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CAUTION

- Do **NOT** Reduce RCS level Below Elv 39 ft. 4.9 in. using this Addendum.
- As pressurizer level lowers to 10% <u>AND</u> before the PZR level goes off scale low, COMPARE Pressurizer level with RC-LG-3662 "RCS LEVEL SIGHTGLASS (SLINKY)" level indication.
- 3.0 PERFORM the following to lower RCS inventory:
 - 3.1 The following instrumentation SHALL be operable prior to RCS Draindown to 39' 4.9":
 - 3.1.1 RHR heat exchanger inlet and outlet temperature with indication on QDPS OR chart recorder for all operable trains.
 - 3.1.2 RHR pump flow with indication on QDPS for all operable trains.
 - 3.1.3 RHR pump motor current indication (amps) for all operable trains.
 - 3.1.4 "RHR PUMP CURRENT LO" Annunciators on Lampbox 1M02 for the operable RHR pumps is **NOT** removed from service.
 - 3.1.5 RCS level sightglass has been walked down in last twelve hours and satisfies requirements.
 - 3.1.6 Both trains of RVWL are operable with at least two QDPS displays available.
 - 3.1.7 Core Exit Thermocouples, five per train, two trains.
 - 3.2 ENSURE Adequate capacity is available in radwaste to receive volume drained from RCS. REFER TO Addendum 3, Determination of RCS Volume to be Drained.
 - 3.3 A dedicated, reliable communication line, headphones being the preferred method, is established between Control Room personnel and RCB sightglass watch.

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	<u>NOTE</u>
•	Venting and draining operations should be coordinated with the Radwaste Operator and Health Physics.
	RCS level changes SHALL be made slowly in a controlled manner to minimize effects on reactor vessel level indications.
	The RCS level sightglass SHALL be continuously monitored during all draining and refill operations.
•	One of the following pressurizer vent paths SHALL be established prior to draining the pressurizer:
	• The Pressurizer spray line vent valves RC-0502 and RC-0503 OPEN to atmosphere. (Preferred Method)
	• A minimum of one Pressurizer Code Safety Valve REMOVED.
•	In addition to the specified temperature limits, the intent is to maintain RCS temperature as low as allowed by existing plant conditions, core cooling capabilities or other limiting criteria to maximize margin to core boiling:
	• (Mode 5) MAINTAIN RCS core exit temperature (RHR HX inlet temp when CETs NOT available) less than 140°F.

3.4 IF PZR level wll be lowered below 10% Cold Calibrated level, <u>THEN</u> ENSURE RCS level sightglass is in service.

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<u>NOTE</u>

<u>WHEN</u> RC-LG-3662 "RCS LEVEL SIGHTGLASS (SLINKY)" is in service, <u>THEN</u> ENSURE "0POP07-RC-0001, RC Vent Rig/Sightglass Installation and Removal", LINEUP 1, "RC-LG-3662 "RCS LEVEL SIGHTGLASS (SLINKY)" in Service Lineup" PERFORMED daily and DOCUMENTED in a temporary log.

3.5 COMMENCE temporary logging of LG-3662 sightglass valves once per shift (Ref. Procedure Step 3.37).

CAUTION

Prior to draining the pressurizer, a vent path SHALL be established to prevent drawing a vacuum.

- 3.6 <u>IF</u> lowering pressurizer level to below 10% Cold Calibrated level (55 ft 6 inch elevation), <u>THEN</u> PERFORM the following:
 - 3.6.1 ENSURE RCS level sightglass is being monitored.
 - 3.6.2 ENSURE pressurizer vent path established.
 - 3.6.3 DOCUMENT Vent Path.
 - 3.6.4 OPEN PZR Spray Valves RC-PCV-655B and RC-PCV-655C.
 - 3.6.5 DETERMINE RCS volume to be drained using Addendum 3, Determination of RCS Volume to be Drained.
 - 3.6.6 VERIFY "DIVERT LCV-0112A" in the AUTO position. {CP004}
 - 3.6.7 Manually RAISE letdown flow using "PRESS CONT PCV-0135". {CP004}
 - 3.6.8 COMPARE Pressurizer level with RCS level sightglass level indication (Should agree within 6 inches. 1% of Pressurizer Cold Calibrated level equals 4 inches).

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CAUTION

<u>IF</u> RVWL system indicates less than 100%, <u>THEN</u> the draindown SHALL be stopped and the level difference between RVWL and RCS level sightglass investigated.

- 3.6.9 IF pressurizer Cold Calibrated level <u>AND</u> RCS level sightglass do <u>NOT</u> agree within 6 inches, <u>THEN</u> STOP the drain and REFER TO Addendum 15, Rx Head Venting and RCS level Instruments Disagreements Evaluation Guideline.
- 3.6.10 <u>IF</u> pressurizer level will be maintained, <u>THEN</u> REFER TO Step 3.22 of this Addednum.

CAUTION

<u>IF</u> during any RCS draining process, fluctuations are observed in RHR pump flow, amps, <u>OR</u> discharge pressure, <u>THEN</u> any RCS drain in progress SHALL be stopped to allow RCS water level to stabilize and any RCS water level recovery SHALL be initiated as necessary to ensure RHR system operation.

- 3.7 PLACE Reactor Vessel head temperature on trend display. (Plant Computer points IITE2040 and IITE3040)
- 3.8 <u>IF</u> lowering of RCS level is to continue, <u>THEN</u> DRAIN to between 47 ft. 4 in. and 46 ft. 5 in. elevation as indicated on RC-LG-3662 "RCS LEVEL SIGHTGLASS (SLINKY)".

This procedure, when completed, SHALL be retained.

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<u>NOTE</u>

<u>WHEN</u> RVWL Sensor Point 1 has been uncovered, <u>THEN</u> indicated temperature will rise to about 750°F due to heating from the heated junction thermocouple.

- 3.9 VERIFY water in reactor vessel head less than 180°F as indicated by Plant Computer display RC12 (8112).
- 3.10 PLACE the Reactor Vessel head to pressurizer equalizing line in service as follows:
 - 3.10.1 ENSURE the reactor vessel head venting manifold is connected per 0POP02-RC-0003, Addendum 1, Filling and Venting the RCS. {RCB on RV Head}
 - 3.10.2 ENSURE the RV to PZR Equalizing Line is aligned IAW 0POP03-ZG-0009, Mid-Loop Operation.

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Addend	um 19	Controlling RCS Inventory at or above Elv 39 ft. 4.9 in.	Page 8 of 10
3.11		RCS level is between 47 ft. 4 in. and 46 ft. 5 in. as indicated o 3662 "RCS LEVEL SIGHTGLASS (SLINKY)", <u>THEN</u> PER wing:	
	3.11.1	OPEN the following valves to remove water plug (Loop Sea Head and Pressurizer vent manifold and Rx Head Vent line	
		 "1(2)-RC-0509 RX VESSEL HEAD" "VENTING MANIFOLD DRAIN" {RCB On RV Head} 	
		 "1(2)-RC-0507 RX VESSEL HEAD" "VENTING MANIFOLD VENT VALVE" {RCB On RV Head} 	
	3.11.2	IF the reactor vessel head vent valves are operable, THEN Reactor Vessel head vent valves. {CP005}	OPEN the
		• "ISOL HV-3657A"	
		• "ISOL HV-3657B"	
		• "ISOL HV-3658A"	
		• "ISOL HV-3658B"	
		• "HEAD VENT THROT VLV HCV-0601"	
		• "HEAD VENT THROT VLV HCV-0602"	
	3.11.3	IF desired, <u>THEN</u> OPEN PRT "1(2)-RC-0025 N2 SUPPLY {RCB 6 ft E of PRT}	ISOL".
	3.11.4	IF RCS level will be maintained, THEN REFER TO Step 3	22

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- 3.12 IF RCS level is to remain below 47 ft. 4 in, <u>THEN</u> PERFORM the following:
 - CLOSE "1(2)-RC-0507 RX VESSEL HEAD" "VENTING MANIFOLD VENT VALVE" {RCB On RV Head}
 - ENSURE "1(2)-RC-0509 RX VESSEL HEAD" "VENTING MANIFOLD DRAIN" {RCB On RV Head} remains OPEN to vent the HEAD.

<u>NOTE</u>

- The temperature rise will occur when sensor is uncovered prior to RVWL point indicating dry.
- <u>WHEN</u> RVWL Sensor Point 1 has been uncovered, <u>THEN</u> indicated temperature will rise to about 750°F due to heating from the heated junction thermocouple.
- <u>WHEN</u> RC-LG-3662 "RCS LEVEL SIGHTGLASS (SLINKY)" is in service, <u>THEN</u> ENSURE "0POP07-RC-0001, RC Vent Rig/Sightglass Installation and Removal", LINEUP 1, "RC-LG-3662 "RCS LEVEL SIGHTGLASS (SLINKY)" in Service Lineup" PERFORMED daily and DOCUMENTED in a temporary log.

CAUTION

Inadequate head venting due to rapid draining can cause Reactor Vessel water level to remain higher than loop level.

- 3.13 <u>IF</u> lowering of RCS level is to continue, <u>THEN</u> DRAIN RCS until RVWL Sensor Point 1 (45' 3.4") heated junction thermocouple temperature rises by 20°F. (Plant Computer Points IITE2004, A-Train, IITE3004, C-Train)
- 3.14 COMPARE RCS level sightglass indication with RVWL level.
- 3.15 IF RVWL AND RCS level sightglass do <u>NOT</u> agree within 6 inches, <u>THEN</u> STOP the drain and REFER TO Addendum 15, Rx Head Venting and RCS level Instruments Disagreements Evaluation Guideline.
- 3.16 IF RCS level will be maintained, THEN REFER TO Step 3.22.

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3.17 REFER TO Addendum 3, Determination of RCS Volume to be Drained, to determine draindown volume.

 The temperature rise will occur when the sensor is uncovered prior to RVWL point indicating dry. <u>WHEN</u> RC-LG-3662 "RCS LEVEL SIGHTGLASS (SLINKY)" is in service, <u>THEN</u> ENSURE "0POP07-RC-0001, RC Vent Rig/Sightglass Installation and Removal", LINEUP 1, "RC-LG-3662 "RCS LEVEL SIGHTGLASS (SLINKY)" in Service Lineup PERFORMED daily and DOCUMENTED in a temporary log. Do <u>NOT</u> Reduce RCS level Below Elv 39 ft. 4.9 in. using this Addendum. 3.18 <u>IF</u> lowering of RCS level is to continue, <u>THEN</u> DRAIN RCS until RVWL Sensor Point 2 (39' 4.9") heated junction thermocouple temperature rises by 20°F. (Plant Computer Points IITE2007, A-Train, IITE3007, C-Train) 3.19 COMPARE RCS level sightglass indication with RVWL level. 3.20 <u>IF</u> RVWL <u>AND</u> RCS level sightglass do <u>NOT</u> agree within 6 inches, <u>THEN</u> STOP the drain and REFER TO Addendum 15, Rx Head Venting and RCS level Instruments Disagreements Evaluation Guideline. 3.21 ADJUST RHR HX flow as required to maintain RHR HX inlet temperature less than 140°F. 3.22 MAINTAIN RCS level as necessary using gravity drain from RWST through the LHSI pump cold leg injection valves in the idle RHR train Raise RCS level as necessary using CCP normal charging or seal injection Reduce RCS level as necessary using low pressure letdown 3.23 MAINTAIN temporary log 0POP07-RC-0001, LINEUP 1, "RC-LG-3662 RCS LEVEL SIGHTGLASS (SLINKY)" in Service Lineup as required. (Ref. Procedure Step 3.37) 		NOTE
 WHEN RC-LG-3662 "RCS LEVEL SIGHTGLASS (SLINKY)" is in service, <u>THEN</u> ENSURE "0POP07-RC-0001, RC Vent Rig/Sightglass Installation and Removal", LINEUP 1, "RC-LG-3662 "RCS LEVEL SIGHTGLASS (SLINKY)" in Service Lineup PERFORMED daily and DOCUMENTED in a temporary log. Do <u>NOT</u> Reduce RCS level Below Elv 39 ft. 4.9 in. using this Addendum. 3.18 IF lowering of RCS level is to continue, <u>THEN</u> DRAIN RCS until RVWL Sensor Point 2 (39' 4.9") heated junction thermocouple temperature rises by 20°F. (Plant Computer Points IITE2007, A-Train, IITE3007, C-Train) 3.19 COMPARE RCS level sightglass indication with RVWL level. 3.20 IF RVWL <u>AND</u> RCS level sightglass do <u>NOT</u> agree within 6 inches, <u>THEN</u> STOP the drain and REFER TO Addendum 15, Rx Head Venting and RCS level Instruments Disagreements Evaluation Guideline. 3.21 ADJUST RHR HX flow as required to maintain RHR HX inlet temperature less than 140°F. 3.22 MAINTAIN RCS level using any of the following: Raise RCS level as necessary using gravity drain from RWST through the LHSI pump cold leg injection valves in the idle RHR train Raise RCS level as necessary using low pressure letdown 3.23 MAINTAIN temporary log 0POP07-RC-0001, LINEUP 1, "RC-LG-3662 RCS LEVEL SIGHTGLASS (SLINKY)" in Service Lineup as required. (Ref. Procedure Step 3.37) 		emperature rise will occur when the sensor is uncovered prior to RVWL point
 3.18 IF lowering of RCS level is to continue, <u>THEN</u> DRAIN RCS until RVWL Sensor Point 2 (39' 4.9") heated junction thermocouple temperature rises by 20°F. (Plant Computer Points IITE2007, A-Train, IITE3007, C-Train) 3.19 COMPARE RCS level sightglass indication with RVWL level. 3.20 IF RVWL AND RCS level sightglass do <u>NOT</u> agree within 6 inches, <u>THEN</u> STOP the drain and REFER TO Addendum 15, Rx Head Venting and RCS level Instruments Disagreements Evaluation Guideline. 3.21 ADJUST RHR HX flow as required to maintain RHR HX inlet temperature less than 140°F. 3.22 MAINTAIN RCS level using any of the following: Raise RCS level as necessary using gravity drain from RWST through the LHSI pump cold leg injection valves in the idle RHR train Raise RCS level as necessary using low pressure letdown 3.23 MAINTAIN temporary log 0POP07-RC-0001, LINEUP 1, "RC-LG-3662 RCS LEVEL SIGHTGLASS (SLINKY)" in Service Lineup as required. (Ref. Procedure Step 3.37) 	ENS LINE PER	URE "0POP07-RC-0001, RC Vent Rig/Sightglass Installation and Removal", EUP 1, "RC-LG-3662 "RCS LEVEL SIGHTGLASS (SLINKY)" in Service Lineup" FORMED daily and DOCUMENTED in a temporary log.
 Sensor Point 2 (39' 4.9") heated junction thermocouple temperature rises by 20°F. (Plant Computer Points IITE2007, A-Train, IITE3007, C-Train) 3.19 COMPARE RCS level sightglass indication with RVWL level. 3.20 IF RVWL AND RCS level sightglass do NOT agree within 6 inches, THEN STOP the drain and REFER TO Addendum 15, Rx Head Venting and RCS level Instruments Disagreements Evaluation Guideline. 3.21 ADJUST RHR HX flow as required to maintain RHR HX inlet temperature less than 140°F. 3.22 MAINTAIN RCS level using any of the following: Raise RCS level as necessary using gravity drain from RWST through the LHSI pump cold leg injection valves in the idle RHR train Raise RCS level as necessary using CCP normal charging or seal injection Reduce RCS level as necessary using low pressure letdown 3.23 MAINTAIN temporary log 0POP07-RC-0001, LINEUP 1, "RC-LG-3662 RCS LEVEL SIGHTGLASS (SLINKY)" in Service Lineup as required. (Ref. Procedure Step 3.37) 	• Do <u>f</u>	NOT Reduce RCS level Below Elv 39 ft. 4.9 in. using this Addendum.
 3.20 IF RVWL AND RCS level sightglass do NOT agree within 6 inches, THEN STOP the drain and REFER TO Addendum 15, Rx Head Venting and RCS level Instruments Disagreements Evaluation Guideline. 3.21 ADJUST RHR HX flow as required to maintain RHR HX inlet temperature less than 140°F. 3.22 MAINTAIN RCS level using any of the following: Raise RCS level as necessary using gravity drain from RWST through the LHSI pump cold leg injection valves in the idle RHR train Raise RCS level as necessary using CCP normal charging or seal injection Reduce RCS level as necessary using low pressure letdown 3.23 MAINTAIN temporary log 0POP07-RC-0001, LINEUP 1, "RC-LG-3662 RCS LEVEL SIGHTGLASS (SLINKY)" in Service Lineup as required. (Ref. Procedure Step 3.37) 	3.18	Sensor Point 2 (39' 4.9") heated junction thermocouple temperature rises by
 STOP the drain and REFER TO Addendum 15, Rx Head Venting and RCS level Instruments Disagreements Evaluation Guideline. 3.21 ADJUST RHR HX flow as required to maintain RHR HX inlet temperature less than 140°F. 3.22 MAINTAIN RCS level using any of the following: Raise RCS level as necessary using gravity drain from RWST through the LHSI pump cold leg injection valves in the idle RHR train Raise RCS level as necessary using CCP normal charging or seal injection Reduce RCS level as necessary using low pressure letdown 3.23 MAINTAIN temporary log 0POP07-RC-0001, LINEUP 1, "RC-LG-3662 RCS LEVEL SIGHTGLASS (SLINKY)" in Service Lineup as required. (Ref. Procedure Step 3.37) 	3.19	COMPARE RCS level sightglass indication with RVWL level.
 than 140°F. 3.22 MAINTAIN RCS level using any of the following: Raise RCS level as necessary using gravity drain from RWST through the LHSI pump cold leg injection valves in the idle RHR train Raise RCS level as necessary using CCP normal charging or seal injection Reduce RCS level as necessary using low pressure letdown 3.23 MAINTAIN temporary log 0POP07-RC-0001, LINEUP 1, "RC-LG-3662 RCS LEVEL SIGHTGLASS (SLINKY)" in Service Lineup as required. (Ref. Procedure Step 3.37) 	3.20	STOP the drain and REFER TO Addendum 15, Rx Head Venting and RCS
 Raise RCS level as necessary using gravity drain from RWST through the LHSI pump cold leg injection valves in the idle RHR train Raise RCS level as necessary using CCP normal charging or seal injection Reduce RCS level as necessary using low pressure letdown 3.23 MAINTAIN temporary log 0POP07-RC-0001, LINEUP 1, "RC-LG-3662 RCS LEVEL SIGHTGLASS (SLINKY)" in Service Lineup as required. (Ref. Procedure Step 3.37) 	3.21	•
 LHSI pump cold leg injection valves in the idle RHR train Raise RCS level as necessary using CCP normal charging or seal injection Reduce RCS level as necessary using low pressure letdown 3.23 MAINTAIN temporary log 0POP07-RC-0001, LINEUP 1, "RC-LG-3662 RCS LEVEL SIGHTGLASS (SLINKY)" in Service Lineup as required. (Ref. Procedure Step 3.37) 	3.22	MAINTAIN RCS level using any of the following:
 Reduce RCS level as necessary using low pressure letdown 3.23 MAINTAIN temporary log 0POP07-RC-0001, LINEUP 1, "RC-LG-3662 RCS LEVEL SIGHTGLASS (SLINKY)" in Service Lineup as required. (Ref. Procedure Step 3.37) 		
3.23 MAINTAIN temporary log 0POP07-RC-0001, LINEUP 1, "RC-LG-3662 RCS LEVEL SIGHTGLASS (SLINKY)" in Service Lineup as required. (Ref. Procedure Step 3.37)		• Raise RCS level as necessary using CCP normal charging or seal injection
LEVEL SIGHTGLASS (SLINKY)" in Service Lineup as required. (Ref. Procedure Step 3.37)		• Reduce RCS level as necessary using low pressure letdown
3.24 To raise RCS level, GO TO Step 2.0 of this addendum.	3.23	LEVEL SIGHTGLASS (SLINKY)" in Service Lineup as required. (Ref.
\mathbf{r}	3.24	To raise RCS level, GO TO Step 2.0 of this addendum.
3.25 RETURN TO Procedure Step in effect.	3.25	RETURN TO Procedure Step in effect.

