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PSEG
Nuclear LLC

10 CFR 50.54(q)

LR-N22-0039
April 21, 2022

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Salem Nuclear Generating Station, Units 1 and 2
Renewed Facility Operating License Nos. DPR-70 and DPR-75
NRC Docket Nos. 50-272 and 50-311

Hope Creek Generating Station
Renewed Facility Operating License No. NPF-57
NRC Docket No. 50-354

Subject: Emergency Plan Document Revisions Implemented March 24, 2022.

Pursuant to 10 CFR 50.54(q) and 10 CFR 50.4(b)(5), PSEG Nuclear LLC (PSEG) is submitting 10 CFR 50.54(q) Summary Analysis Report numbered 2022-03 for the revision to procedures EP-SA-325-141, Salem Emergency Classification Guide Wall Chart, EP-SA-325-117, Salem Section S - S4 – RCS Activity, and EP-SA-325-217, Salem Section S EAL Technical Basis implemented on March 24, 2022 (Attachments 1, 2, 3 and 4).

There are no regulatory commitments contained in this letter.

Should you have any questions, or require further information regarding this submittal, please contact Ms. Megean M. Brown (856) 339-1773.

Respectfully,

Barr, Stephen T.

Digitally signed by Barr, Stephen T.
Date: 2022.04.18 07:49:43 -04'00'

Stephen T. Barr
Manager, Emergency Preparedness

Attachment 1 – 10 CFR 50.54(q) Summary Analysis Report 2022-03
Attachment 2 – EP-SA-325-141 – Salem Emergency Classification Guide Wall Chart
Attachment 3 – EP-SA-325-117 – Salem Section S - S4 – RCS Activity
Attachment 4 – EP-SA-325-217 – Salem Section S EAL Technical Basis

cc (w/ Attachments):	USNRC Administrator, Region I USNRC Project Manager USNRC Senior Resident Inspector, Salem USNRC Senior Resident Inspector, Hope Creek
(w/o Attachments):	NJDEP Bureau of Nuclear Engineering PSEG Corporate Commitment Tracking Coordinator

ATTACHMENT 1

10 CFR 50.54(q) Summary Analysis Report 2022-03

ATTACHMENT 3
10CFR50.54(q) SUMMARY ANALYSIS REPORT
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Revision 0

50.54Q I.D. Number: **2022-03**

50.54Q Title: **Revision to Emergency Action Level (EAL) SU4.2:**
EP-SA-325-117, Rev. 1, RCS Activity (Flow Chart)
EP-SA-325-217, Rev. 1, RCS Activity (Technical Basis)
EP-SA-325-141, Rev. 2, EAL Wall Chart – Hot Conditions

(Doc #, Rev. #, Name, If applicable)

Description of the change made to the Emergency Plan/Procedures:
EAL SU4.2, as defined in EP-SA-325-117, EP-SA-325-217 and EP-SA-325-141, is being revised IAW Technical Specification Amendment Nos. 337 and 318 (LAR S20-01). The amendment removed Figure 3.4-1 and associated references from the Technical Specifications for both Salem U1 (TS 3.4.8) and Salem U2 (TS 3.4.9), and inserts a limit of less than or equal to the site-specific Dose Equivalent Iodine (DEI) spiking limit of 60 microcuries per gram. A new specific activity for Dose Equivalent Xe-133 (DEX) is also implemented by this amendment. The Technical Specifications have been modified to provide an action for when DEX is not ≤ 600 μCi/gram, and to remove the limit associated with gross activity of the reactor coolant Ebar (Ē). The site-specific limit of 600 μCi/gram DEX is established based on the maximum accident analysis RCS activity corresponding to 1 percent fuel clad defects. IAW with the analysis provided in LAR S20-01, if iodine or noble gas spiking were to occur, the normal coolant concentration would be restored within the 48 hour time period provided. Also, there is a low probability of a Steam Line Break (SLB) or Steam Generator Tube Rupture (SGTR) occurring during this time period. The EAL will be revised to remove the reference to Technical Specification Figure 3.4-1, and replace with the following EAL Threshold values: SU4.2 Reactor coolant activity > ANY: • 60 μCi/gram Dose Equivalent I-131 • 1.0 μCi/gram Dose Equivalent I-131 for > 48 hrs. • 600 μCi/gram Dose Equivalent XE-133 for > 48 hrs.
Description of why the change is editorial (if not editorial, N/A this block): N/A

