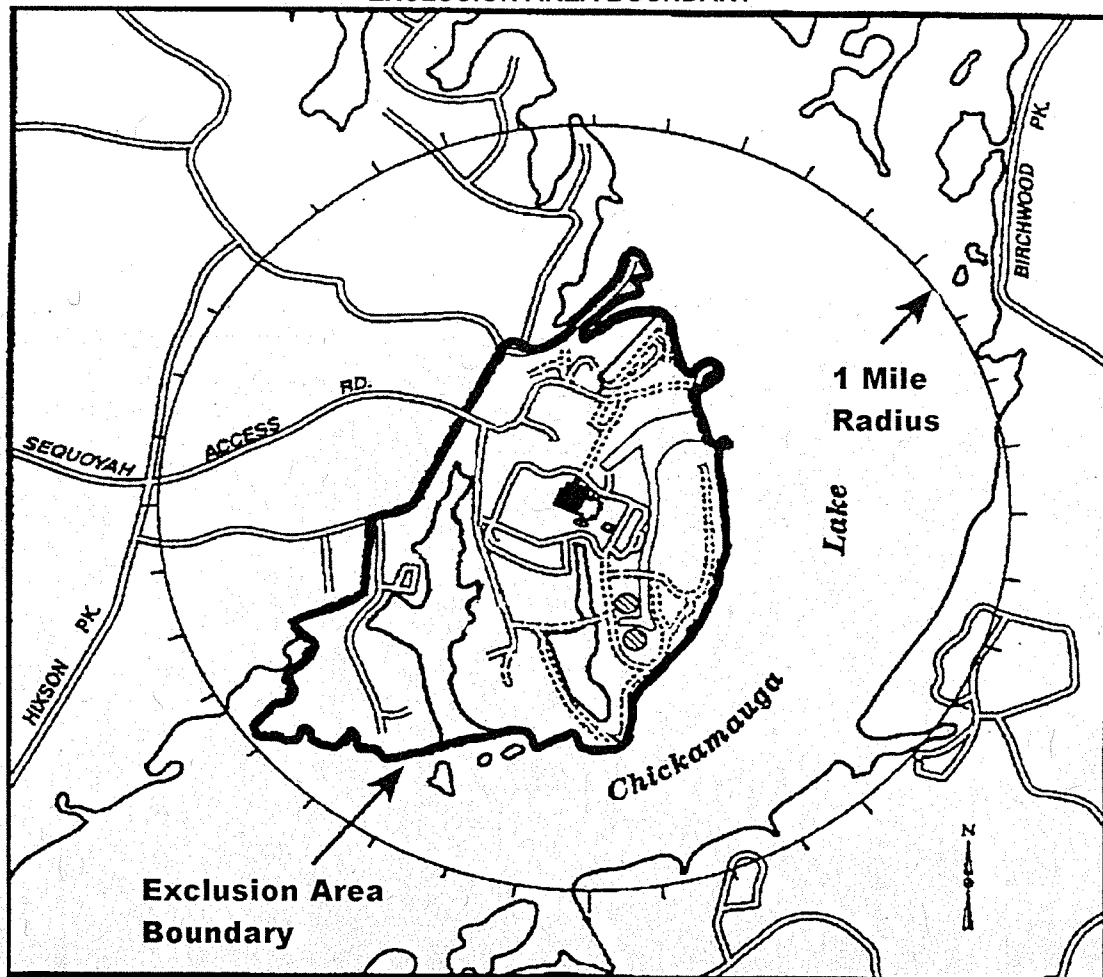
Table 7-2
ALERT - RADIATION LEVELS

For purposes of comparing the meter/monitor reading values to this table, it can be assumed that mR is equivalent to mrem.