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Addendum 20	Addendum 20 Venting Reactor Vessel Head Using Head Vent Throttle Valve(s)			

<u>NOTE</u>

- This addendum vents a voided Reactor Vessel head when the RCS in Modes 5 or 6.
- HJTC Train "A" or Train "C" RVWL Sensor 1 indications are found at Computer Points IITE2004 and IITE3004, respectively.
- 1.0 CHECK the following indications for Reactor Vessel Head voiding:
 - Pressurizer level rising

<u>AND</u>

• VCT level constant <u>OR</u> rising

<u>AND</u>

- RVWL Sensor 1 temperature rising
- 2.0 <u>IF</u> Reactor Vessel Head voiding is indicated, <u>THEN</u> vent the Reactor Vessel Head to the PRT as follows:
 - 2.1 OPEN Head Vent Isolation Valves:
 - ISOL HV-3657A and ISOL HV-3658A

<u>OR</u>

- ISOL HV 3657B and ISOL HV-3658B
- 2.2 OPEN Head Vent Throttle Valve(s):
 - HCV-0601
 - HCV-0602

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Addendum 20	Venting Reactor Vessel Head Using Head Valve(s)	Vent Throttle	Page 2 of 2		

- 2.3 MONITOR for the following indications of the Reactor Vessel Head being vented:
 - Pressurizer level lowering

AND

• RVWL Sensor 1 temperature lowering

<u>AND</u>

- PRT pressure rising
- 2.4 <u>WHEN</u> the Reactor Vessel Head void is vented, <u>THEN</u> PERFORM the following:
 - 2.4.1 ENSURE CLOSED the Head Vent Throttle Valve:
 - HCV-0601
 - HCV-0602
 - 2.4.2 ENSURE CLOSED the Head Vent Isolation Valves:
 - HV-3657A
 - HV-3657B
 - HV-3658A
 - HV-3658B

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Addendum 21	Page 1 of 2					

- 1.0 Purpose:
 - 1.1 Provides instructions for closing Personnel Air Lock (PAL) doors to establish containment closure during work activities inside Reactor Containment Building (RCB) that require both PAL doors to be open.

2.0 <u>Instructions</u>:

- 2.1 PERFORM the following to close Personnel Air Lock doors (M-90):
 - 2.1.1 ENSURE the following conditions at RCB PAL door:
 - Door-seating surfaces are clear of all obstructions.
 - Handwheel position indication in "OPEN" position.
 - Both door latch pins fully retracted to ensure NO interference with door closure.
 - 2.1.2 PULL RCB PAL door CLOSED.
 - 2.1.3 <u>WHEN</u> RCB PAL door is flush with door jam, <u>THEN</u> ROTATE RCB PAL door hand wheel to "CLOSED" position to engage door latch pins.
 - 2.1.4 ENSURE the following conditions at MAB PAL door:
 - Door-seating surfaces are clear of all obstructions.
 - Handwheel position indication in "OPEN" position.
 - Both door latch pins fully retracted to ensure NO interference with door closure.
 - 2.1.5 PULL MAB PAL door CLOSED.
 - 2.1.6 <u>WHEN MAB PAL door is flush with door jam, THEN ROTATE</u> MAB PAL door hand wheel to "CLOSED" position to engage door latch pins.

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Addendum 21Closure of Personnel Air Lock DoorsPage 2 of 2								

<u>NOTE</u>

An ECO Caution Tag will be hanging on 1(2)-XC-0037 and MCB switch for PAL Seal OCIVs to prevent pressurizing door seals with the doors open.

- 2.2 PERFORM the following to pressurize PAL door seals:
 - 2.2.1 OPEN "1(2)-XC-0037 RCB PERSONNEL AIRLOCK SEAL AIR SUPPLY ISOLATION VALVE". (MAB 60′, Room 326)
 - 2.2.2 NOTIFY Control Room {8614, 8610, 1111(7953, 8683, 2222)} that BOTH PAL doors are in the CLOSED position AND valve 1(2)-XC-0037 is OPEN.
 - 2.2.3 REQUEST Control Room to pressurize PAL door seals by taking "PERS AIR LOCK SEAL OCIVS" switch to OPEN. (CP005)

	· · · · · ·								
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	Plant Cooldown								
	Data Sheet 1RCS and Pressurizer Cooldown RatesPage 1 of 4								
	Unit: Da	te: Initial Time:							
1.	RECORD data at	15 minute intervals.							
2.	Pressurizer cooldown rate SHALL <u>NOT</u> exceed 200°F (160°F/hr ADMIN LIMIT) in any one hour. (TRM 3.4.9.2) Cooldown rate is the actual cooldown over the hour period. (i.e.:°F _{Temperature recorded 60 minutes prior to current time $-$°F_{Temperature current} = Δ°F/hour)}								
3.	1	essure from QDPS Detail Data Menu Page 1 PT405, PT406 or PT407)							
4.	OBTAIN RCS ter	mperature from QDPS Detail Menu Page 1 (7	TE414, TE424, T	TE434, or TE444)					
5.	during cooldown Cooldown rate is	the SHALL <u>NOT</u> exceed 100°F (80°F/hr ADI within the limits of Addendum 1. (Technical the actual cooldown over the hour period. (i. $a_{ature \ current} = \Delta^{\circ}F/hour$)	Specification 3	.4.9.1) (Ref 2.70)					
6.	temperature (4). <u>I</u>	ferential temperature between the Pressurizer <u>F</u> cooldown is due to an unisolable RCS leak or equal to 250°F. Differential temperature lin s. (Ref 2.62)	, <u>THEN</u> differen	ntial temperature					
7.	INITIAL the appr	ropriate column.							
8.	15 minute cooldo	wn rate will be converted to an hourly rate.							

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Calculation: (i.e.: (°F_{Temperature recorded 15 minutes prior to current time -°F_{Temperature current}) x 4 = Δ °F/hour)}

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Plant Cooldown						
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- 9. <u>IF any readings obtained are outside the specified limits, THEN PERFORM the following:</u> (Ref 2.7)
 - 9.1 Immediately NOTIFY the Shift Manager/Unit Supervisor of the out of specification reading.
 - 9.2 STOP the plant cooldown.
 - 9.3 ENSURE RCS temperature/pressure within the specified limits from the QDPS.
 - 9.4 MAINTAIN existing RCS temperature and pressure.

<u>NOTE</u>

The cooldown SHALL **NOT** be resumed until Engineering authorizes the cooldown to be restarted.

- 9.5 NOTIFY Engineering to perform an Engineering Evaluation to determine the effects of the out-of-limit (<u>NOT</u> ADMIN LIMIT) condition on the structural integrity of the RCS or Pressurizer, as applicable.
- 10. <u>WHEN</u> Pressurizer vapor space temperature TI-0607 is <u>NOT</u> functional, <u>THEN</u> use the associated functional Pressurizer water space temperature TI-0608 for all Pressurizer temperature indications called out in this procedure. Use of the liquid temperature element alone is more conservative [will provide higher indicated change for a given actual system change] and better represents actual metal temperature. Use of the liquid temperature indication alone will provide assurance that cooldown limits will <u>NOT</u> be exceeded. (CREE 02-3367)

Example:

 Pressurizer vapor space temperature TI-0607 is non-functional, <u>THEN</u> substitute Pressurizer water space temperature TI-0608, for Pressurizer vapor space temperature TI-0607 in this procedure.

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Data Sheet 1	RCS and Pressurizer Cooldown Rates		Page 3 of 4

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INITIAL (7)

Unit:_____ Date:_____ Initial Time:_____

TIME (1)	P	RZR VAPOR SPA (10)	кСЕ	P	RZR WATER SPA	ΛСЕ	RCS PRESS QDPS (3)		RCS		PRZR-RCS WTR
	TI-0607	°F/HR (8) 15 minute rate	°F/HR (2) Rolling Hourly rate	TI-0608	°F/HR (8) 15 minute rate	°F/HR (2) Rolling Hourly rate		QDPS (4)	°F/HR (8) 15 minute rate	°F/HR (5) Rolling Hourly rate	Delta T (6)
			Ť AS				-				

Personnel Participating in cooldown:				
-	Name	Initials	Name	Initials
Cooldown completed by:	Operator	Date	Time	
Data Sheet 1 Reviewed by:	Shift Manager/Unit Supervisor	Date		

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	Plant Cooldown		
Data Sheet 1	RCS and Pressurizer Cooldown Rates		Page 4 of 4

Unit:_____ Date:_____ Initial Time:_____

Page___ of ___

(CONTINUATION SHEET)

TIME (1)	PR	ZR VAPOR SPA (10)	ACE	PRZ	R WATER SP	PACE	RCS PRESS QDPS (3)		RCS			INITIAL (7)
	TI-0607	°F/HR (8) 15 minute rate	°F/HR (2) Rolling Hourly rate	TI-0608	°F/HR (8) 15 minute rate	°F/HR (2) Rolling Hourly rate		QDPS (4)	°F/HR (8) 15 minute rate	°F/HR (5) Rolling Hourly rate	Delta T (6)	
1 1 1											- 	
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	Plant Cooldown		
Lineup 1	RV to PZR Equalizing Line I	Lineup	Page 1 of 2
UNIT 1	(Circle Unit Performing Li	neup)	UNIT 2
	EXCEPTIONS		
DEVICE NUMBER	COMPONENT NOUN DESCRIPTION	RE	MARKS
Demonstration of the second se			
Personnel participating in device manipulation:	N 1		T '/' 1
	Name		Initials
-			
-			
- Device lineup completed by: _			
	Operator	Date	Time
Lineup 1 Reviewed:	Unit Supervisor		Date

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	Plant Cooldown							
Lineup 1 RV to PZR Equalizing Line Lineup			Page 2 of 2					

DEVICE NUMBER	COMPONENT NOUN DESCRIPTION	LOCATION	POSITION REQUIRED	ALIGNED BY	VERIFIED BY	NEW TAG NEEDED
1(2)-RC-0507	RX VESSEL HEAD VENTING MANIFOLD VENT VALVE	RCB On RV Head	CLOSED			
1(2)-RC-0509	RX VESSEL VENTING MANIFOLD DRAIN	RCB On RV Head	** OPEN/ CLOSED			
1(2)-RC-0504	RV HEAD/PZR EQUALIZING LINE DRAIN VLV	RCB 73' Outside On SG 1A(2A) N Shield Wall	CLOSED			
1(2)-RC-0132	RX VESSEL HEAD ATMOSPHERIC VENT VALVE	RCB On RV Head	** OPEN/ CLOSED			
1(2)-RC-0508	RX VESSEL HEAD VENTING MANIFOLD PZR EQUAL LINE ISOL	RCB On RV head	OPEN			
1(2)-RC-0501	RV HEAD/PZR EQUALIZING LINE ISOL VLV	RCB 73' Outside On SG 1A(2A) N Shield Wall	OPEN			
1(2)-RC-0163	PZR SPRAY LINE VENT VALVE	RCB Top of PZR	OPEN			
1(2)-RC-0103	PZR SPRAY LINE VENT VALVE	RCB Top of PZR	OPEN			
1(2)-RC-0506	RX VESSEL HEAD VENTING MANIFOLD PI-3636 ISOL	RCB On Head	* OPEN/ CLOSED			
1(2)-RC-0070	RX VESSEL HEAD VENT ISOL	RCB On RV Head inside shield wall	LOCKED OPEN			

* IF PI-3636 is installed, THEN OPEN "1(2)-RC-0506 RX VESSEL HEAD VENTING MANIFOLD PI-3636 ISOL".