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Revision 0

50.54Q I.D. Number: **2022-03**

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EP-SA-325-117, Rev. 1, RCS Activity (Flow Chart)
EP-SA-325-217, Rev. 1, RCS Activity (Technical Basis)
EP-SA-325-141, Rev. 2, EAL Wall Chart – Hot Conditions

(Doc #, Rev. #, Name, If applicable)

Description of the licensing basis affected by the change to the Emergency Plan/Procedure (if not affected, omit this element):

The following Emergency Plan Sections were reviewed:

- Emergency Plan Section 5.0 - Emergency Classification System
- Emergency Plan Section 6.0 - Notification Methods
- Emergency Plan Section 10.0 - Accident Assessment
- Emergency Plan Section 16.0 - Radiological Emergency Response Training

The emergency plan sections listed above describe methods and processes for accident assessment, classification and notifications, as well as training requirements. These sections do not specify emergency action thresholds or specific activity limits for reactor coolant and therefore are not impacted by the proposed change.

A description of how the change to the Emergency Plan/Procedures still complies with regulation:

The addition of the specific activity limits (e.g. DEX) is consistent with guidance provided in NEI 99-01, Revision 6 that states: *Developers may reword the EAL to include the reactor coolant activity parameter(s) specified in Technical Specifications and the associated allowable limit(s) (e.g., values for dose equivalent 1-131 and gross activity, time-dependent or transient values, etc.). If this approach is selected, all RCS activity allowable limits should be included.*

For 10 CFR 50.47(b)(4), Emergency Classification System, Reg. Guide 1.219 states that the following examples would generally not require prior NRC approval:
(1) A change to an EAL numeric threshold to reflect an approved change in a technical specification, provided that the basis of the approved EAL is unchanged (e.g., an EAL basis refers to a particular technical specification but not a limiting condition for operation value), and (2) A change to an EAL numeric threshold to reflect a change in a plant design parameter, instrument response characteristics, or design calculation, provided that the meaning or intent of the basis of the approved EAL is unchanged.

The proposed change complies with 10 CFR 50 Appendix E, Regulatory Guide 1.219, Revision 1, 10 CFR 50.47, and with industry guidance in NEI 99-01, Revision 6.

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10CFR50.54(q) SUMMARY ANALYSIS REPORT
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Revision 0

50.54Q I.D. Number: **2022-03**

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EP-SA-325-141, Rev. 2, EAL Wall Chart – Hot Conditions

(Doc #, Rev. #, Name, If applicable)

A description of why the proposed change was not a reduction in the effectiveness of the Emergency Plan/Procedure:

The proposed revision aligns with Technical Specification Amendment Nos. 337 and 318 (LAR S20-01) that affects Technical Specifications for both Salem U1 (TS 3.4.8) and Salem U2 (TS 3.4.9). The EAL change will require training be provided to Operations personnel through the Licensed Operator Training program. Training for the Emergency Coordinators in the TSC and EOF and their Direct Reports will be provided through Emergency Preparedness Focus Area Drills (FADs).

There is no reduction in effectiveness to the Emergency Plan resulting from the proposed change to EAL SU4.2, as defined in EP-SA-325-117, EP-SA-325-217, and EP-SA-325-141.