Monitor No.	Location - Area and Elevation	Meter Reading
1,2-RM-90-1	Spent Fuel Pit ARM El. 734.0	1.5E+03 mR/hr
0-RM-90-5	SFP Pumps ARM El. 714.0	1.5E+03 mR/hr
1,2-RM-90-6	CCS HXS ARM El. 714.0	1.5E+03 mR/hr
1,2-RM-90-7	Sample Rm ARM El. 690.0	1.5E+03 mR/hr
1,2-RM-90-8	AFW Pumps ARM El. 690.0	1.5E+03 mR/hr
0-RM-90-9	Waste Cnds Tks ARM El. 669.0	1.5E+03 mR/hr
1,2-RM-90-10	CVCS Bd ARM El. 669.0	1.5E+03 mR/hr
0-RM-90-11	CS and RHR Pumps Radmon El. 653.0	1.5E+03 mR/hr
0-RM-90-102	Spent Fuel Pit Radmon El. 734.0	1.5E+03 mR/hr
0-RM-90-103	Spent Fuel Pit Radmon El. 734.0	1.5E+03 mR/hr
0-RM-90-230	CNDS Demineralizer ARM El. 685.0	1.5E+03 mR/hr
0-RM-90-231	Cnds Demineralizer ARM El. 706.0	1.5E+03 mR/hr

Note: All of the above monitors have a range of 0.1 to 1E+4 mrem/hr.

Figure 7-A
EXCLUSION AREA BOUNDARY



TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT
EMERGENCY PLAN IMPLEMENTING PROCEDURE

EPIP-5

GENERAL EMERGENCY

REVISION 39

PREPARED BY: BILL PEGGRAM

RESPONSIBLE
ORGANIZATION: EMERGENCY PREPAREDNESS

APPROVED BY: RUSSELL THOMPSON

EFFECTIVE DATE: 02/19/2010

LEVEL OF USE: REFERENCE USE

QUALITY-RELATED

Revision History

Rev	Date		Reason for Revision
30	04/01/2003		General Revision to restructure EPIP for better flow. Moved ODS notification earlier in procedures. Added evacuation sectors to Initial Notification Appendix and to consider issuance of KI in accordance with the State Plan in the PAR. Intent Change.
31	06/25/2003	9, 14	Non intent change. Phone number correction. Changed title of Appendix B and added note o match Figure 10-1 of the TVA REP.
32	10/23/2003	4, 8, 12, 13	Intent change. Added step to record time of declaration upon entry into the procedure. Steps concerning dose assessment, PAR, PAR changes, and announce GE to Plant Mgmt., NRC, SM/SED that notifications are complete. Split step that had two actions in one step. Specified Security implement EPIP-11
33	04/22/2004	3, 6, 13	Intent Change: Made corrections to the Table of Contents making sections titles consistent with EPIPs 3,4 and sections within the procedure. Added "SED's Initials" to Section 3.2. Clarified that MSS/WWM in the OSC is verifying ERO response and that SM is to ENSURE that this is in progress. Added guidance to utilize EPIP-6 Apdx B to initially brief NRC using ENS line.
34	09/23/2004	6, 7, 8, 9, 13	Intent Change: Changed Bradley County EMA phone number, removed TEMA satellite phone numbers, added classification validation to Sect.3.2, moved transfer of PARs from Sect. 3.3 to Sect 3.2 , App. A: added to GE announcement to staff TSC/OSC and corrected "SAE" to "General Emergency".
35	06/10/2005	4, 14, 15, 16	Revision Change: replaced SSI-1 with SSI-7.1. Replaced the PAR Chart with the new PAR Chart, NRC Regulatory Issue Summary 2004-13, which addressed the range of protective actions that includes sheltering for the public.
36	04/28/2006	15,16	Revision summary: Changed App.C, Step 10 from being the time and date info was provided to the ODS to faxing App. C to the ODS. Made App. D consistent with App. C by putting "THIS IS A DRILL" before "THIS IS A REAL EVENT".
37	01/23/2007	6, 7, 8, 12, 13	Plan effectiveness determinations reviews indicate the following revisions do not reduce the level of effectiveness of the procedure of REP: Changed the phone number for Bradley County. Changed the call to request a dose assessment from Chemistry to Radiation Protection and changed the corresponding phone numbers so that calls are directed to the RP Lab and no longer the Chem Lab. Changed to current organization titles, added to make Alert announcement on old plant PA and the x4800 bridge. Annual review. Revised responsibility of dose assessment from Chemistry to RP.
38	12/15/2008	8, 11, 15	Plan effectiveness determinations reviews indicate the following revisions do not reduce the level of effectiveness of the procedure of REP. Annual Review. Added a place keeping box in Sect. 3.3[1]. Clarified use of App C when a PAR is changed and CECC has not been activated. Changed SHELTER wording on App. C to match the PAR chart wording on recommendations 1&2.
39	02/19/2010	15	Revision Summary: Plan effectiveness determinations reviews indicate the following revisions do not reduce the level of effectiveness of the procedure of REP: Annual review. PER 162926-001, Revised Appendix C to change step 10 to say, "When completed, FAX this information to the ODS or TEMA as required by Sections 3.1 or 3.2".