** OPEN IF RCS LEVEL below 47 ft. 4 in.

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	Plant Co	ooldown		
Form 1	CVCS Line Bor	ation Tracking Fo	orm	Page 1 of 2
UNIT 1	(Circle Unit I	Performing Form)	
	NOTE	2		
	ponent and line NOT to be borated is Manager prior permission. (Forr		greater requires	Unit
associated With the 1	nes in Steps 2.0 and 3.0 of this for I component and piping. RCS borated to greater than or equ blowing lines ensuring the concen	al to 2875 ppm, C	CVCS system d	esign will
entering the RCS piping. (Reference 2.112) CVCS sections that do NOT Mixing volume				
	quire prior boration 201, CCP 1A(2A) recirc to VCT	Volume Control	Tank	
· · /	202, CCP 1B(2B) recirc to VCT 119, Aux Spray line	Volume Control Pressurizer	Tank	
	CVCS charging discharge	Charging pump including Regen		
1(2)-CV-0206, 0	CCP 1B(2B) discharge bypass	Charging pump including Regen		
	onent will NOT be borated, <u>THEN</u> s required, <u>AND</u> N/A components red.			dum NOT
.0 ENSURE	the following components borated	to 2800 ppm or g	reater as follow	U/s:
boi	SURE CCP 1A(2A) borated with rated water from VCT through CC 0POP02-CV-0004.	e	· · · · ·	
	SURE CCP 1B(2B) borated with rated water from VCT through CC			

1

- per 0POP02-CV-0004.
 ENSURE 1(2)-CV-MOV-0003 and associated lines to RCS borated with 300
- ENSURE 1(2)-CV-MOV-0006 and associated lines to RCS borated with 300 gallons of 2800 ppm or greater borated water per Step 7.12 of this procedure.

gallons of 2800 ppm or greater borated water per Step 7.12 of this procedure.

	0POP03-ZG-0007	Rev. 71	Page 216 of 216
	Plant Cooldown		
Form 1	CVCS Line Boration Trackin	ng Form	Page 2 of 2

- 3.0 RECORD Time/Date when each component and associated lines were determined to be borated.
 - CCP 1A(2A) and associated lines have been inservice with 600 gallons of 2800 ppm or greater borated water flushed.
 - CCP 1B(2B) and associated lines have been inservice with 600 gallons of 2800 ppm or greater borated water flushed.
 - 1(2)-CV-MOV-0003 and associated lines to RCS have been inservice with 300 gallons of 2800 ppm or greater borated water flushed through it. ____/____
 - 1(2)-CV-MOV-0006 and associated lines to RCS have been inservice with 300 gallons of 2800 ppm or greater borated water flushed through it. ____/____

RS1

	- NICOCONI	CALCULATION COVER SHEET		·····		
	ENERCON CALCULATION COVER SHEET REV.		1	· · · · · · · · · · · · · · · · · · ·		
		PAGI		PAGE NO.	1 of 49	
Title:	Radiological Release TI Action Levels	hresholds for Emergency	Client: Sou	uth Texas Proj	ect	
	Action Levels		Project: S'	FPNOC013	<u></u>	
Item		Cover Sheet Item	18	<u></u>	Yes	No
1	Does this calculation contain the assumptions)	any open assumptions that require	confirmation? (If Y	ES, Identify		~
2	Does this calculation serve as Calculation.) Design Verific	s an "Alternate Calculation"? (If YE ed Calculation No.	CS, Identify the desig	gn verified		~
3	Does this calculation Superse Calculation.) Superseded C	ede an existing Calculation? (If YES Calculation No.	S, identify the supers	seded		~
		PEDE runs.				
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Revisio	n Impact on Results; Value	es calculated in Attachment 1 de	creased and have	become the lim	ting values.	
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Excellence—Every project. Every day.			,	PAGE NO.	2 of 49	
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Item		CHECKLIST ITEMS		Yes	No	N/A
1		e design inputs correctly selected, referenced (la the design basis and incorporated in the calculation		\checkmark		
2	Assumptions – Were the justified and/or verified,	e assumptions reasonable and adequately descril and documented?	bed,	1		
3	Quality Assurance – W assigned to the calculation	ere the appropriate QA classification and requir	ements	~		
4		egulatory Requirements – Were the applicabl requirements, including issue and addenda, pro rements satisfied?		✓		
5	Construction and Oper operating experience bee	ating Experience – Have applicable construction considered?	on and			~
6	Interfaces – Have the de interactions with other ca	esign interface requirements been satisfied, inclu alculations?	uding	✓		
7	Methods – Was the calc –-satisfy-the calculation of	ulation methodology appropriate and properly a jective?	pplied to			
8	Design Outputs – Was the conclusion of the calculation clearly stated, did it correspond directly with the objectives and are the results reasonable compared to the inputs?		~			
- 9	Radiation Exposure – I to the public and plant p	Has the calculation properly considered radiation ersonnel?	n exposure	_ √		
10		Are the acceptance criteria incorporated in the c cation that the design requirements have been ned?	alculation	~		
11	Computer Software - 1 requirements of CSP 3.0	is a computer program or software used, and if s 2 met?	o, are the	1		~
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ENERCON Excellence-Every project. Every day	DESIGN VERIFICATION PLAN AND SUMMARY SHEET	REV. 1
		PAGE NO. 4 of 49
Calculation Design Verification Pla	n;	••••••••••••••••••••••••••••••••••••••
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Radiological Release Thresholds for Emergency Action Levels

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1.0 OBJECTIVE/SCOPE

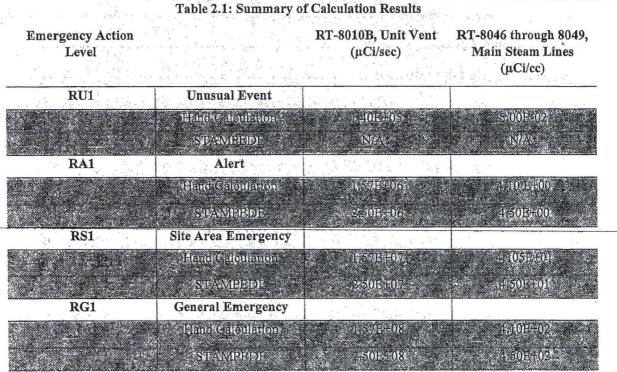
The purpose of this calculation is to determine the Emergency Action Level (EAL) threshold values of a radiological release from the Unit Vent or Main Steam Lines for an Unusual Event, Alert, Site Area Emergency, or General Emergency. The calculated threshold values are to be included in the STP EAL Technical Basis document, which implements the new NEI 99-01, Revision 6, Emergency Action Level Scheme and will be submitted to the NRC for approval. Upon NRC approval, the values will be used in 0ERP01-ZV-IN01, Revision 10, Emergency Classification.

Both a hand calculation and the South Texas Assessment Model Projecting Emergency Dose Evaluation (STAMPEDE) software program were used to generate the results. The hand calculation is included as Attachment 1.

Revision 1 of this calculation incorporated decay for a release taking place one hour after reactor shutdown. This was done to create continuity between the two methodologies present.

2.0 SUMMARY OF RESULTS

The results of the calculations for the radiation monitors specified in the STP EAL Basis Document and are listed in Table 2.1, below.



*STAMPEDE was not used to determine the threshold for RU1. Reference 5.10 indicates that the ODCM methodology should be used to determine the threshold value.

This calculation will be used to establish the threshold values for abnormal radiation based emergencies in the STP EAL Technical Basis document.



Radiological Release Thresholds for Emergency Action Levels

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3.0 METHOD OF ANALYSIS

Previously, STAMPEDE was used to calculate the Emergency Action Level threshold values for effluent releases. A hand calculation will verify the STAMPEDE calculations. The hand calculation is described in Attachment 1 of this document STAMPEDE conforms to the requirements of STP Procedure 0PGP07-ZA-0014, Software Quality Assurance Program. STAMPEDE was run at STP on an STP computer and under the supervision of an ENERCON employee with access to the STP site as a critical worker.

4.0 INPUTS

- 4.1 Per NEI 99-01, Revision 6, Initiating condition AU1, EAL 1, the Notice of Unusual Event initiating condition is a release of gaseous or liquid radioactivity greater than two times the ODCM limit for sixty minutes or longer (Reference 5.10).
- 4.2 The ODCM offsite dose limit is exceeded if the Xe-133 release concentration exceeds 7.41E-04 μCi/cc (Reference 5.6).
- 4.3 The Unit Vent flow rate is 9.4E+07 cc/sec (Reference 5.1).
- 4.4 The main steam line pressure and PORV choke flow rate are 1285 psig and 1.05E+06 lbm/hr, respectively (Reference 5.2).
- 4.5 The specific volume of saturated steam at 1285 psig is 0.338 ft³/lbm (Reference 5.3).
- 4.6 The release concentration is varied to find the release concentration which correlates to each emergency action level. Emergency action levels are taken from NEI 99-01, Revision 6 (Reference 5.10) for initiating conditions AA1, AS1 and AG1. EAL 1 is the EAL of interest in each initiating condition. The doses at the Site Boundary that correlate to the threshold concentrations are listed in Table 4.1.

Table 4.1 EAL Offsite Dose Initiating Conditions

	Alert	Site Area	General
TEDIE V. T. S.	10 mirem	100 mrem	1000 mrem
Thyroid CDE	50 mrem	500 mrem	5000 mrem

5.0 REFERENCES

- 5.1 Offsite Dose Calculation Manual, Revision 17, March 2011
- 5.2 Main Steam PORV Capacity Verification MC05591, Revision: 1
- 5.3 NIST Steam Tables, 2011
- 5.4 0ERP01-ZV-IN01, Emergency Classification Draft Revision 10
- 5.5 0ERP01-ZV-TP01, Offsite Dose Calculations, Revision 21
- 5.6 STP Calculation NC-9012, CRMS Rad Monitor Setpoints, Revision 7
- 5.7 STP Calculation NC-9011, Revision 2
- 5.8 STAMPEDE Computer Program, Revision 7.0.3.3
- 5.9 STAMPEDE User's Manual
- 5.10 NEI 99-01, Revision 6, Development of Emergency Action Levels for Non-Passive Reactors
- 5.11 0PGP07-ZA-0014 Quality Assurance Program
- 5.12 ITWMS Call Number 1000010987 Design Document, Revision 0



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6.0 ASSUMPTIONS

6.1 Unit Vent Noble Gas Monitor

To be consistent with the ODCM methodology, the unit vent release is assumed to be entirely Xe-133. The unit vent noble gas monitor is calibrated to Xe-133 (Reference 5.1) therefore; the monitor reading accurately reflects the Xe-133 release magnitude.

To be consistent with ODCM methodology, the main steam line release is assumed to be entirely Xe-133. The noble gas monitor is calibrated to Xe-133 (Reference 5.6).

6.2 Release Duration

Per Reference 5.10, Sections IC AA1, AS1, and AG1 developer notes, the release should be assumed to last one hour.

6.3 Release following Reactor Shutdown

The release initiates one hour after reactor shutdown. While a release initiating at reactor shutdown is likely, significant decay of short lived nuclides occurs during the migration time. A release at reactor shutdown would have a significantly higher activity at the monitor location than at the reception site. It is important for the threshold to not be calculated at shutdown as this ------would create a very high threshold which would not be appropriate for releases which occur shortly after shutdown. One hour after reactor shutdown is sufficient time to decay short lived nuclides and create a conservative threshold.

6.4 Source Term

Per Reference 5.1, any unit vent release with increased RCS activity and no core melt should be calculated using the gap inventory. Therefore, the gap inventory is used for all unit vent releases.

Per Reference 5.1, for a main steam line release following a steam generator tube rupture it is appropriate to use an inventory of noble gases plus 0.2% iodine. A steam generator tube rupture is the only scenario which would create significant offsite doses through a main steam line release.

6.5 Default STAMPEDE Input Values

Reference 5.10 developer notes for initiating conditions AA1, AS1 and AG1 suggest using the ODCM or the site's emergency dose assessment methodology. STAMPEDE is used for emergency dose assessment. Per Reference 5.1, when actual meteorology is not available, the default STAMPEDE values should be used. Had the ODCM methodology been used, the 500 hour peak χ/Q value would be used which is less conservative than the χ/Q value produced by STAMPEDE using default meteorological conditions. Therefore, the use of STAMPEDE default values provides a more conservative estimate than that of the alternative method outlined in Reference 5.10.

6.6 Average Effluent Concentration (χ/Q)

The same χ/Q is used for the unit vent and main steam line release. Reference 5.1 applies the same unit vent χ/Q to Units 1 and 2 which would also be applicable to the main steam line. All releases are considered to be ground level releases.



7.0 STAMPEDE CALCULATIONS

- 7.1 Unusual Event RU1
 - 7.1.1 Unit Vent Monitor

AU1 recommends declaring an unusual event due to a release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer (Reference 5.10).