ATTACHMENT 2

EP-SA-325-141
Salem Emergency Classification Guide Wall Chart

		GENERAL EMERGENCY Implement Att. 4	SITE AREA EMERGENCY Implement Att. 3	ALERT Implement Att. 2	UNUSUAL EVENT Implement Att. 1 (Att. 24 for Common Site)
S System Malfunct.	**1** Loss of AC Power	Prolonged loss of **ALL** offsite and **ALL** onsite AC power to vital buses **SG1.1** 1 2 3 4 Loss of **ALL** offsite and **ALL** onsite AC power to 4 KV vital buses **AND** **EITHER** of the following: • Restoration of at least one vital bus in < 4 hrs is **NOT** likely (Note 1) • CFST Core Cooling **RED** path conditions met	Loss of **ALL** offsite power and **ALL** onsite AC power to vital buses for 15 minutes or longer **SS1.1** 1 2 3 4 Loss of **ALL** offsite and **ALL** onsite AC power to 4 KV vital buses for **≥ 15 min.** (Note 1)	Loss of **ALL but one** AC power source to vital buses for 15 minutes or longer **SA1.1** 1 2 3 4 AC power capability to 4 KV vital buses reduced to a single power source for **≥ 15 min.** (Note 1) **AND** **ANY** additional single power source failure will result in loss of **ALL** AC power to **SAFETY SYSTEMS**	Loss of **ALL** offsite AC power capability to vital buses for 15 minutes or longer **SU1.1** 1 2 3 4 Loss of **ALL** offsite AC power to 4 KV vital buses for **≥ 15 min.** (Note 1)
	2 Loss of DC Power	Loss of **ALL** vital AC and vital DC power sources for 15 minutes or longer **SG2.1** 1 2 3 4 Loss of **ALL** offsite and **ALL** onsite AC power to 4 KV vital buses for **≥ 15 min.** **AND** **EITHER:** • **< 114 VDC** bus voltage indications on **ALL** 125 VDC vital buses for **≥ 15 min.** • **< 25 VDC** bus voltage indications on **both** 28 VDC vital buses for **≥ 15 min.** (Note 1)	Loss of **ALL** vital DC power for 15 minutes or longer **SS2.1** 1 2 3 4 **< 114 VDC** bus voltage indications on **ALL** 125 VDC vital buses for **≥ 15 min.** **OR** **< 25 VDC** bus voltage indications on **both** 28 VDC vital buses for **≥ 15 min.** (Note 1)	None	None
	3 Loss of CR Indications	None		**UNPLANNED** loss of Control Room indications for 15 minutes or longer with a significant transient in progress **SA3.1** 1 2 3 4 An **UNPLANNED** event results in the inability to monitor one or more **Table S-1** parameters from within the Control Room for **≥ 15 min.** (Note 1) **AND** **ANY** significant transient is in progress, **Table S-2**	**UNPLANNED** loss of Control Room indications for 15 minutes or longer **SU3.1** 1 2 3 4 An **UNPLANNED** event results in the inability to monitor one or more **Table S-1** parameters from within the Control Room for **≥ 15 min.** (Note 1)
	4 RCS Activity	None	None		Reactor coolant activity greater than Technical Specification allowable limits **SU4.1** 1 2 3 4 Letdown Line Monitor readings indicating fuel clad degradation based on receipt of **EITHER** of the following (Note 11): • 1R31A in warning • 2R31 in alarm **SU4.2** 1 2 3 4 Reactor coolant activity **> ANY**: (Note 11) • **60 µCi/gram** DOSE EQUIVALENT I-131 • **1.0 µCi/gram** DOSE EQUIVALENT I-131 for **>** 48 hrs. • **600 µCi/gram** DOSE EQUIVALENT XE-133 for **>** 48 hrs.
	5 RCS Leakage	None	None	None	RCS leakage for 15 minutes or longer **SU5.1** 1 2 3 4 RCS **UNIDENTIFIED** or **PRESSURE BOUNDARY LEAKAGE > 10 gpm** for **≥ 15 min.** **OR** RCS **IDENTIFIED LEAKAGE > 25 gpm** for **≥ 15 min.** **OR** Leakage from the RCS to a location outside containment **> 25 gpm** for **≥ 15 min.** (Notes 1, 11)
	6 RPS Failure	None	Inability to shutdown the reactor causing a challenge to RCS water level or RCS heat removal **SS6.1** 1 2 An automatic or manual trip did **NOT** shut down the reactor as indicated by reactor power **≥ 5%** **AND** **ALL** actions to shut down the reactor are **NOT** successful as indicated by reactor power **≥ 5%** **AND** **EITHER:** • CFST Core Cooling **RED** path conditions met • CFST Heat Sink **RED** path exists due to actual loss of secondary heat sink and heat sink is required	Automatic or manual trip fails to shut down the reactor and subsequent manual actions taken at the reactor control consoles are **NOT** successful in shutting down the reactor **SA6.1** 1 2 An automatic or manual trip did **NOT** shut down the reactor as indicated by reactor power **≥ 5%** **AND** Manual trip actions taken at the reactor control console (reactor trip switches, trip bkr bezels, supply breakers 1/2E6D and 1/2G6D) are **NOT** successful in shutting down the reactor as indicated by reactor power **≥ 5%** (Note 8)	Automatic or manual trip fails to shut down the reactor **SU6.1** 1 2 An automatic or manual trip did **NOT** shut down the reactor after **ANY** RPS setpoint is exceeded or a manual trip action was initiated **AND** A subsequent automatic trip or manual trip action taken at the reactor control console (reactor trip switches, trip bkr bezels, supply breakers 1/2E6D and 1/2G6D) is successful in shutting down the reactor as indicated by reactor power **< 5%** (Note 8)
	7 Loss of Commun.	None	None		Loss of **ALL** onsite or offsite communications capabilities **SU7.1** 1 2 3 4 Loss of **ALL Table S-3** onsite communication methods **OR** Loss of **ALL Table S-3** offsite communication methods **OR** Loss of **ALL Table S-3** NRC communication methods
	8 CMT Failure			None	Failure to isolate containment or loss of containment pressure control **SU8.1** 1 2 3 4 **ANY** penetration is not isolated within **15 min.** of a **VALID** containment isolation signal **OR** Containment pressure **> 15 psig** with **< one full train of containment depressurization equipment** operating per design for **≥ 15 min.** (Notes 1,10)
	9 Hazardous Event Affecting Safety Systems			Hazardous event affecting **SAFETY SYSTEMS** needed for the current operating mode **SA9.1** 1 2 3 4 The occurrence of **ANY Table S-4** hazardous event **AND** Event damage has caused indications of degraded performance on one train of a **SAFETY SYSTEM** needed for the current operating mode **AND** **EITHER:** • Event damage has caused indications of degraded performance on the second train of the **SAFETY SYSTEM** needed for the current operating mode • Event damage has resulted in **VISIBLE DAMAGE** to the second train of the **SAFETY SYSTEM** needed for the current operating mode (Notes 12, 13)	