Table of Contents

1.0 PURPOSE4

2.0 REFERENCES4

3.0 INSTRUCTIONS.....4

3.1 General Emergency Declaration by the Main Control Room4

3.2 General Emergency Declaration by the TSC.....7

3.3 Monitor Conditions.....8

3.4 Termination of the Event.....10

4.0 RECORDS RETENTION11

4.1 Records of Classified Emergencies11

4.2 Drill and Exercise Records11

5.0 ILLUSTRATIONS AND APPENDICES11

5.1 Appendix A, Notifications and Announcements12

5.2 Appendix B, Protective Action Recommendations.....14

5.3 Appendix C, TVA Initial Notification of General Emergency15

5.4 Appendix D, General Emergency Follow-up Information16

1.0 PURPOSE

- 1.1 To provide a method for timely notifications of appropriate individuals or organizations when the Shift Manager (SM)/Site Emergency Director (SED) has determined by EPIP-1 that events have occurred that are classified as a **GENERAL EMERGENCY (GE)**.
- 1.2 To provide the SED/SM a method for periodic reanalysis of current conditions to determine whether the **GENERAL EMERGENCY** should be terminated or continued.

2.0 REFERENCES**2.1 Interface Documents**

- [1] SPP-3.5 "Regulatory Reporting Requirements"
- [2] EPIP-6, "Activation and Operation of the Technical Support Center"
- [3] EPIP-7, "Activation and Operation of the Operations Support Center, OSC"
- [4] EPIP-8, "Personnel Accountability and Evacuation"
- [5] EPIP-10, "Emergency Medical Response"
- [6] EPIP-13, "Dose Assessment"
- [7] EPIP-14, "Radiation Protection Response"
- [8] EPIP-16, "Termination and Recovery"
- [9] CECC EPIP-9, "Emergency Environmental Radiological Monitoring Procedures"
- [10] SSI-7.1, "Post Requirements and Responsibilities, Central and Secondary Alarm Stations"

3.0 INSTRUCTIONS

NOTE: IF there are personnel injuries, **THEN IMPLEMENT** EPIP-10, "Emergency Medical Response" in parallel with this procedure.

NOTE: IF there are immediate hazards to plant personnel, **THEN** consider immediately implementing EPIP-8 "Personnel Accountability and Evacuation" in parallel with this procedure

3.1 GENERAL EMERGENCY DECLARATION BY THE MAIN CONTROL ROOM

Upon classifying events as a "**GENERAL EMERGENCY**", the SM/SED shall:

- [1] IF TSC is OPERATIONAL, (SED transferred to TSC), **THEN GO TO** Section 3.2.
- [2] **RECORD time of Declaration**

Time

3.1 GENERAL EMERGENCY DECLARATION BY THE MAIN CONTROL ROOM (Continued)

- [3] **ACTIVATE** Emergency Paging System (EPS) as follows.
 - [a] IF EPS has already been activated, **THEN GO TO Step 4.**
 - [b] IF ongoing onsite Security events may present risk to the emergency responders, **THEN CONSULT** with Security to determine if site access is dangerous to the life and health of emergency responders.
 - [c] IF ongoing events makes site access dangerous to the life and health of emergency responders, **THEN SELECT STAGING AREA** button on the EPS terminal **INSTEAD** of the EMERGENCY button.
 - [d] **ACTIVATE** EPS using touch screen terminal. IF EPS fails to activate, **THEN** continue with step 4.
- [4] **EVALUATE** Protective Action Recommendations (PARs) using Appendix B.
- [5] **COMPLETE** Appendix C (TVA Initial Notification for General Emergency).