STP sets the ODCM limit at 7.41E-04 μ Ci/cc (Reference 5.6, pg. 16). Two times the limit would be 1.48E-03 μ Ci/cc. The threshold is listed in μ Ci/sec so that variations in flow rate do not change the threshold. The normal flow rate from the unit vent is 9.4E+07 cc/sec (Reference 5.1).

$$Concentration\left(\frac{\mu Ci}{cc}\right) * Flow Rate\left(\frac{cc}{sec}\right) = Release Rate\left(\frac{\mu Ci}{sec}\right)$$
$$(1.48E - 03)\left(\frac{\mu Ci}{cc}\right) * (9.4E + 07)\left(\frac{cc}{sec}\right) = 1.4E + 05\left(\frac{\mu Ci}{sec}\right)$$
$$Equation 7.1.1.1$$

7.1.2 Main Steam Line Monitor

The ODCM does not calculate a release corresponding to allowable limits for the main steam line monitors. Since the unit vent release calculated in the ODCM was assumed to be primarily Xe-133, the assumption is made in the ODCM that other noble gases and iodine may be ignored in the calculation. This assumption is equally justifiable for the main steam line and the same limiting release will be used.

The magnitude of the release calculated for the unit vent Unusual Event applies to the main steam lines as well. The main steam line PORV's will create a dose exceeding two times the ODCM limit by releasing 1.4E+05 μ Ci/sec of activity which is equivalent to the release from the unit vent.

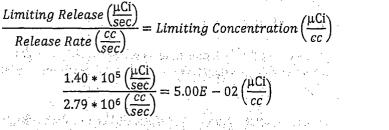
The steam lines hold saturated steam at 1285 psig, per Reference 5.2, which has a specific volume of 0.338 ft³/lbm (Reference 5.3). The PORVs will release the steam at 1.05E+06 lbm/hr per Reference 5.2. This creates a set flow rate of steam from the main steam lines of 2.79E+06 cc/sec as shown below.

$$F\left(\frac{lbm}{hr}\right) * Density\left(\frac{ft^3}{lbm}\right) * 28316.846\left(\frac{cc}{ft^3}\right) \div 3600\left(\frac{sec}{hr}\right) = \frac{cc}{sec}$$

$$1.05E + 06\left(\frac{lbm}{hr}\right) * 0.338\left(\frac{ft^3}{lbm}\right) * 28316.846\left(\frac{cc}{ft^3}\right) \div 3600\left(\frac{sec}{hr}\right) = 2.79E + 06\frac{cc}{sec}$$
Equation 7.1.2.1

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Equation 7.1.2.2

7.2 Alert, Site Area and General Emergencies - RA1, RS1, RG1

7.2.1 Unit Vent Monitor

Input

The Alert EAL is set to 10 mrem TEDE and 50 mrem Thyroid CDE per Reference 5.10. The emergency offsite dose calculation software STAMPEDE was used to calculate the release which corresponds to this dose. A release concentration correlating to the EAL threshold value was calculated by varying the input. The following assumptions and inputs were used for the calculation as described in Sections 4.0 and 6.0.

• Release begins at reactor trip

• Release lasts for one hour

• Gap inventory source term

Default STAMPEDE input values

- \circ Windspeed = 13.2 mph
- o Stability class D

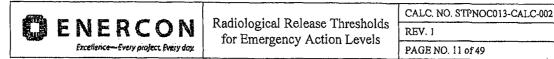
Results

Given a monitored unit vent release of $2.50E+06 \ \mu$ Ci/sec, the Thyroid CDE is 51 mrem/hr at the closest portion of the site boundary and the EAL Initiating Condition is exceeded.

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Threshold values for the Site Area Emergency and General Emergencies are multiples of 10 and 100 of the Alert. Since the correlation between release concentration and dose is linear, threshold values for the steam line monitors are 2.50E+07 and $2.50E+08 \mu Ci/sec$ for the SAE and GE respectively. Both are also limited by Thyroid CDE. Additional STAMPEDE iterations were performed to confirm this and are attached.

The input and output files can be found at the end of this document in Attachment 3.



7.2.2 Main Steam Line Monitor

Input

A release concentration correlating to the EAL threshold value was calculated by varying the input. The following assumptions and inputs were used for this calculation as described in Sections 4.0 and 6.0.

- Release begins at reactor trip
- Release lasts for one hour
- Noble gas + iodine with 0.2% iodine source term
- Default STAMPEDE input values
 - \circ Windspeed = 13.2 mph
 - o Stability class D

Results

Given a monitored main steam line release of 4.5 μ Ci/cc, the Thyroid CDE is 50 mrem/hr and the BAL Initiating Condition is exceeded.

The input and output-files can be found at the end of this document in Attachment 3. -

7.3 Threshold values for the Site Area Emergency and General Emergencies are multiples of 10 and 100 of the Alert. Since the correlation between release concentration and dose is linear, threshold values for the steam line monitors are 45 and 450 μ Ci/cc for the SAE and GE respectively. Both are also limited by Thyroid CDE. Additional STAMPEDE iterations were performed to confirm this and are attached.



Radiological Release Thresholds for Emergency Action Levels Attachment 1 CALC. NO. STPNOC013-CALC-002 REV. 1 PAGE NO. 12 of 49

Attachment 1 – Hand Calculations

1.0 OBJECTIVE/SCOPE

Each release calculated using STAMPEDE in the main document is calculated by hand in this attachment and the results compared to STAMPEDE.

2.0 SUMMARY OF RESULTS

Table 2.1 is displayed again below showing the results from all the calculations. The minor difference is due to STAMPEDE using decay factors over a one hour period after shutdown. This also accounts for the change in the limiting dose being TEDE in the hand calculations and Thyroid CDE in the STAMPEDE calculations. The accuracy of the hand calculation is considered sufficient and recommended for use in Emergency Action Levels.

Table 2.1 Results

Emergency Action	RT-8010b, Unit Vent	RT-8046 through 8049,
Level	(μCi/sec)	— Main Steam-Line-—
	a share in the second of	(µCi/cc)
BUI Unusual Event	an a	To the second

RUI	Unusual Event		
	Elenteke ellentletiteta	J1:4(0)E3∺0(6)	5 (0)015;(0)2;
	STAMPEDE	N/A	
RA1	Alert		
	Hanis Callentairen	11.571日中06	S 90E + 00)
	SANANAIDIRIDIR	2.5(0)25-(0)6	4. ×019400
RS1	Site Area Emergency		
	Hends Catentation	1. 57TE ±07/	3.9012+011
The state of the second s	SHEAWIPEIDIE	2., 1015 + 0/7	4.501.01
RG1	General Emergency		
	Banoneallantaison	21,57E #08	3.90E±027
	ST AMPEIDE	2,501:108	4/50E402
THE REAL PROPERTY OF THE PROPERTY OF THE REAL PROPE		COSTECENT ACCOUNT OVERTINGCO & CONSCIONED MEDICALIONAL	CARAMETERSKETERSCHWARTE (SELECTERSKETERSKETERSKETERSKETERSKETERSKETERSKETERSKETERSKETERSKETERSKETERSKETERSKETER

3.0 METHOD OF ANALYSIS

Using the limiting dose at the site boundary, the release is back calculated using atmospheric dispersion models. The X/Q value used is calculated from Regulatory Guide 1.145, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants. Rather than using the most conservative meteorology, average meteorological conditions are used as inputs



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to most closely agree with STP emergency dose assessment methodology per the ODCM and STAMPEDE. Assumed nuclide inventories are taken from Reference 5.4. The dose conversion factors are taken from Reference 5.2. A release concentration is used to find an initial projected dose at the Site Boundary. Using the projected dose at the Site Boundary, the release concentration is scaled to find the limiting dose for each EAL.

4.0 INPUTS

- The Unit Vent flow rate is taken from the Offsite Dose Calculation Manual; Revision 17, March 2011 and is 9.44E+07 cc/sec.
- The main steam line pressure and PORV choke flow rate were taken from Reference 5.5 and are 1285 psig and 1.05E+06 lbm/hr respectively.
- The specific volume of saturated steam at this pressure is taken from the NIST steam tables and is $0.338 \text{ ft}^3/\text{lbm}$.
- The release concentration is varied to find the release concentration which correlates to each emergency action level dose. Emergency action level doses are taken from NEI 99-01 Revision 6 for initiating conditions AA1, AS1 and AG1. EAL 1 is the EAL of interest in each initiating condition. The limiting doses are listed in Table 4.1. NEI 990-01 Revision 6 states that these
 values are based on fractions of the Environmental Protection Agencies Protective Action
 Guidelines (EPA PAGs) and the General Emergency represents the protective action values recommended by the EPA.

18	DIC 4.1 LAL I	nresnotas	
	Alert	Site Area	General
TEDE.	u0).mem	1001micmi	10009 mrem
Thyroid CDE	50 mrem	500 mrem	5000 mrem

- A release lasting one hour is selected per NEI 99-01 Revision 6 developer notes.
- Atmospheric dispersion factors are calculated per Regulatory Guide 1.145 (Reference 5.1). The reactor building dimensions used as inputs for this calculation are taken from Reference 5.13.
- Nuclide inventories are taken from TGX/THX 3-1, (Reference 5.4) which is the source document for the nuclide inventories used in STAMPEDE. The release inventories are a gap release and noble gases plus 0.2% iodine which are listed below. Each nuclide inventory was normalized to one so it could be scaled to various release activities.

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104007		·	1	1.1.4	Tes West	
		EXCEU	ence-Ever	cy prop	sci zver	y aay.

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, + + <i>ia</i>	Nuclide	Activity (µCi/cc)	Normalized	Nuclide	Activity (µCi/cc)	Normalized
	11.1134	1,1,012055	1, 12)5, (08),	Xie-135	5 5015,006	5.62E.02
	1-132	1.50E+05	1.53E-03	Xe-137	1.90E+07	1.94E-01
	1-133	2.2015-0.5	2:25E+03	Xe-138	1(18(0)至于0)71。	JI 8447 (04)
	I-134	2.40E+05	2.45E-03	Cs-134	3.70E+01	3.78E-07
6 . <u>8</u> 9	14135	2.0015+05	.s. 2.0592403	Cs=137	21910124#09[2, 0,74:
÷	Kr-83m	1.30E+06	1.33E-02	Te132	4.80E+00	4.91E-08
و بالا الأثنية الان	Kr-85m	2*9(0)84=406	2,971,025	Mio99	1.2215401	1,2512-07
	Kr-85	3.70E+05	3.78E-03	Ru103	8.80E-03	9.00E-11
المحرجي والم	Kre87	(6)(+; <u>-1</u> (9)(-;+/0)(6)	15 (6 2 1 5 (0) <u>2</u>)	Runnie	<u>?-\\$(0)[``-(0)\$</u>	2.9715.10
	Kr-88	7.80E+06	7.98E-02	Zr95	1.10E-02	1.12E-10
1.12	1546-189	(9),510)E+H(0)6+	9) 7 <u>67</u>][s (9 <u>6</u>],	ll, ale ()	16 9 (015 - 10 /2)	1.9415.40
	Xe-131m	1.10E+05	1.12E-03	Ce144	7.40E-03	7.57E-11
	Xe-133m	(678(0)F-1-(0)	(51-94515;=(0)S)	Ce 441	it (010)5, (0,2°,	, 11.(0,24E) 41(0)
	Xe-133	2.20E+07	2.25E-01	Sr89	6.40E-02	6.55E-10
	Xe-135m	4:2(9):-1:(0)6	4 31015 (0)2°	Si(90)	3/_2(9)=_(0)\$	3,2745-14

Table 4.2 Gap Inventory

Table 4.3 Noble Gases+0.2% Iodine Inventory

Nuclide	Inventory	Normalized
164835	61 1801E -012	252[6][(0]4]
I-132	8.61E-02	3.19E-04
41-1381] [*] (0)0)÷, (0)0	31 7/3JE 04
I-134	1.86E-02	6.92E-05
JE135	2.73E_0.0	JF (0)88E - (8)85
Xe-131m	2.80E+00	1.04E-02
Ne-HSB	2. 4(0)를 13(0)2)	8091018-1048 -
Xe-133m	4.20E+00	1.56E-02
Xe-135	74(6' 015)#19(96	2.8215.02
Xe-135m	4.00E-01	1:48E-03
-Xe-137		5,93E=04
Xe-138	5.80E-01	2.15E-03
-Ky-85m	≦3,7401⊒,(031)⊂.	jit, 3771E) (0/37, 0) }}
Kr-85	7.60E+00	2.82E-02
-Kr-85m	11, 5101E +(010) .	5.50E-08
Kr-87	9.80E-01	3.63E-03
Ki-88	2.80日平00 1	1.04E-02
Kr-89	8.40E-02	3.12E-04

• The dose conversion factors taken from EPA 400R92001 (Reference 5.2) are listed in Tables 4.4 and 4.5 below.

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			Dose Conversion
	Dose Conversion		Factor
	Factor		(rem per
Nuclide	(rem per uCi*hr/cc)	Nuclide	uCi*hr/cc)
J-131	5.30E+04	Xe-135	1,40E+02
I-132	4.90E+03	Xe-137	1.10E+02
·I-133	1.50B+04	Xe-138	7.20E±02
I-134	3.10E+03	Cs-134	6.30E+04
1-135	8:10E+03	Cs-137-	4.10E#04
Kr-83m	alera da alera de la constante	Te132	1.20E+04
Kr-85m	9:30E+01	Mo99	5.20日十03
Kr-85	1.30E+00	Ru103	1.30E+04
Kr-87	5.10E+02	Ru106	5.70E+05
Kr-88	1.30E+03	Zr95	3.20E+04
Kr-89	4.20E+03	La140	L.10E+04
<u>Xe-131m</u>	4.9	_Ce144	4.50E+05
Xe-133m	1.70E+01	Ce-141	1.10E±04
Xe-133	2.00E+01	Sr89	5.00E+04
Xe-135m	2,50E+02	Sr90	1.60E±06

Table 4.4 TEDE Dose Conversion Factors

Table 4.5 Thyroid CDE Dose Conversion Factors

Nuclide	Thyroid CDE DCF (rem per uCi*hr/cc)
I-131	130E H06
I-132	7.70E+03
1-133	2,20E+05
I-134	1.30E+03
-1-135	3-801+04

• The unit vent noble gas monitor energy efficiency by nuclide is taken from Offsite Dose Calculation Manual (Reference 5.3). The values are relative to Xe-133 efficiency since the monitor is calibrated to Xe-133. Table 4.6 displays the energy efficiency by nuclide relative to Xe-133.



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Table 4.6 Energy Efficiency Relative to Xe-133

	Efficiency Relative
	to Xe-133
Nuclide	(uCi/cc) _{equivalent}
Kr-83m	
Kr-85m	1.9
Ku-85	2.4
Kr-87	2.8
Kat-88	2.3
Kr-89	2.8
Ac. 131m.	(0f:0al(5)
Xe-133m	0.14
X6-438	
Xe-135m	0.042
Xte-41315	25
Xe-137	2.8
- XIC 138	2.8

*There is no relative efficiency available for Kr-83m. Assumption 6.4 further justifies the omission.

and the second second	1 able 4.	Table 4. / Nuclide Hall Lives					
Nuclide	Half Life (hr)	Nuclide	Half Life (hr)				
્યાનકોઈ 👘	11.9837.002	Xe-135	2 9 08E # 90				
I-132	2.38E+00	Xe-137	6.38E-02				
ikasa	- 2:10 SRE 1110dC .	Xe-138	2.36E.0U				
I-134	8.77E-01	Cs-134	1.80E+04				
-i-135	67641244010	Co-1137	2,601: 105				
Kr-83m	1.83E+00	Te132	7.79E+01				
Kar-8.5m,	4,4815,400	M699	6-62 <u>E</u> +01				
Kr-85	9.40E+04	Ru103	9.44E+02				
<u>Ka</u> r-87	1 27BH00	Ru106	8.8415±0.94				
Kr-88	2.84E+00	Zr95	1.55E+03				
Kir-89	5.101.02	-La t 40	4\$(0±1 <u>5</u> ;+0Hk				
Xe-131m	2.83E+02	Ce144	6.82E+03				
Xe-183m	5.42E+01	Ce-141	7.77E+02				
Xe-133	1.27E+02	Sr89	1.21E+03				
Xe-135m	2.60E-01		2,50B±05P				

Table 4.7 Nuclide Half Lives

• The half-lives are taken from Reference 5.15 which lists the input data used by STAMPEDE.