Table S-2 Significant Transients

- Automatic turbine runback > 25% thermal reactor power
- Electrical load rejection > 25% full electrical load
- Reactor Trip
- Safety Injection Activation

Table S-1 Safety System Parameters

- Reactor power
- RCS level
- RCS pressure
- CET temperature
- Level in at least one SG
- Auxiliary or emergency feedwater flow to at least one SG

Table S-3 Communications Methods

System	Onsite	Offsite	NRC
Direct Inward Dial System (DID)	X	X	X
Station Page System (Gaitronics)	X		
Station Radio System	X		
Nuclear Emergency Telephone System (NETS)		X	X
Centrex Phone System (ESSX)		X	X
NRC (ENS)			X

Table S-4 Hazardous Events

- Seismic event (earthquake)
- Internal or external **FLOODING** event
- High winds or tornado strike
- **FIRE**
- **EXPLOSION**
- Other events with similar hazard characteristics as determined by the Shift Manager

NOTES

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 8: A manual trip action is **ANY** operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does **NOT** include manually driving in control rods or implementation of boron injection strategies.

Note 10: One full train of depressurization equipment consists of **EITHER**:
- at least 5 CFCUs running in low speed with **NO** Containment Spray train in service
- at least 3 CFCUs running in low speed with one Containment Spray train in service

Note 11: Refer to the Fission Product Barrier Table for possible event escalation due to RCS leakage or high RCS activity.

Note 12: If the affected **SAFETY SYSTEM** train was already inoperable or out of service before the hazardous event occurred, then emergency classification is **NOT** warranted.

Note 13: If the hazardous event **ONLY** resulted in **VISIBLE DAMAGE**, with **NO** indications of degraded performance to at least one train of a **SAFETY SYSTEM**, then this emergency classification is **NOT** warranted.

Use of Fission Product Barrier Table

MODEs 1 2 3 4

A point system is used to determine the Emergency Classification Level based on the Fission Product Barrier Table. Each Fission Product Barrier Loss and Potential Loss threshold is assigned a point value as noted below.
Perform the following:

1. Review all columns of the Fission Product Barrier Table and identify which need further review.
2. For each of the three barriers, determine the EAL with the highest point value. No more than one EAL should be selected for each barrier.
3. Add the point values for the three barriers.
4. Classify based on the point value sum as follows:

If sum is:	Classify as:	Emergency Action Levels (EALs)	Refer to ECG ATT#
4, 5	ALERT	**ANY** loss or **ANY** potential loss of either Fuel Clad or RCS	2
6 - 11	SITE AREA EMERGENCY	Loss or potential loss of **ANY** two barriers, **OR** Potential loss of 2 barriers with the loss of the 3rd barrier	3
12 or 13	GENERAL EMERGENCY	Loss of **ANY** two barriers **AND** Loss or potential loss of the third barrier	4

5. Implement the appropriate ECG Attachment per above table.
6. Continue to review the Fission Product Barrier Table for changes that could result in emergency escalation or de-escalation.