NOTE: ODS should be notified within 5 minutes after declaration of the event.

[6] **NOTIFY** ODS. _____ Initial _____ Time

ODS: Ringdown Line or
5-751-1700 or 5-751-2495 or 9-785-1700

- [a] IF EPS failed to activate from SQN, **THEN DIRECT** ODS to activate SQN EPS. IF ODS is also unable to activate EPS, **THEN** continue with step [5] [b].
- [b] **READ** completed Appendix C to ODS.
- [c] **FAX** completed Appendix C to ODS.

5-751-8620 (Fax)

[d] **MONITOR** for confirmation call from ODS that State/Local notifications complete: **RECORD** time State notified. _____ Notification Time

3.1 GENERAL EMERGENCY DECLARATION BY THE MAIN CONTROL ROOM (Continued)

[7] IF ODS CANNOT be contacted within 10 minutes of the declaration, **THEN**

[a] CONTACT Hamilton County Emergency Management Agency (EMA) **AND READ** completed Appendix C. _____
Initial _____
Time

9-209-6900 or 9-622-7777 or 9-622-0022

[b] CONTACT Bradley County EMA **AND READ** completed Appendix C. _____
Initial _____
Time

9-728-7289 or 9-728-7290

[c] NOTIFY Tennessee Emergency Management Agency (TEMA) **AND READ** completed Appendix C. _____
Initial _____
Time

9-1-800-262-3300 or 9-1-615-741-0001

[d] FAX completed Appendix C to TEMA.

9-1-615-242-9635 (Fax)

[8] ENSURE MSS/WWM in the OSC (x6427) is monitoring Emergency Response Organization (ERO) responses using printed report available in the OSC.

[a] IF any ERO positions are not responding, **THEN DIRECT** MSS to **CALL** personnel to staff TSC/OSC positions. (Use REP Duty Roster and Call List.)

[9] NOTIFY plant staff using Appendix A. (Delegate as needed.)

[10] GO TO Section 3.3

3.2 GENERAL EMERGENCY DECLARATION BY THE TSC

Upon classifying events as a "GENERAL EMERGENCY", the SED shall:

NOTE: CECC Director should be notified within 5 minutes after declaration of the event.

- [1] **RECORD** Time of Declaration _____
- [2] **RECORD** EAL(s) _____
- [3] **VALIDATE** time and EAL numbers with the Ops Mgr, Site VP or EP Mgr.
- [4] **IF** PAR responsibility has **NOT** been transferred to the CECC Director,
 - [a] **THEN REFER** to Appendix B (Protective Action Recommendations)
 - [b] **FAX** Appendix C (Notification of General Emergency) to CECC Director
- [5] **CALL** CECC Director and inform of escalation, time of declaration, EAL(s) declared, and description of events.

 SED's Initials Time

Ringdown Line or 5-751-1614 or 5-751-1680

- [6] **IF** CECC Director **CANNOT** be contacted within 10 minutes of the declaration, **THEN**
 - [a] **COMPLETE** Appendix C (TVA Initial Notification for General Emergency) using Appendix B to evaluate Protective Actions.
 - [b] **NOTIFY** Hamilton County EMA **AND READ** Appendix C. _____

SED's Initials Time

9-209-6900 or 9-622-7777 or 9-622-0022

- [c] **NOTIFY** Bradley County EMA **AND READ** Appendix C. _____

SED's Initials Time

9-728-7289 or 9-728-7290

3.2 GENERAL EMERGENCY DECLARATION BY THE TSC

[d] **NOTIFY TEMA AND READ** completed Appendix C. _____
SED's Initials Time

9-1-800-262-3300 or 9-1-615-741-0001

[e] **FAX** completed Appendix C to TEMA.