5.0 REFERENCES

- 5.1 Regulatory Guide 1.145, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants, Revision 1, November 1982.
- 5.2 EPA 400R92001, Manual of Protective Action Guides and Protective actions for Nuclear Incidents, Revision 1, May 1992.
- 5.3 Offsite Dose Calculation Manual, Revision 17, March 2011.
- 5.4 TGX/THX 3-1, Revision 5, Westinghouse Radiation Analysis Manual.
- 5.5 MC05591, Main Steam PORV Capacity Verification, Revision 1.
- 5.6 NIST Steam Tables, 2011.
- 5.7 0ERP01-ZV-IN01, Emergency Classification, Revision 10.
- 5.8 0ERP01-ZV-TP01, Offsite Dose Calculations, Revision 21.
- 5.9 STP Calculation NC-9012, Process and Effluent Radiation Monitor Set Points, Revision 7
- 5.10 STP Calculation NC-9011, CRMS Rad Monitor Setpoints, Revision 2.
- 5.11 STAMPEDE Computer Program, Revision 7.0.3.3.
- 5.12 STAMPEDE User's Manual
- 5.13 STP Drawing 6C189N5007, General Arrangement Reactor Containment Building, Revision 6
- 5.14 NEI 99-01, Revision 6, Development of Emergency Action Levels for Non-Passive Reactors
- 5.15 ITWMS Call Number 1000010987 Design Document, Revision 0.

6.0 ASSUMPTIONS

6.1 Release lasts for one hour

Per NEI 99-01 (Reference 5.14), IC AA1, AS1, AG1 developer notes, the release should be assumed to last one hour.

For this to be true for the main steam line, it is assumed that the PORV is open for one hour. To calculate the most limiting case, it is assumed that the maximum flow possible is being released from the PORV.

6.2 Nuclide mix

Per 0ERP01-ZV-TP01, Offsite Dose Calculations (Reference 5.8) any unit vent release with increased RCS activity and no core melt should be calculated using a gap inventory. It is conservative to assume an increased RCS activity and not within the intended scope of the relevant initiating conditions to assume core melt. Therefore, a gap inventory is used for all unit vent releases.

Per 0ERP01-ZV-TP01, Offsite Dose Calculations (Reference 5.8) for a main steam line release following a steam generator tube rupture it is appropriate to use an inventory of 100 percent noble gases plus 0.2 percent iodine. Since a steam generator tube rupture releasing through the PORVs is the only steam generator tube rupture scenario which would create offsite doses large enough to meet or exceed the EALs, this assumption is made.



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6.3 Atmospheric Dispersion

NEI 99-01 (Reference 5.14) developer notes for initiating conditions AA1, AS1 and AG1 suggest using the ODCM or the site's emergency dose assessment methodology. Per 0ERP01-ZV-TP01, Offsite Dose Calculations (Reference 5.8), when actual meteorology is not available, the default STAMPEDE values should be used. The default STAMPEDE values assume a stability class D for atmospheric dispersion and a windspeed of 13.2 mph. These values were used as inputs for the atmospheric dispersion calculation.

It is clear that STAMPEDE uses the same method for calculating atmospheric dispersion factor (X/Q) outlined in section 7.1.1 of this Attachment. However, STAMPEDE does not follow the same logic in selecting the appropriate result from the three calculations. The STAMPEDE value printed in the results found in attachment 3 is consistent with the largest of the three hand calculated X/Q values. This suggests that STAMPEDE simply selects the largest of the three X/Q values resulting in a much more conservative estimate. This calculation will deviate from the recommendations of Regulatory Guide 1.145 and conform to the methodology STAMPEDE uses.

The close proximity of all release points allows for a single atmospheric dispersion coefficient to be used. This assumption is also made by STAMPEDE.

6.4 Exposure Pathways

The dose conversion factors used in table 4.4 and 4.5 represent a summation of dose conversion factors for external plume exposure, inhalation from the plume, and external exposure from deposition. Because the dose estimations are used for implementing early phase protective actions, conversion factors using limited pathways are appropriate.

The EPA does not provide a dose conversion factor for Kr-83m. Because the PAGs are based on EPA dose calculations, it is appropriate to only use the nuclides for which dose conversion factors are provided. Additionally, Kr-83m represents only 1.33% of the nuclide inventory activity and its exclusion would not significantly affect the final dose.

6.5

The release initiates one hour after reactor shutdown. While a release initiating at reactor shutdown is likely, significant decay of short lived nuclides occurs during the migration time. A release at reactor shutdown would have a significantly-higher activity at the monitor location than at the reception site. It is important for the threshold to not be calculated at shutdown as this would create a very high threshold which would not be appropriate for releases which occur shortly after shutdown. One hour after reactor shutdown is sufficient time to decay short lived nuclides and create a conservative threshold.

Decay is incorporated for one hour from reactor shutdown as well as migration time. Half-lives are taken from Reference 5.15. Migration time is assumed to be the reciprocal of the wind speed.



7.0 HAND CALCULATIONS

7.1 Unit Vent Monitor

7.1.1 X/Q

ł

The atmospheric dispersion factor, X/Q, determines the change in concentration between the unit vent discharge and the dose reception site. This value is based on meteorological conditions and will vary with wind speed and stability class. The ODCM uses the highest annual average X/Q value at the site boundary which is $5.3E-06 \text{ sec/m}^3$. However, for an accident related release STAMPEDE is used rather than the ODCM. STAMPEDE uses real time, user entered, or default meteorological conditions to calculate the X/Q for a specific accident. Default values will be used as inputs into the Regulatory Guide 1.145 method for calculating X/Q as described below. Default values are identified in section 6.0, Atmospheric Dispersion.

For a neutral atmospheric stability class, which is the default in STAMPEDE, X/Q values can be determined through the following set of equations.

 $\frac{X}{Q} = \frac{1}{\overline{U}_{10} \left(\pi \sigma_y \sigma_z + \frac{A}{2} \right)}$

$\frac{X}{Q} = \frac{1}{\overline{U}_{10}(3\pi\sigma_y\sigma_z)}$	

Equation 7.1.1.1

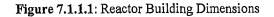
Equation 7.1.1.2

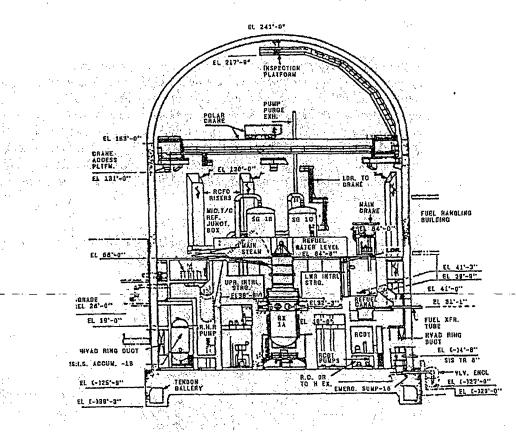
Equation 7.1.1.3

	Equation 7.1.1.5
Where	
X/Q	= relative concentration (sec/m^3)
π	= 3.14159
\overline{U}_{10}	= windspeed at 10 meters above plant grade (m/s)
 σ_y	= lateral plume spread (m), a function of atmospheric stability and distance, determined from Regulatory Guide 1.145 Figure 1
σ_z	= vertical plume spread (m), a function of atmospheric stability and distance, determined from Regulatory Guide 1.145 Figure 2
Σ_y	= $(M-1)\sigma_{y800m} + \sigma_y$ = lateral plume spread with meander and building wake effects (m), a function of atmospheric stability, windspeed \overline{U}_{10} , and distance; M is determined from Regulatory Guide 1.145 Figure 3
Á	= the smallest vertical-plane cross-sectional area of the reactor building (m^2) , taken from Reference 5.13 and shown below

 $\frac{X}{Q} = \frac{1}{\overline{U}_{10}\pi\Sigma_y\sigma_z}$

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Assuming the reactor building cross section to be a perfect rectangle and half sphere, the variables are defined as follows;

$$\overline{U}_{10} = 13.2 \text{ mph} = 5.9 \text{ m/s}$$

 $\sigma_{y} = 1200 \text{ m}$ $\sigma_{z} = 4.2 \text{ m}$ $\Sigma_{y} = (M - 1)\sigma_{y800m} + \sigma_{y} \text{ ; } M=1 \rightarrow \sigma_{y} = 1200 \text{ m}$ $A = (135' * 158') + (\frac{\pi * 79^{2}}{2}) = 31128.37$

The three equations become;

$$\frac{X}{Q} = \frac{1}{5.9\left(\pi 1200 * 4.2 * \frac{31128.37}{2}\right)} = 5.398 * 10^{-6}$$

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	Excel	lence-	-Ever	y atoh	ect Evei	ry day	Attac

$$\frac{X}{Q} = \frac{1}{5.9(3\pi * 1200 * 4.2)} = 3.568 * 10^{-6}$$
$$\frac{X}{Q} = \frac{1}{5.9 * \pi * [(1-1)\sigma_{y800m} + 1200] * 4.2} = 1.07 * 10^{-5}$$

To select the appropriate X/Q value, the first two X/Q values should be compared and the higher value selected. This value is then compared with the third X/Q value and the lower of those two is the appropriate X/Q value. The appropriate X/Q is 5.39E-06 sec/m³ for default meteorological conditions by the methodology recommended in Regulatory Guide 1.145.

This calculated value is very similar to the ODCM highest average value of 5.3E-06 sec/m³ which was not selected for use. Additionally, the value shown in the STAMPEDE output file at one mile is 1.032E-05 sec/m³. This suggests that STAMPEDE uses the same methodology and simply selects the largest atmospheric dispersion value to remain conservative. This methodology will be replicated and 1.07E-05 will be used as the X/Q.

7.1.2 Nuclide Inventory

As previously stated, a gap inventory is appropriate for this problem. The gap inventory is taken from TGX/THX 3-1 (Reference 5.4) which is used as the source term for STAMPEDE inventories. The concentrations were then normalized so they could be scaled to the varying emergency classifications. The values for the normalized inventory can be found in Table 4.2.

7.1.3 Dose Conversion Factors

As stated in NEI99-01 (Reference 5.14) developer notes, the purpose of dose projections is to check if the Environmental Protection Agencies Protective Action Guidelines (EPA PAGs) have been exceeded. The dose conversion factors provided by the EPA in EPA 400R92001 are used. These dose conversion factors account for external plume exposure, inhalation from the plume, and external exposure from deposition and are listed Tables 4.4 and 4.5, and taken from tables 5-1, 5-2 in EPA 400R92001 (Reference 5.2).

The EPA does not provide a dose conversion factor for Kr-83m. This nuclide contributes 1.33% of the inventory activity. The lack of this nuclide's contribution to the final dose will not significantly affect the outcome.

7.1.4 Decay Time

One hour of decay is incorporated to the monitor response due to the release initiating one hour after reactor shutdown. Decay is also incorporated for the duration of the migration time. The total decay time is one hour plus the reciprocal of wind speed, or 1.07575 hours.



7.1.5 Dose Calculations

The dose rate at the site boundary is calculated using Equation 7.1.5.1.

$$\dot{D} = \frac{X}{Q}F\sum_{i}^{n}C_{i} * 0.5^{\frac{1.07575}{T_{1/2_{i}}}} * DCF_{i}$$

Equation 7.1.5.1

Where

 \dot{D} = dose rate per hour at the site boundary

= atmospheric dispersion coefficient as calculated in section 7.1.1

F = unit vent flow rate

 C_i = concentration of nuclide i at the time of shutdown

1.07575 = the total decay time of interest from section 7.1.4

 $T_{1/2i}$ = the half-life of nuclide i

 DCF_i = the dose conversion factor for nuclide i listed in tables 4.4 and 4.5

The total concentration of the nuclides is varied to find the dose rate of interest. Beginning with an arbitrary release concentration of 1 μ Ci/cc, the dose rate is calculated. Since the dose is linearly correlated to concentration, the release concentration may be scaled to find the dose rate of interest.

The Alert EAL is 10 mrem TEDE or 50 mrem Thyroid CDE. Using the above method to calculate TEDE with the appropriate conversion factors, a limiting release rate of $2.33E+06 \ \mu Ci/sec$ from the unit vent results in 5.7 mrem TEDE. Using the calculated release rate to find Thyroid CDE with the appropriate conversion factors, the same release results in a 50 mrem Thyroid CDE at the site boundary. Thus, $2.33E+06 \ \mu Ci/sec$ is the limiting release rate based on the 50 mrem Thyroid CDE EAL initiating condition.

The limiting release rate threshold values for the Site Area Emergency and General Emergencies are multiples of 10 and 100 of the Alert release rate threshold value.

These calculations can be found in Attachment 2.

7.1.6 Monitor Response

The unit vent noble gas monitor is calibrated to Xe-133. Monitor efficiencies relative to Xe-133 by nuclide are listed in ODCM Table B3-2. To find the monitor reading associated with each limiting release, the noble gas concentrations must be multiplied by the monitor response and summed. Table 4.6 shows the indicated response of the unit



vent noble gas monitor by nuclide and Equation 7.1.5.1 shows how the monitor response was calculated.

$$Monitor \ Response = \sum_{i}^{n} C_{i} * Re_{i}$$

Equation 7.1.5.1

Where

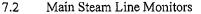
 C_i = concentration of nuclide i (µCi/cc)

 Re_i = monitor response to nuclide i (μ Ci/cc)_{Xe-133} equivalent</sub>

In the case of an Alert, the $2.33E+06 \ \mu$ Ci/sec release rate will read as $1.57E+06 \ \mu$ Ci/sec on the monitor. Kr-83m does not have an indicated monitor response coefficient. Because Kr-83m is only 1.34% of the noble gases and does not contribute to the dose calculation, its exclusion is acceptable.

This again is a linear correlation and the SAE and GE scale by factors of 10 and 100 respectively.

These calculations can be found in Attachment 2.



7.2.1 X/Q

Since the atmospheric dispersion is independent of nuclide inventory or release rate and the close proximity of the releases, the X/Q value will be the same for a main steam line release as it is for a unit vent release. This assumption is also taken by STAMPEDE and outline in Assumption 6.3.

7.2.2 Nuclide Inventory

Per 0ERP01-ZV-TP01, if the release path is the main steam line with a steam generator tube rupture, the nuclide inventory should be 100% noble gas and 0.2% of the iodine from the reactor coolant.