Salem – Fission Product Barrier Table

FPB Category	Fuel Clad Barrier – Potential Loss (4 pts)	Fuel Clad Barrier – Loss (5 pts)	Reactor Coolant System Barrier – Potential Loss (4 pts)	Reactor Coolant System Barrier – Loss (5 pts)	Containment Barrier – Potential Loss (2 pts)	Containment Barrier – Loss (3 pts)
RCS or SG Tube Leakage	None	None	**RB1.P** RCS leakage **> 50 gpm** due to **EITHER:** • **UNISOLABLE** RCS leakage • SG tube leakage **RB2.P** CFST Thermal Shock **RED** path conditions met	**RB1.L** An automatic or manual ECCS (SI) actuation required by **EITHER:** • **UNISOLABLE** RCS leakage • SG tube **RUPTURE**	None	**CB1.L** A leaking or **RUPTURED** SG is **FAULTED** outside of containment
Inadequate Heat Removal	**FB1.P** CFST Core Cooling **PURPLE** path conditions met **FB2.P** CFST Heat Sink **RED** path exists due to actual loss of secondary heat sink and heat sink is required	**FB1.L** CFST Core Cooling **RED** path conditions met	**RB3.P** CFST Heat Sink **RED** path exists due to actual loss of secondary heat sink and heat sink is required	None	**CB1.P** CFST Core Cooling **RED** path conditions met **AND** Restoration procedure 1(2)EOP-FRCC-1 **NOT** effective within **15 min.**	None
CMT Radiation / RCS Activity	None	**FB2.L** Containment radiation monitor 1(2)R44A or 1(2)R44B reading **> 300 R/hr** **FB3.L** Coolant activity **> 300 µCi/gm dose equivalent I-131**	None	**RB2.L** **ANY** of the following containment radiation monitor readings: • 1(2)R2 **> 1000 mR/hr** • 1(2)R44A **> 10 R/hr** • 1(2)R44B **> 10 R/hr**	**CB2.P** Containment radiation monitor 1(2)R44A or 1(2)R44B reading **> 2000 R/hr**	None
CMT Integrity or Bypass	None	None	None	None	**CB3.P** CFST Containment **RED** path conditions met **CB4.P** Containment hydrogen concentration **> 4%** **CB5.P** Containment pressure **> 15 psig** with **< one full train of containment depressurization equipment** operating per design for **≥ 15 min.** (Notes 1, 10)	**CB2.L** Containment isolation is required **AND EITHER:** • Containment integrity has been lost based on Emergency Coordinator judgment • **UNISOLABLE** pathway from containment to the environment exists **CB3.L** Indications of RCS leakage outside of containment
EC Judgment	**FB3.P** **ANY** condition in the opinion of the Emergency Coordinator that indicates potential loss of the Fuel Clad barrier	**FB4.L** **ANY** condition in the opinion of the Emergency Coordinator that indicates loss of the Fuel Clad barrier	**RB4.P** **ANY** condition in the opinion of the Emergency Coordinator that indicates potential loss of the RCS barrier	**RB3.L** **ANY** condition in the opinion of the Emergency Coordinator that indicates loss of the RCS barrier	**CB6.P** **ANY** condition in the opinion of the Emergency Coordinator that indicates potential loss of the Containment barrier	**CB4.L** **ANY** condition in the opinion of the Emergency Coordinator that indicates loss of the Containment barrier

NOTES

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 10: One full train of depressurization equipment consists of **EITHER**:
- at least 5 CFCUs running in low speed with **NO** Containment Spray train in service
- at least 3 CFCUs running in low speed with one Containment Spray train in service

Modes: 1 Power Operations | 2 Startup | 3 Hot Standby | 4 Hot Shutdown

EAL WALL CHART - HOT CONDITIONS (RCS > 200°F)

SALEM GENERATING STATION

EAL WALL CHART (HOT)

EP-SA-325-141

Revision 02

ATTACHMENT 3

EP-SA-325-117, Salem Section S
S4 – RCS Activity

Section S – System Malfunction
S4 – RCS Activity

Initiating Condition	Reactor coolant activity greater than Technical Specification allowable limits	
MODE	1, 2, 3, 4	
EAL #	**SU4.1**	**SU4.2**
	IF	**IF**
EMERGENCY ACTION LEVELS	Letdown Line Monitor readings indicating fuel clad degradation based on receipt of **EITHER** of the following: • 1R31A in warning • 2R31 in alarm (Note 11)	Reactor coolant activity **>** **ANY**: • **60 µCi/gram** DOSE EQUIVALENT I-131 • **1.0 µCi/gram** DOSE EQUIVALENT I-131 for **>** 48 hrs. • **600 µCi/gram** DOSE EQUIVALENT XE-133 for **>** 48 hrs. (Note 11)
	THEN	**THEN**
Action Required	Refer to Attachment 1 **UNUSUAL EVENT**	Refer to Attachment 1 **UNUSUAL EVENT**