9-1-615-242-9635 (Fax)

3.3 MONITOR CONDITIONS

[1] **MONITOR** radiation monitors.

[2] **WHEN** indication exists of an unplanned radiological release,
THEN ENSURE Dose Assessment is performed.

[a] **IF** the CECC has not assumed Dose Assessment responsibility,
THEN NOTIFY Radiation Protection to perform a dose
assessment using EPIP-13, "Dose Assessment"

AND

PROVIDE the following information:

1. **Type Of Event** (SGTR/L, LOCA, WGDT, Cntmt Bypass)
 2. **Release Path** (SG/PORV, Aux, Shld, Turb, Serv, Cond)
 3. **Expected Duration** (If unknown assume 4 hour duration)
-

7865 (RP Lab) or 6417 (RP Lab) or
Use Call List to Page RP Lead

[b] **IF** changes to PARs are necessary,
THEN complete Appendix C (IF CECC has not assumed responsibility
for PARs) and D.

CAUTION: Accountability should **NOT** be initiated at this time **IF** Assembly will
present a danger to employees - For example:
A severe weather condition exists or is imminent (such as a Tornado)
An onsite Security risk condition exists (Consult with Nuclear Security)

[3] **IF** personnel accountability has not been previously initiated,
THEN ACTIVATE assembly and accountability by using EPIP-8,
Appendix C (may be delegated).

3.3 MONITOR CONDITIONS (Continued)

[4] MONITOR plant conditions:

[a] EVALUATE conditions using EPIP-1:

[1] IF additional conditions satisfy criteria of other GENERAL EMERGENCY(s) THEN complete Appendix D.

[2] IF conditions warrant a need for follow-up information, THEN complete Appendix D.

[b] IF Appendix D completed, THEN

[1] REPORT to CECC for State notification. _____
Initial Time

CECC Director: Ringdown Line or
 5-751-1614 or 5-751-1680
 OR
 ODS: Ringdown Line or 5-751-1700 or
 5-751-2495 or 9-785-1700

[2] FAX completed Appendix D to CECC.

CECC: 5-751-1682 (Fax) OR ODS: 5-751-8620 (Fax)

[3] IF neither the CECC or ODS can be reached, THEN

[a] NOTIFY TEMA AND READ completed Appendix D. _____
Initial Time

9-1-800-262-3300 or 9-1-615-741-0001

[b] FAX completed Appendix D to TEMA.

9-1-615-242-9635 (Fax)

3.4 TERMINATION OF THE EVENT

[1] IF the situation no longer exists, THEN

[a] **TERMINATE** emergency per EPIP-16, "Termination and Recovery".

[b] **COMPLETE** Appendix D including Time and Date Event Terminated.

[c] **FAX** completed Appendix D to CECC Director.

ODS: 5-751-8620 (Fax) OR
CECC: 5-751-1682 (Fax)

[2] **COLLECT** documentation and **FORWARD** to Emergency Preparedness.

END OF SECTION

4.0 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 and 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 ILLUSTRATIONS AND APPENDICES

5.1 Appendix A - Notifications and Announcements

Appendix A provides guidance for security threats, and for prompt notification of the NRC Resident and plant personnel.

5.2 Appendix B - Protective Action Recommendation Logic Diagram

Appendix B, Protective Action Recommendation Logic Diagram, is used to determine the Protective Action Recommendation which is made to the State and is part of the initial notification made to the State. Protective Action Recommendations are the responsibility of the CECC Director after assuming the responsibility from the SED.

5.3 Appendix C - TVA Initial Notification of General Emergency

Appendix C, TVA Initial Notification of General Emergency, is the form used to initially notify the Operations Duty Specialist who notifies the Tennessee Emergency Management Agency.

5.4 Appendix D - General Emergency Follow-up Information

Appendix D, General Emergency Follow-up Information is the form used to provide additional information concerning other General Emergencies or other information concerning additional conditions to the ODS for State notification and event termination.

Appendix A
 NOTIFICATIONS AND ANNOUNCEMENTS

(Page 1 of 2)

[1] IF there is a security threat, THEN

- [a] NOTIFY Security Shift Supervisor to implement SSI-1, "Security Instructions For Members Of The Security Force" and EPIP-11 "Security and Access Control".