The secondary steam concentration for noble gases and iodine after a steam generator tube rupture are taken from TGX/THX 3-1 (Reference 5.4). Values for the reactor coolant inventory are listed in table 4.3. All of the noble gases are used and the iodine concentration from the coolant inventory is scaled to total 0.2% of iodine in the total coolant inventory. These inventories are then normalized to one. These values are listed in Table 4.3.

7.2.3 Dose Conversion Factors

The dose conversion factors used are found in Tables 4.4 and 4.5, taken from tables 5-1, 5-2 in EPA 400R92001.



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7.2.4 Decay Time

One hour of decay is incorporated to the monitor response due to the release initiating one hour after reactor shutdown. Decay is also incorporated for the duration of the migration time. The total decay time is one hour plus the reciprocal of wind speed, or 1.07575 hours.

7.2.5 Dose Calculations

Equation 7.1.5.1 applies to the release from the main steam lines. The main steam line flow rate is used instead of the unit vent flow rate for the value F. The main steam line flow rate was calculated in Equation 7.1.2.2 of the STAMPEDE CALCULATIONS section of this document as 2.79E+06 cc/sec.

The Alert EAL threshold is 10 mrem TEDE or 50 mrem Thyroid CDE at the site boundary (Table 4.2). Using the method in Equation 7.1.5.1 to calculate TEDE with the appropriate conversion factors, a concentration at time of shutdown of 4.10 μ Ci/cc would result in 6.89 mrem TEDE at the site boundary if the steam line PORV was open for an hour. Using the same steam line concentration to calculate Thyroid CDE results in 50 mrem Thyroid CDE at the site boundary.

The steam line concentrations at the time of shutdown for the Site Area Emergency and General Emergencies are multiples of 10 and 100 of the Alert. Since the correlation between release concentration and dose is linear, values for the steam line concentration at time of shutdown are 41.0 and 410 μ Ci/cc for the SAE and GE respectively. Both are also limited by Thyroid CDE.

These calculations can be found in Attachment 2.

7.2.6 Monitor Response

Because the main steam line monitor is adjacent to the main steam line, significant shielding takes place between the source and monitor. STP calculation NC-9011 Revision 2 calculates a conversion factor for the main steam lines for a noble gas inventory which is incorporated into the monitor readout. No monitor response needs to be calculated.

The concentration of the main steam line one hour after shutdown given a concentration of 4.10 μ Ci/cc at the time of shutdown is 3.90 μ Ci/cc. This calculation is also found in Attachment 2. Additionally, the monitor feadings for the SAE and GE one hour after shutdown are 39.0 and 390 μ Ci/cc respectively. These values are the thresholds for the main steam line monitor.

CALC. NO. STPNOC013-CALC-002 Radiological Release Thresholds

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Attachment 3

REV.1

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STAMPEDE User Supplied Information Revision 7033 9/28/2011 User Name: Unit Vent Alert \mathcal{F} Date/Thme: 12/17/2013 15:24 ֈ. Converts

User Supplied Information

Nietem ological Data Iopais: Granndlezel aind velocity: Granndlezel aind velocity: Granndlezel aind franz: User-relacted Simbility Class 13.2 mi/hr 180 degrees "D - Neutral" Stability Class:

Munifored Unit Vent Release: Unit Vent Release Rate entered 2.50E+006 aCitat

Reatter Shufdown Date/Time: 12/17/2013 14:24 Referre Start Date/Time: 12/17/2013 15:24 Estimated Release Doration:

Nerlich Matere:

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DENERCON

Excellence-Every project Every day

Calculated NOBLE CAS release rate: 1.192-005 of Views

100 hours **G**epTreetery

NOHL	2 GA S	IODE	SUS	PARTICI		
vaciide	uCi/sec	Norit	nCi/sec	Nuclide	DC1/SEC	
G-8338	2536+004	L131:	1128+003	Co-134:		an an the second se
G-85.	1.05E+004	1-132:	3188+008	CE-137.		
G-8514	7068+004	I-133:	6.0584008	Ce/Pr-145:		
G-87:	9.036+004	1-114:	3.028+008	Ce-141:	2.848-004	
Kr-88:	1748+005	I-135 :	5.128+003	12-140:	5.31E-004	
Kr-89:	100-380 E			Ma-99:	3438-001	
Xo-331M	3.128+003			Ra/Rh-106:	8258-005	
Se-133.	6278+005			Rn-105:	2.225-004	
V-133M	191E+004			Sr/K-90:	9.10B-005	
Ke-125:	1458+005			Sr-89	1.872-003	
Xr-335M	8148+008	•		Te-132.	135E-001	
Xe-137.	9.5384600	• •	· ·	Zr-95:	3.138-004	
Xe-138:	2.602+004					

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12/17/2013 3:24:46 PM

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Table A2-1: Unusual Event Emergency Calculations

	(L)CAVCO)	LA MICO NEWYORK	Union/sec
	1.48E-03	9.44E+07	1.40E+05
WISL L	initing Release Rate	Flow Rate	Timeting Concentration
1	(ICTree)		Ref (iChoolean)
	1.40E+05	2.79E+06	5.00E-02

Table A2-2: Input Values for Calculations

	LUL AUGA	Rolense Rate	Release corsant -	Luid Conversion dor Reicess Constant	Total Release Variable	Decay Time-
	and services) and a	Colsector	(steelines) to the	hid marsten and	2. Micheol	A (ha)
5.40E-06	3600	9.44E+07	1.83E+06	5.10E-04	1.79E-02	1.07575

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Table A2-3: Calculations for Boundary Concentrations and TEDE dose due to Unit Vent Release

	an a	11.11 ×		a at da					
				Release	Concertration	Half Life	Deerved Concernsition	LOSS Conversion	Dose
. America	Inventory	Normalized	concentration	Consign	o-Berndary			Fachar	Contraction
							NICE IF COLUMN	ા હોવી છોડા	
				Contraction CC 4CC 15					1.4772.02
. I-131	1.10E+05	1.12E-03	2,76E-05	1.01E-03	2.79E-08	1.93E+02	2.78E-08	5.30E+04	1.47E-03
I-132	1.50E+05	1.53E-03	3.77E-05	1.01E-03	3.81E-08	2.38E+00	2.79E-08	4.90E+03	1.37E-04
I-132 I-133	2.20E+05	2,25E-03	5 55E-05	1.01E-03	5.61E-08	2.03E+01	5.40E-08	1.50E+04	8.11E-04
I-135 I-134	2.40E+05	2.45E-03	6.04E-05	1.01E-03	6.11Ė-08	8.77E-01	2.61E-08	3.10E+03	8.09E-05
I-134 I-135	2.00E+05	2.05E-03	5.06E-05	1.01E-03	5.11E-08	6.61E+00	4.56E-08	- 8.10E+03	3.70E-04
	1.30E+06	1.33E-02	3 28E-04	1.01E-03	3.31E-07	1.83E+00	2.21E-07	ж.,	0.00E+00
Kr-83m	2.90E+06	2.97E-02	733E-04	1.01E-03	7,40E-07	4.48E+00	6.27E-07	9.30E+01	5.83E-05
Kr-85m	2.90E+00 3.70E+05	3.78E-03	933E-05	1.01E-03	9.42E-08	9.40E+04	9.42E-08	1.30E+00	1.22E-07
Kr-85	5.50E+06	5.62E-02	1.39E-03	1.01E-03	1.40E-06	1.27E+00	7.79E-07	5.10E+02	3.97E-04
Kr-87		7.98E-02	1.97E-03	1.01E-03	1.99E-06	2.84E+00	1.53E-06	1.30E+03	1.99E-03
Kr-88	7.80E+06		2.40E-03	1.01E-03	2.42E-06	5.10E-02	1.08E-12	1.20E+03	1.30E-09
Kr-89	9.50E+06	9.72E-02	요즘 나는 것은 것을	1.01E-03	2.79E-08	2.83E+02	2.78E-08	2.50E+02	1.36E-07
Xe-131m	1.10E+05	1.12E-03	2,76E-05	1.01E-03	1.73E-07	5.42E+01	1.71E-07	1.40E+02	2.90E-06
Xe-133m	6.80E+05	6.95E-03	171E-04	法意识的 网络戴属科教教 医马根核	5,61E-06	1.27E+02	5.57E-06	1.10E+02	1.11E-04
Xe-133	2.20E+07	2.25E-01	5 55E-03	1.01E-03	e status de la contra de la contr	2.60E-01	6.09E-08	7.20E+02	1.52E-05
Xe-135m	4.20E+06	4.30E-02	1 06E-03	1.01E-03	1.07E-06	9.08E+00	1.29E-06	5.30E+04	1.81E-04
Xe-135	5.50E+06	5.62E-02	1 39E-03	1.01E-03	1.40E-06	Standard Selection March	나라고 제품이 귀엽했다. 이는 것은 것같	4.90E+03	4.47E-09
Xe-137	1.90E+07	1.94E-01	4 79E-03	1.01E-03	4.83E-06	6.38E-02	4.06E-11	4.90E+03	1.40E-04
Xe-138	1.80E+07	1.84E-01	4.54E-03	1.01E-03	4.59E-06	2.36E-01	1.95E-07	1.306+04	1.401-04

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		to the test to a state of a state of the state of the	•						
u dide	Taventony	Normalizeti	/apaod	Relieve	Concentration @ Boundary	Hall L		Dose Consursion Rector	
			enneentration	Constant		- totha	Concentration	Reine Octor Coche/co	Contrabatio
Cs-134	3.70E+01	3.78E-07	9.33E-09	1.01E-03	9.42E-12	1.80E+	PACENE CONTRACTOR SUCCESSION AND A CONTRACTOR	6.30E+04	5.93E-07
Cs-137	2.90E+01	2.97E-07	7.33E-09	1.01E-03	7.40E-12	2.60E+	05 7.40E-12	4.10E+04	3.03E-07
Te132	4.80E+00	4.91E-08	1.2 E-09	1.01E-03	1.22E-12	7.79E+	01 1.21E-12	1.20E+04	1.45E-08
Mo99	1.22E+01	1.25E-07	3.08E-09	1.01E-03	3.11E-12	6.62E+	01 3.08E-12	5.20E+03	1.60E-08
Ru103	8.80E-03	9.00E-11	2.22E-12	1.01E-03	2.24E-15	9.44E+	02 2.24E-15	1.30E+04	2.91E-11
Ru106	2.90E-03	2.97E-11	7.33E-13	1.01E-03	7.40E-16	8.84E+	03 7.40E-16	5.70E+05	4.22E-10
Zr95	1.10E-02	1.12E-10	2.76E-12	1.01E-03	2.79E-15	1.55E+	03 2.79E-15	3.20E+04	8.93E-11
La140	1.90E-02	1.94E-10	4.79E-12	1.01E-03	4.83E-15	4.03E+	01 4.75E-15	1.10E+04	5.22E-11
	7.40E-03	7.57E-11	1.87E-12	1.01E-03	1.89E-15	6.82E+	03 1.89E-15	4.50E+05	8.49E-10
Ce144		1.02E-10	2.52E-12	1.01E-03	2.54E-15	7.77E+	02 2.54E-15	1.10E+04	2.79E-11
	1.00E-02	1.021-10				* 1			
Ce144 Ce-141 Sr89	1.00E-02 6.40E-02	6.55E-10	1.62E-11	1.01E-03	1.63E-14	1.21E+	03 1.63E-14	5.00E+04	8.16E-10

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Table A2-4: Thyroid Dose Calculation for Unit Vent Release

Nucline	Doonyei Cancentra wit	Thyrofd DCF	Thyroid Dose
a series and	(puCt@therec.)	Frein arnare Shiree	
I-131	2.78E-08	1.30E+06	3.61E-02
I-132	2.79E-08	7.70E+03	2.15E-04
I-133	5.40E-08	2.20E+05	1.19E-02
I-134	2.61E-08	1.30E+03	3.39E-05
I-135	4.56E-08	3.80E+04	1.73E-03
		and our share of the	
1			5007-02

Table A2-5: Unit Vent Monitor Response to Nuclide Inventory

. Niteirde	Concentration	Halflife	Concentrations After Lar	Response	NESTIMUS!
	Le Micoteo	E o irs			
Kr-83m	\$.28E-04	1.83E+00	2.25E-04		0.00E+00
Kr-85m	7.33E-04	4.48E+00	6.28E-04	1.9	1.19E-03
Kr-85	9.33E-05	9.40E+04	9.33E-05	2.4	2.24E-04
Kr-87	1.39E-03	1.27E+00	8.03E-04	2.8	2.25E-03
Kr-88	1,97E-03	2.84E+00	1.54E-03	2.3	3.55E-03
Kr-89	2.40E-03	5.10E-02	3.00E-09	2.8	8.40E-09
Xe-131m	2.76E-05	2.83E+02	2.76E-05	0.015	4.13E-07
Xe-133m	1.71E-04	5.42E+01	1.69E-04	0.14	2.37E-05
Xe-133	\$.55E-03	1.27E+02	5.52E-03	1 - 1	5.52E-03
Xe-135m	1.06E-03	2.60E-01	7.38E-05	0.042	3.10E-06
Xe-135	1.39E-03	9.08E+00	1.28E-03	2.5	3.21E-03
Xe-137	4.79E-03	6.38E-02	9.15E-08	2.8	2.56E-07
Xe-138	4.54E-03	2.36E-01	2.41E-04	2.8	6.74E-04
			Moni	tor Reading:	1001202

(uCi/cc)

(uCi/sec)

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Table A2-6: Input for Main Steam Line Release Calculation