Note 11: Refer to the Fission Product Barrier Table for possible event escalation due to RCS leakage or high RCS activity

ATTACHMENT 4

EP-SA-325-217, Salem Section S
EAL Technical Basis

EAL Category: S – System Malfunction

EAL Subcategory: 4 – RCS Activity

Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits

Mode Applicability: 1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

EAL# & Classification Level: **SU4.1 – UNUSUAL EVENT**

EAL:

Letdown Line Monitor readings indicating fuel clad degradation based on receipt of **EITHER** of the following (Note 11):

- 1R31A in warning
- 2R31 in alarm

Note 11: Refer to the Fission Product Barrier Table for possible event escalation due to RCS leakage or high RCS activity.

Basis:

This EAL addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

Escalation of the emergency classification level would be via category F or category R ICs.

Explanation/Discussion/Definitions:

Letdown Line Monitors serve as a failed fuel detector by monitoring gamma levels in the reactor coolant letdown line. Unit 1 Letdown Line Monitor (1R31A) and Unit 2 Letdown Line Monitor (2R31) measures letdown line activity. The Letdown Line Monitor “warning” setpoints are administratively set at 50% of the “alarm” setpoints.

- 1R31A “alarm” setpoint is based on 1% failed fuel. The “warning” setpoint represents about 0.5% failed fuel and has been selected because the setpoint would be readily identifiable on Control Room instrumentation.

- 2R31 “alarm” setpoint is based on 0.1% failed fuel. This setpoint is readily identifiable and also representative of typical values of coolant activity at Technical Specification limits.

Read-outs for these monitors can be obtained in the Control Room.

Other radiation monitors that may be used to confirm a **VALID** Letdown Line Monitor alarm include:

- 1(2)R4 Charging Pump Room
- 1(2)R26 Reactor Coolant Filter
- Containment Area Rad Monitors (1(2)R2, 1(2)7, 1(2)10A, 1(2)10B)

Definitions:

VALID: An indication, report, or condition, is considered to be valid when it is verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator’s operability, the condition’s existence, or the report’s accuracy is removed. Implicit in this definition is the need for timely assessment

EAL Bases Reference(s):

1. NEI 99-01, Rev. 06, SU3 Example EAL #1
2. PSBP 315733 Radiation Monitoring System Manual, Unit 1
3. PSBP 315734 Radiation Monitoring System Control Manual, Unit 2
4. UFSAR 9.3.5.3 Safety Evaluation (Failed fuel Detection System)
5. UFSAR 11.4 Radiological Monitoring
6. S1(S2).OP-AB.RC-0002 (Q) High Activity in the Reactor Coolant System

EAL Category: S – System Malfunction

EAL Subcategory: 4 – RCS Activity

Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits

Mode Applicability: 1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

EAL# & Classification Level: **SU4.2 – UNUSUAL EVENT**

EAL:

Reactor coolant activity **>** **ANY**:

- **60 μCi/gram** DOSE EQUIVALENT I-131
- **1.0 μCi/gram** DOSE EQUIVALENT I-131 for **>** 48 hrs.
- **600 μCi/gram** DOSE EQUIVALENT XE-133 for **>** 48 hrs.

(Note 11)

Note 11: Refer to the Fission Product Barrier Table for possible event escalation due to RCS leakage or high RCS activity.

Basis:

This EAL addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

Escalation of the emergency classification level would be via category F or Category R ICs.

Explanation/Discussion/Definitions:

An **UNUSUAL EVENT** is only warranted when actual fuel clad damage is the cause of the elevated coolant sample (as determined by RCS sample analysis confirmation).

Escalation to an **ALERT** or higher emergency classification occurs if a sample analysis of reactor coolant activity exceeds 300 μCi/gm DEI-131 via fission product barrier monitoring.

EAL Bases Reference(s):

1. NEI 99-01, Rev. 06, SU3, Example EAL #2

2. SGS Technical Specification Section 3.4.8 - Unit 1 Specific Activity
3. SGS Technical Specification Section 3.4.9 - Unit 2 Specific Activity
4. S1(S2).OP-AB.RC-0002(Q) High Activity in Reactor Coolant System

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