Initial Time

6144 or 6568

- [b] DETERMINE if Security recommends implementing the "Two Person Line of Sight" Rule.

- [c] IF Nuclear Security recommends establishing the "Two Person Line of Sight" Rule, THEN INFORM the SM/SED. ("Two Person Line of Sight" requires use of EPIP-8.)

Initial Time

[2] NOTIFY Radiation Protection Lead:

- [a] STATE: "A GENERAL EMERGENCY HAS BEEN DECLARED, BASED UPON (*Describe the conditions*), AFFECTING UNIT(s) _____."

Initial Time

7865 (RP Lab) or 6417, (RP Lab)
 Use Call List to Page RP Lead

- [b] DIRECT Radiation Protection to implement EPIP-14, "Radiation Protection Response".

- [c] DIRECT Radiation Protection to implement CECC EPIP-9, "Emergency Environmental Radiological Monitoring Procedures" which includes activation of the radiological monitoring van.

[3] NOTIFY personnel in the Chemistry Lab:

- [a] STATE: "A GENERAL EMERGENCY HAS BEEN DECLARED, BASED UPON (*Describe the conditions*), AFFECTING UNIT(s) _____."

Initial Time

7285 (Lab) or 6348 (Lab) or 20126 (Pager)

- [b] DIRECT Chemistry to implement EPIP-14, "Radiation Protection Response".

Appendix A
NOTIFICATIONS AND ANNOUNCEMENTS
 (Page 2 of 2)

- [4] **ANNOUNCE** to plant personnel on old plant PA and x4800:

 - [a] "ATTENTION PLANT PERSONNEL. ATTENTION PLANT PERSONNEL. A **GENERAL EMERGENCY** HAS BEEN DECLARED BASED ON (Describe the condition), AFFECTING UNIT(s) _____. (if not already staffed, add) STAFF THE TSC AND OSC."
 - [b] **REPEAT** Announcement.
- [5] **NOTIFY** Plant Management in accordance with SPP-3.5 **AND PROVIDE** General Emergency Information. _____
Initial _____
Time
- [6] **NOTIFY** the "On Call" NRC Resident **AND PROVIDE** General Emergency Information. _____
Initial _____
Time

NOTE: NRC ENS notification should be made as soon as practicable, but within 1 hour of "**GENERAL EMERGENCY**" declaration. Whenever NRC requests, a qualified person must provide a continuous update to NRC Operations Center. Use EPIP-6, Appendix B as a briefing guide.

- [7] **NOTIFY** NRC of plan activation via ENS phone _____
Initial _____
Time

9-1-(301) 816-5100 (Main)
 9-1-(301) 951-0550 (Backup)
 9-1-(301) 816-5151 (Fax)
- [8] **NOTIFY** the SM/SED that notifications are complete. _____
Initial _____
Time

Appendix B

PROTECTIVE ACTION RECOMMENDATIONS

Note 1: If conditions are unknown utilizing the flowchart, then answer is NO.

Note 2: A short term release is defined as "a release that does not exceed a 15 minute duration".