Choke Flow	Lamia Rele	《11》·受快》。这次方式就是35.3%,公司是需要要提供	Specific from Participants (Minite	volumento kale		te Raue Averstoneste	i aliadie conceni 1. aliadie conceni	ralian (1993)	ceas Thurch
1.05E+06	3.37E-	+06 0	338	3.55E+05	2.79	E+06	4.05		1.07575
			1						
	The local data is a second second second second second second	le A2-7: Calcula	A second s	dary Concentration		lose due to Ma		Release	
- Nucltice	Shom	Normalized .	Variation	THE STAND BREAD OF BRIDE STANDARD STATES	Concentration .	Had the St	Concentration	and the second second	Contribution
	- Inventity	Sector Para south a los	hiseittistikti 1		ê Brandâny		Z ORCER PREIM		
							and the state of t	elle tehr (etc) elle	
I-131	6.10E-02	2.26E-04	9.27E-04	2.9853E-05	2.77E-08	1.93E+02	2.76E-08	5.30E+04	1.46E-03
I-132	8.61E-02	3.19E-04	1.31E-03	2.9853E-05	3.90E-08	2.38E+00	2.85E-08	4.90E+03	1.40E-04
I-133	1.00E-01	3.72E-04	1.53E-03	2.9853E-05	4.55E-08	2.03E+01	4.39E-08	1.50E+04	6.58E-0
I-134	1.86E-02	6.92E-05	2.84E-04	2.9853E-05	8.47E-09	8.77E-01	3.62E-09	3.10E+03	1.12E-0:
I-135	2.73E-01	1.01E-03	4.14E-03	2.9853E-05	1.24E-07	6.61E+00	1.10E-07	8.10E+03	8.95E-04
Xe-131m	2.80E+00	1.04E-02	4.26E-02	2.9853E-05	1.27E-06	2.83E+02	1.27E-06	4.90E+00	6.22E-0
Xe-133	2.40E+02	8.90E-01	3.65E+00	2.9853E-05	1.09E-04	5.42E+01	1.07E-04	2.00E+01	2.15E-0
Xe-133m	4.20E+00	1.56E-02	6.40E-02	2.9853E-05	1.91E-06	1.27E+02	1.90E-06	1.70E+01	3.23E-05
Xe-135	7.60E+00	2.82E-02	1.16E-01	2.9853E-05	3.45E-06	2.60E-01	1.96E-07	1.40E+02	2.75E-0
Xe-135m	4.00E-01	1.48E-03	6.07E-03	2.9853E-05	1.81E-07	9.08E+00	1.67E-07	2.50E+02	4.17E-0
Xe-137	1.60E-01	5.93E-04	2.43E-03	2.9853E-05	7.26E-08	6.38E-02	6.10E-13	1.40E+02	8.53E-1
Xe-138	5.80E-01	2.15E-03	8.82E-03	2.9853E-05	2.63E ₇ 07	2.36E-01	1.12E-08	7.20E+02	8.04E-00
Kr-83m	3.70E-01	1.37E-03	5.62E-03	2.9853E-05	1.68E-07	1.83E+00	1.12E-07		0.00E+00
Kr-85	7.60E+00	2.82E-02	1.16E-01	2.9853E-05	3.45E-06	4.48E+00	2.92E-06	1.30E+00	3.80E-06
Kr-85m	1.50E+00	5.56E-03	2.28E-02	2.9853E-05	6.81E-07	9.40E+04	6.81E-07	9.30E+01	6.33E-05
Kr-87	9.80E-01	3.63E-03	1.49E-02	2.9853E-05	4.44E-07	1.27E+00	2.47E-07	5.10E+02	1.26E-04
Kr-88	2.80E+00	1.04E-02	4.26E-02	2.9853E-05	1.27E+06	2.84E+00	9.79E-07	1.30E+03	1.27E-03
Kr-89	8.40E-02	3.12E-04	1.28E-03	2.9853E-05	3.82E-08	5.10E-02	1.71E-14	1.20E+03	2.05E-11
100000 (1797)	والبوار متديد والأسر				1			Total Dose	6.89E-0

*Release Constant = X/Q * duration * release rate

Lotal Dose 6.89E-03

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Table A2-8: Main Steam Line Release Thyroid Dose Calculation

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New York	de Contennation	Second Defendent	Thyroid Doxe
	AUGY MUSE	rem periodi intec	Otema)
I-131	2.76E-0	1.30E+06	3.58E-02
I-132	2.85E-0)8 7.70E+03	2.20E-04
I-132	4.39E-0		9.66E-03
I-134	3.62E-0)9 1.30E+03	4.71E-06
I-134	1.10E-0		4.20E-03
1-135		A CONTRACTOR OF	

المالية الروسونية واليقو فيعدنه

F	N	F	R	C	0	Μ
Elect	B WI	Element		-date		BF BR
	EXCEN	ence	-cver	y proje	ict éve	a and a

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Table A2-9: Main Steam Line Reading at Release

Nordhile	Cancemperson	That Line	Comecutivation 1 titomentica Simickawa
	s (nutch/te(c))	I Ronuirs	(juCizec)
I-131	9.27E-04	1.93E+02	9.23E-04
I-132	1.31E-03	2.38E+00	9.77E-04
I-133	1.53E-03	2.03E+01	1.47E-03
I-134	2.84E-04	8.77E-01	1.29E-04
I-135	4.14E-03	6.61E+00	3.73E-03
Xe-131m	4.26E-02	2.83E+02	4.25E-02
Xe-133	3.65E+00	5.42E+01	3.60E+00
Xe-133m	6.40E-02	1.27E+02	6.36E-02
Xe-135	1.16E-01	2.60E-01	8.04E-03
Xe-135m	6.07E-03	9.08E+00	5.62E-03
Xe-137	2.43E-03	6.38E-02	4.65E-08
Xe-138	8.82E-03	2.36E-01	4.67E-04
Kr-8_3m	5.62E-03	1.83E+00	3.85E-03
Kr-85	1.16E-01	4.48E+00	9.90E-02
Kr-85m	2.28E-02	9.40E+04	2.28E-02
Kr-87	1.49E-02	1.27E+00	8.62E-03
Kr-88	4.26E-02	2.84E+00	3.34E-02
Kr-89	1.28E-03	5.10E-02	1.60E-09
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STAMPEDE User Supplied Information Revision 70.3.3 9/28/2011 ther Name: Unit Vent Alert Date/Time: 12/17/2013 15:24 Concents: User Supplied Information Million ological Data Inputs; Grannel level and velocity; Grannel level wind from; Uner-selected Stability Glass 13.2 mi/hr 180 degrees in para Stability Class: "D - Mentral" Munifored Unit Vent Release: Unit Vent Release Rate entered: 2.502+006 nCl/sec Reactor Shufdown Date/Time: 12/17/2013 14:24 Referes Start Date/Time: 12/17/2013 15:24 Extinuated Referse Doration: 1.00 hours Nation Mature: **Gap** Inventory Calculated NCIRLE CAS release rate: 1.192-006 aCi/sec NOHLEGAS IODENE PARTICILATE aCi/sec Nacide oCi/sec Nuclide Norlice **zCi/sec** 2536+004 1128-003 Ki-8384 1-131: Co-134: 1.05E+000 Kr-85: 1058-004 I-132: 3188+008 CS-137: 2.25E-001 7.06EH004 I-133: 6.0584008 CePr-144: Kr-8514 110E 001 1 134 KT-87: 901E+004 3.08EB+00B Ce-141: 2.80E-004 Kr-88: 1.748+005 1-135: 5.128+003 La-140 531E-004 Kr-89: 3.092-001 Mo-99; 3.43B-001 Xe-13116 312E+003 RafRb-166: 8258-005 Xe-333. 627E+005 R=-165: 2.508-004 Xe IX3M 191E+004 Sr/K-90: 9,108-005 No 115: 1452+005 Sr-89: 1,828,000

Te-132:

Zr-95;

1355-001

3.138-004

12/17/2013 3:24:46 PM

Xz-135M

Xe-137: Xe-158;

.

814E+00B

9.538+600

2692+004



CALC. NO. STPNOC013-CALC-002

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REV. 1

DRILI	STAMPEDE Result		DDIT
	Revision 7.0.3.3 9/28/2011	Pagelof2	
Date/Time: 12/17/2013 15:24 Connsents:	nt Alert		
CARLENCE THE CONTRACTOR STRATE	Finene Inform	EDIN. AND AND AND ADDRESS OF ADDR	an a
Bistence	Plums Trues Line	CHIQ Value	CHINO BUPL
(milos)	(hoors:minules)	(secim?)	(11c/m ²)
0.5	0:02	2.6868-005	24368-005
1.0	0:05	1.0528-005	9.1108-006
2.0	0:09	3.7558-005	3.151E-006
5.0	023	1.0048-005	7.373E-007
75	0:34	5.7048-007	3.8458-007
10.0	0:45	3.851E-007	2441E-007
20.0	131	1.5418-007	9.1095-008
	Measurable Dose Rates	PAC	Dose Rutes
	Interstor Whele Body	THE	lodina CDB
Distanca (miles)	Boble ges gemma (remfar)	ortornel + internel (rom/br)	Inyreid
0.5	0.039	0.016	0.137
1.0	0.0033	0.005	0.051
2.0	0.001	0.002	8.018
5:0	0.000	0.001	0.004
73	0.000	0.000	0.002
10.0	0.090	0.000	0.000
20.0	0.000	0.009	0.000
	Measurable Dases	PAC	Dones
		THEFE	Indiae CBR
Bistence	Immersion Whole Body	ortornal + internal	Thyraid
(milse)	nopia Eaz Lumare (1813).	(m: e1)	(rem)
0.5	0.009	0,016	0.137
1.0	0.003	0.006	0.051
2.0	0.001	0.002	0.618
50	0.090	0.001	0.004
7.5	0.000	0.000	0.002
100	0.000	. 0.000	0.001
20.0	0.060	0.000	0.000

12/17/2013 3:24:28 FM

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CALC. NO, STPNOC013-CALC-002

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PAGE NO. 34 of 49

REV.1

STAMPEDE Results Information Revision 7.0.3.3 9/26/2011 Page 2 of 2 Calculations Completed RESULTS Method of Projection Wind Velocity: 13.2 mi/hr Release Rate: 1.19E+005 uCidsec STAMPEDE Wind Direction: 180 Offsite Bose Projection (rem) 1 mile 3 miles 5 miles 10 miles TEDE 0.006 0.002 0.001 0.000 CDE 0.051 0.018 0.004 0.001 Projected duration of release: 1.0 hours A General Emergency Requires a Protective Action Recommendation EVACUATE ZONE(S): 1 SHELTER IN PLACE ZONE(S): 2 AFFECTED DOWNWIND SECTORS: R, A, B All Remaining Zones Go Indones And Monitor EAS Radio Station Based on a Dose Rate Projection of >3 mrem/in (Immersion Whole Body Noble Gas Gamma) at the Site Boundary (1 Mile) for 15 minutes or longer the Emergency Classification Initiating Condition RA1 (ALERT) has been met. . PERFORMED BY: 12/17/2013 3:24:44 PM Name Date/Time REVIEWED BY: Rad Manager/Radiological Director Date/Time

12/17/2013 3:24:28 PM



CALC. NO. STPNOC013-CALC-002

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PAGE NO. 35 of 49

REV.1

	Revision 7.0.3.3 9/28/2011	DKILL
Date/Time: 12/18/2013 07:54 . Comments:	Ther Name: SteamLine Site Alert	
	- <u>-</u>	
क के भिक्रत के की कार्यप्रकार के सिंहत है जो कि की कि की है। इस की भी कि की	Ther Supplied Information. Participation	and the stage start of the start start
Meteorological Data Iogues:		
Ground level word velocity:	13.2 mi/ár	
Ground level wind from:	180 degrees	
Uker-selected Stability Class		
Stability Class:	"D - Neuiral"	
Monitored S/G Tube Rupiure B	elezer:	
Stean Aclinity:	4.502+040 aCMrc	
Steam Flow Rate:	1.050 mHs/hr	
Reactor Shutdown Date/Time:	12/18/2813 06:54	
Release Start Date/Time:	12/18/2013 07:54	
Extensied Release Dorotion:	1.00 hours	
ana ang ang ang ang ang ang ang ang ang		
Naclide Mattere:	Nobie Cas + Iodene	
Isonat as percent of noble gas:	D.296	

.

Calculated NOBLE CAS release rate: 1.1919-007 oCi/sec

NOHL	EGAS	IOH	ME.	PARTIC	ULATE
Nuclide	aCi/sec	Nachide	BCI/SEC	Machike	oCi/sec
Kr-5314	1.148+004	I-131 :	3.0584008	Cs-134:	0.008+000
Kr-85:	3.438+006	1-132:	3228+008	Cs-137:	0.008+000
Kr-25M	\$798+004	I-133:	4.888+003	CePr-144:	0.00E+000
Ka-87:	2555+004	1-134:	4228+002	Ce-141:	0.00E+000
Kir-BR	9.896+004	8-135:	1.238+004	La-140;	0.00E+000
Kr-29;	4.25E-003			140-99:	0.00E+000
Xe-131M	1.268+005			Ru/Rh-106	0.00E+000
Xz-133:	1.088+007			Rp-189:	0.00E+000
Xr-13314	1.878+006			Sr/¥-90:	0.00E+000
Xz-135:	3.1928+005			Sr-89:	0.00E+000
Xe-135M	1.238+003			Te-132:	0.00E+000
Xz-137:	1278-001			Zr-95:	0.00E+000
Xi-138:	1.308+003				

12/18/2013 7:55:19 AM



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CALC. NO. STPNOC013-CALC-002 REV.1 , PAGE NO. 36 of 49

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		a ta ang ang ang ang ang ang ang ang ang an
DRILL	STAMPEDE Results Information Reason 7.033 9/28/2011 Page 1 of 2	DRILL
Date/Time: 12/18/201307:54	User Name: Steam Line Site Alert	
Continents:		
	an a	
an an taon an		
Distance	Pinmo Truval Amo	CHEQ DIPL
(miles)	(hours: min piss) (hours)	(146°m [*])
0.5	0.02 2.6866-005	2436E-005
10 20	009 10828-005 009 37558-005	9.110E-006 3.151E-006
50	023	7.3738-000
75	034 5.7048-007	3.845E-007
10.0	G45 3.851B-007	2.441E-007
20.0	131 15418.007	9.1095-006
	Measurable Doss Rates	AG Doce Rules
	Insusering Wiple Body	Indias CDE
Distance	mille gas gamme external futerant	Thyreid
(milos)	(rom/br)	(remlins)
0.5	0.011	<u> 8.135</u>
1.0	0.0043 0.007	B.050
20 50	6003 0003 0000	0.017 0.004
- 55 75	0.000 0.000	DANZ
100	0.0.0	0.001
200	0.000	8.000
e de la compañía de l		<u> </u>
	Measurable Dases	AGDoses
	31(191)	Indiae CDE
Distance (miles)	Immersian Whole Hody extended finisms mobile gai genemes (rem) (rem)	Thyroid (rem)
	그 상황에 들었다. 이 이 가슴을 물었다. 이 이 이 것이 같아.	
05	0,011 0,019 0,067	0.135 0.050
26	0.021	8017
50	0.001	0.004
75	0.000	8.002
100	0.000	0.001
200	0.000	0000

12/18/2013 7:54:42 AM



PAGE NO. 37 of 49

REV. 1

D	RILL	STAMPEDE R Revision 7.0.3.3 9/28/2		nformation Page2of2	DRILL
		Cakulati RESU	ous Comple		na na kang tang Kabupatana Ka
X	ethod of Projectica: STAMPEDE	Wind Velocity: Wind Direction:	13.2 mi/hr 150	Release Rates	1.19E+007 uCosec
Offs	ite Dose Projection (rem)r			
	I mile	1 miles	5 miles	10 miles	
TEDE	0.007	0.003	0.001	0.000	
CDE	0.050	0.017	0.004	0.001	
Projec	ied duration of release: 1	.0 hours			

A General Emergency Requires a Protective Action Recommendation

EVACUATE	ZONE(S):	1
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SHELTER IN PLACE ZONE(S): 2

AFFECTED DOWNWIND SECTORS: R, A, B

All Remaining Zones Go Indoors And Monitor EAS Radio Station

Based on a Dose Rate Projection of \geq 3 mem/hr (Immersion Whole Body Noble Gas Gamma) at the Site Boundary (1 Mile) for 15 minutes or longer the Emergency Classification Initiating Condition RA1 (ALERT) has been met.