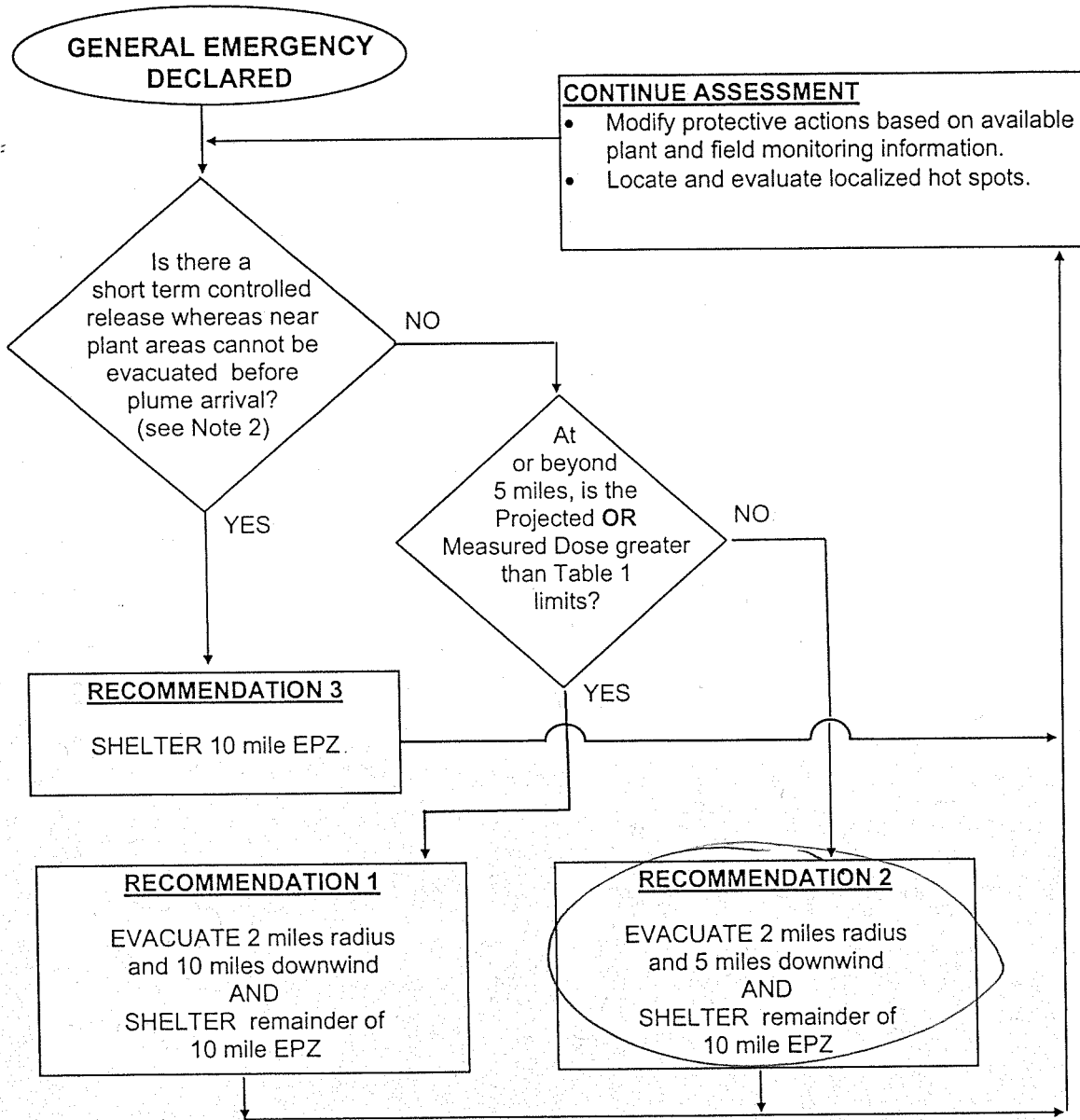


TABLE 1 Protective Action Guides (PAG)	
TYPE	LIMIT
Measured	3.9 E-6 micro Ci/cc of Iodine 131 or 1 REM per hour External Dose
Projected	1 REM TEDE or 5 REM Thyroid CDE

Note: Unknown conditions are assumed less than listed conditions.

Appendix C
TVA INITIAL NOTIFICATION OF GENERAL EMERGENCY

1. This is a Drill This is an Actual Event - Repeat - This is an Actual Event

2. This is _____, Sequoyah has declared a **GENERAL EMERGENCY**
affecting: Unit 1 Unit 2 Both Unit 1 and Unit 2

3. EAL Designator(s): _____

4. Brief Description of the Event: _____

5. Radiological Conditions: (Check one under both Airborne and Liquid column.)

<p><u>Airborne Releases Offsite</u></p> <input type="checkbox"/> Minor releases within federally approved limits ¹ <input type="checkbox"/> Releases above federally approved limits ¹ <input type="checkbox"/> Release information not known (¹ Tech Specs)	<p><u>Liquid Releases Offsite</u></p> <input type="checkbox"/> Minor releases within federally approved limits ¹ <input type="checkbox"/> Releases above federally approved limits ¹ <input type="checkbox"/> Release information not known (¹ Tech Specs)
---------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------	-------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------

6. Event Declared: Time: _____ Date: _____

7. The Meteorological Conditions are: (Use 46 meter data from the Met Tower)

Wind Direction is FROM: _____ degrees Wind Speed: _____ m.p.h

8. Provide Protective Action Recommendation: (Check either 1 or 2 or 3.)

<input type="checkbox"/> Recommendation 1 <ul style="list-style-type: none"> EVACUATE LISTED SECTORS (2 mile Radius and 10 miles downwind) SHELTER remainder of 10 mile EPZ CONSIDER issuance of POTASSIUM IODIDE in accordance with the State Plan. 	R E C 1	WIND FROM *DEGREES DIRECTION (item 7) (Mark)	R E C 2	<input type="checkbox"/> Recommendation 2 <ul style="list-style-type: none"> EVACUATE LISTED SECTORS (2 mile Radius and 5 mile downwind) SHELTER remainder of 10 mile EPZ CONSIDER issuance of POTASSIUM IODIDE in accordance with the State Plan.
A-1, B-1, C-1, D-1, C-2, -6, -7, -8, D-2, -3, -5, -6		12 - 49		A-1, B-1, C-1, D-1, C-2, D-2
A-1, B-1, C-1, D-1, D-2, -3, -4, -5, -6		50 - 70		A-1, B-1, C-1, D-1, D-2
A-1, B-1, C-1, D-1, A-3, -4, D-2, -3, -4, -5		71 - 112		A-1, B-1, C-1, D-1, A-3, D-2
A-1, B-1, C-1, D-1, A-2, -3, -4, -5, -6, D-4		113 - 146		A-1, B-1, C-1, D-1, A-2, A-3,
A-1, B-1, C-1, D-1, A-2, -3, -4, -5, -6, B-2		147 - 173		A-1, B-1, C-1, D-1, A-2, A-3, B-2
A-1, B-1, C-1, D-1, A-2, -5, -6, B-2, -3, -4		174 - 214		A-1, B-1, C-1, D-1, A-2, B-2
A-1, B-1, C-1, D-1, B-2, -3, -4, -5, -6, -7, -8		215 - 258		A-1, B-1, C-1, D-1, B-2, B-5,
A-1, B-1, C-1, D-1, B-2, -3, -5, -6, -7, -8, C-2, -3, -4, -5, -6		259 - 331		A-1, B-1, C-1, D-1, B-2, B-5, C-2
A-1, B-1, C-1, D-1, B-5, C-2, -3, -4, -5, -6, -7, -8		332 - 11		A-1, B-1, C-1, D-1, B-5, C-2

Recommendation 3

- SHELTER all sectors.
- CONSIDER issuance of Potassium Iodide in accordance with the State Plan.

9. Please repeat back the information you have received to ensure accuracy.

10. When completed, FAX this information to the ODS or TEMA as required by Sections 3.1 or 3.2.

Appendix D
GENERAL EMERGENCY FOLLOW-UP INFORMATION

1. THIS IS A DRILL THIS IS A REAL EVENT
2. There has been a GENERAL EMERGENCY declared at Sequoyah affecting:
3. Reactor Status: Unit 1: Shut Down At Power Refueling N/A
4. Additional EAL Designators
5. Significant Changes in Plant Conditions:
6. Significant Changes in Radiological Conditions:
7. Offsite Protective Action Recommendation: (CECC to provide detailed PAR Sector Recommendations)
8. Onsite Protective Actions: Assembly and Accountability Site Evacuation
9. The Meteorological Conditions are: Wind Speed: m.p.h. Wind Direction is from: degrees
10. Event Terminated: Date/Time
11. Please repeat the information you have received to ensure accuracy.
12. FAX to ODS at 5-751-8620 or CECC Director at 5-751-1682 after completing the notification.
Completed by: Date/Time