PERFORMED BY:

12/18/2013 7:55:14 AM Date/Time

REVIEWED BY:

Rad himager/Rullological Director

Name

Date Time

12/18/2013 7:54:42 AM



CALC. NO. STPNOC013-CALC-002

REV.1 PAGE NO. 38 of 49

121



STAMPEDE User Supplied Information DRI Revision 7.0.1.3 9/26/2011

Ther Name: Unit Vent Sile Area DateTime: 12/17/2013 15:25 Comments:

User Supplied Information

(1,1,2)

Antonia in Anna

				·	1. <u>1. 1</u>
м	eteorological	The fai Your		· · · ·	·
	Ground level			13.2	iður
	Greendlevel			130 des	er ees
	User-selecte	dStabill	y Class		
	Shisily Che	35:		"D-1%	efcel ^m
		11 - A 44		1.11	

Station 1

Monifored Unit Vent Relages Duit Vent Release Rain entried: 2.502+007 pCi/sec

Beatfar Shuidson Date/Lime: 12/17/2013 14:25 Release Start Date/Time: 12/17/2013 15:25 Estimated Release Duration: 1.00 hours

Nuclide Mature:

Cap Inventory

140

NOH	EGAS	KODI	NE	PARTIC	TH A772
Anclicle	aCi/sec	Nachde	wCi/sec	Nucide	DCI/sec
G-83ME	2.52E+665	1 131 :	3.128+004	Cz-134:	1.05B+001
Gr-85:	1.058+005	I-13 2:	3.188+004	Cs-137:	8.24E+000
G-85M	7.068+005	E-133:	504B+004	Or.Fr-144:	2.10E-093
G-87:	9.03B+005	I-134 :	3.088+004	Ce-141:	2.BIE-003
G-88-	1.736+005	I-J35:	5128+004	La-140:	5.31E-003
G-89:	3.142+000			Ma 99 :	3ASB+CC0
Ge-1311&	3.126+004	11 I. I. I. I. I.		Ra/Rh-165:	8248-004
Ee 133 :	6.22E+005			Rs-105:	2,508-008
D-13314	1918+005			Scr.40:	9.10E-004
G-135:	1.456+006	·	and the first second	Sr-89:	1.828-002
2-135kA	8.1823+004	•		Te-133:	1358+000
	9.746+001			Zu-95.	3.138-003

12/17/2013 3:25:33 PM



CALC. NO. STPNOC013-CALC-002

REV.1 PAGE NO. 39 of 49

STAMPEDE Results Information DRILL DRILL Resiston 7.0.3.3 9/28/2011 Page lof 2 Ther Manne: Unit Vent Site Area Date/Time: 12/17/2013 15:25 Comments: Frame Information Distance Plume Travel Line CHI/Q Value CHUQ DIPI. (milos) (koura:minu@oa) (sec/m") (uom') 2.686E-005 0,5 1,0 2,0 5,0 7,5 0:02 2.436B-005 9.110B-005 3.151B-005 0:05 0.09 3.755E-005 1.0048-005 0:23 7.373E-007 3.845E-007 0:34 S708-007 10.0 20.0 0:45 3.851E-007 24418-007 1:31 1.541E-007 9.1098-008 Measurable Dose Rates PAG Dose Rates Innersian Whole Body TRUE Indian CDE Distance noble ges gumme (remikr) Thyroid (nean/hal) (miles) 0.088 0.S 1.0 1.364 0.510 0.160 0.050 2.0 5.0 7.5 0.012 0.021 0.176 0.005 0.003 0.002 0.041 0.021 0.014 0.002 10.0 0.001 20.0 0.000 0.001 0.005

	BOESSDESDE LJOSES	PAG Doses		
Bistanca (miloc)	Immersion Whole Hody noble gragement (rem)	IB)E Sutomet + internet (rum)	lodine CDB Thyroid (rem)	
0.5	0.065	0.169	1.364	
1.0	0.033	0.000	0.510	
20	0.012	0.021	0.176	
5.0	CL003	0.005	0.041	
75	0.002	0.005	0.021	
10.0	0.031	0.002	0.014	
20.0	0.000	0.001	0.005	

12/17/2013 3:25:21 PM



CALC. NO. STPNOC013-CALC-002

PAGE NO. 40 of 49

REV.1

STAMPEDE Results Information Revision 7.0.5.3 9/29/2011 Page2af2 Calculations Completed RESULTS Method of Projection: Wind Velocity: 13.2 miller Release Rate: 1.19E+007 uCide STAMPEDE Wind Direction: 180 Offsite Dose Projection (rem). I mile 3 mileo 5 milles 10 miles TEDE 0.060 0.021 0.005 0.002 CDE 0.510 0.176 0.041 0.014 Projected duration of release: 1.0 hours A General Emergency Requires a Protective Action Recommendation EVACUATE ZONE(S): 1 SHELTER IN PLACE ZONE(S): 2 AFFECTED DOWNWIND SECTORS: R, A, B All Remaining Zones Go Indoors And Monitor EAS Radio Station Based on a Site Boundary (1 Mile) Dose Projection 20.1 rem THDE and/or 0.5 rem Thyroid CDE the Emergency Classification Initiating Condition RSI (SITE AREA EMERGENCY) has been met. PERFORMED BY:

		12/17/2013 3:25:28 PM	
REVIEWED BY	Name	Date/Thing	
REVIEWED BY:			
 Rad Manag	ser/Kadiological Director	 Date/Time	

12/17/2013 3:25:21 PM

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CALC. NO. STPNOC013-CALC-002 **REV.** 1

PAGE NO. 41 of 49

STAMPEDE User Supplied Information DRII DRII Revision 7.0.3.3 9/28/2011 Date/Time: 12/17/2013 15:28 User Name: Steam Line Site Area Composents: User Supplied Information Meteorological Data Inputs: Ground level wind velocity: 13.2 mistr Ground level wind from: 180 degrees User-selected Stability Class "D - Renival" Stability Class: Monitored S/G Tube Rupture Release: Steam Activity: Steam Flow Rate: 4.50E+001 aCEcc 1.050 mByler Reactor Shutdown Date/Time: 12/17/2013 14:25

Release Start Date/Time: 12/15/2013 15:28 Etimates Release Duration: 1.00 hours

Nuclide Mattere: Nohle Gas + Toffne Indian as percent of noble gas: 8,2%

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Calculated NOHLE GAS release rate: 1.20E+008 uCi/sec

NOHL	EGAS	KOD	NE	PARTIC	TLATE
Nuclide	uCli ista	Naclife	BCifec	Nucāde	o Citsec
K7-5312	1.148+005	3-131:	3.078+004	Cs-184:	0:03E+000
Kr-85:	3.458+006	1-132:	3238+004	Cs-137:	0.00B+000
Kr-8514	5.8IB+005	I-13 3:	4903+004	Ce/Pr-144:	0.008+000
Kr-87:	2.55E+005	8-134:	4.228+008	Ce-141:	0.008+000
Kr-88:	9918+005	8-135:	1.248+005	La-140:	0.00E+000
14 -29 :	3.92E-002			14 5-99 :	0.00E+600
Xe-IXIM	1.278+006			Ra/Rh-105	0.0008+000
X2-133:	1.082+008			En-163:	0.00E+000
Xe-13314	1.888+005			Sc/¥-90:	0.008+000
Xe-135:	3.192+005			Sr-89:	0.00000+0000
Xz-135M	1.218+004			Te-132:	0.00E+600
Xa-137:	1.198+000			Zr-95:	0.00E+000
X2-138:	1.3492+004				

12/17/2013 3:29:03 PM

ENERCON Radiological Release Thresholds CALC. NO. STPNOC013-CALC For Emergency Action Levels REV. 1 Facelience—Every project Every drop Attachment 3	2-002
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DRI	IL .	STAMPEDE Resu Reddon 7.0.13 9280011	A state of the	DRILL
	2/17/2013 15:28	User Menne: Steam	Line Site Area	
Continents:	a di peterto di suber di			en en e
			<u></u>	
2 - W 200 2 2 1 4 1 4	a and the second se	Finme Infor	anation .	n forsk af lest orden i Stand och bok som s Stand och som standare som
Distant		Plane Travel Hms	CBIQ Value	CHUQ DEPL
(wilos)		(Roars:minutes)	(milan)	(sacks")
0.5	1.11.14	0.02	2.0365-005	2436-005
10	. No. 1	0:05	LOBZEOOS	9.1108-005
2.0		0:09	3.7556-006	3.151E-006
50	· · · · · · · · · · · · · · · · · · ·	0.23	1.0046-005	7.373E-007
75		0:34	5.7548-007	3.845E-007
10.0		0.45	3.851E-007	2441E-007
20.0		131	1.5418-007	9.100E-008
		Measurable Dose Rates		GDose Rates
		Lange and the second		the state of the state of
Distance		Immersion Whole Sody	ertornel finisrael	Indine CUR Terreid
(miles)		soble rus renam (rendin)	(remily)	(remlist)
0.5 1.0		0.111 0.0423	0.189	1.154
		0.015	0.072	0.506
20 50	· · ·	0.005	0.025 0.006	0.175 0.041
75	•	0.003	0.03	0.021
100		0.001	0.002	0.013
200		0.001	6.001	0.065
		Loresarable Doses	PA PA	C-Doses
de la composición de		1. 17 6 (2. C. H. C. L.		
Distance		Immersion Whole Body	analite interest	Jodine CDE Thyreid
(mibes)	a shekara a	nebla gas gamma (ram)	farai	(rem)
0.5		am	0.129	1354
05 10		0.042	0.129 0.672	1.104
20		0.015	0.025	0.175
50		0.004	0.005	0.041
75		0.002	0.003	0.021
100	S. 1994	0.001	0.000	0.013
20.0	and the second second	0.00	0.001	0.005
	1 1 1 A 1			

12/17/2013 3:28:53 ###

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PAGE NO. 43 of 49

DRII	L	STAMPEDE R Revision 7.0.3.3 9/28/2	011 Page	ermatian 20f2	DRILL
	n de la sector de participante de la sector d 	Calculati RESU	ons Completed LTS		anayan da ku
Method of P STAM		Wind Velocitys Wind Direction:	13.2 mi/br 180	Release Rate:	1.20E+008 aCi/sec
Offsite Dose P	rojecticu (rem)	1			
	1 mile	2 miles	ô miles	10 miles	
TEDE	0.072	0.025	0.006	0.002	
CDE	0.506	0.175	0.041	0.013	
			0.071	0.015	

Projected duration of release: 1.0 hours

A General Emergency Requires a Protective Action Recommendation

EVACUATE ZONE(S): 1

SHELTER IN PLACE ZONE(S): 2

AFFECTED DOWNWIND SECTORS: R, A, B-

All Remaining Zones Go Indoors And Monitor EAS Radio Station

Based on a Site Boundary (1 Mile) Dose Prejection 20.1 rem TEDE and/or 0.5 rem Thyroid CDE the Emergency Classification Initiating Condition RSI (SITE AREA EMERGENCY) has been met.

PERFORMED BY:

12/17/2013 3:29:00 PM Date/Time

REVIEWED BY:

Name

Rad Manager Kadiological Director

Date/Thue

12/17/2013 3:28:53 PM



PAGE NO. 44 of 49

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STAMPEDE User Supplied Information DKI Revision 7.0.3.3 9/28/2011 Date/Trate: 12/17/2013 15:25 Ther Mante: Unit Vent General Comments:

quiden set

	User User	Supplied In	formation
Meteorological Data Ingests:			· · ·
Crossed level wind whetity:	13.2 mi/hr	• .	
Ground lovel wind from:	130 degrees		
User-selected Stability Class	a Tist		• •
Statenty Chase:	"D - Neutral"		

Munifored Unit Vent Release. Unit Vent Release Rate entered: 150E+008 pCl/sec

Reacter Shutdown Date/Time: 12/17/2013 14:26 Release Start Date/Time: 12/17/2913 15:26 Etimated Release Doration: 1.00 hours

Noclide Mixture:

. . .

(2,2)

Capleventory

Calculated NOBLE GAS release rate: 1.195-098 aCi/sec

NOBL	EGAS	FOD	ne		PARTIC	ULATE
Nuchde	aCi/sec	Nuclide	nCi/sec		Nuclick	oCi/sec
Kr-Silf	2.528+005	1-131:	3.128+005		Cs-134:	1.053+002
Kr-85:	1.05E+005	F-132:	3.18B+005		Cr-137:	£25E+001
Kr-8512	7/066=006	I-133 :	6.04E+005		Ce/Pr-144:	2.108-002
Kr-87:	9:03E+006	1-134 :	3.088+005		Ce-141:	2.846-002
Kr-85:	1.73E+607	I-135:	5,128+005	1	Lz-140:	531E-CM
Ka-89:	3102+001				Ma-99:	3.438+001
Xe-131M	3.128+005				Ra/Rb-106:	825B-003
Xa-133:	6.22E+007	. *			Ra-103:	2.508-002
Xe-133ME	191E+006	·· .·			Sc/¥-90:	9.IOR-003
Xe 135	1458+007		····	· · · · · ·	Sr-89:	1.828-001
Xe-1358£	8168+065			11	Te-322:	1.35E++001
X2-137.	9.548+002				Zr-95:	3.136-002
Xe-138:	2.668+606					· · · ·

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12/17/2013 3:26:37 PM



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Radiological Release Thresholds for Emergency Action Levels Attachment 3

CALC. NO. STPNOC013-CALC-002

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PAGE NO. 45 of 49

REV. 1

NDITI	STAMPEDE Result	s Information	ΤΠΠΠ
UKILL	Revision 7.0.3.3 9/28/2011	Page 1of 2	DRILL
Date/Time: 12/17/2013 15:28 Comments:	User Name: Unit Ver	ut General	
C. C. C. C. C. R. C. Marked Street, and a	and a second	alinet management of the second second	and the state of the
Distance	Plame Ravel Time	CHLQ Value	CREW DEPT
(miles)	(hours:minutes)	(soc/m [*]]	(sec/m*)
0.5	0:02	2.6868-005	24368-005
1.0	0:05	1.0328-005	9.1108-006
2.0	0:09	3.7558-005	3.1518-006
5.0	023	1.0048-005	7.373E-007
7.5	0:34	5.7048-007	3.845E-007
10.0	0:45	3.851E-007	2441E-007
20.0	1-31	1.541B-007	9.1092-008
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0.5	0,879	1.598	13.646
1.0	0.3322	0.601	5.099
2.0	0.117	0.219	1,762
5.0	0.029	0.050	0411
7.5	0.015	0.027	0.214
100	0.010	0.017	0.135
20.0	0.003	0.006	0.050
	Measurable Dases	PAG	Dases
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1.0	0.332	0.60)	5.099
20	0.117	0.210	1.762
5.0	0.029	0.050	0.411
7.5	0.016	0.027	0.214
10.0	0.010	0.017	0.135
20.0	0.003	0.005	0.050

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	in the second	lations Complete		
Method of Projection: STAMPEDE	Wind Velocit Wind Directio	ULTS 37 13.22201/hr nr 180	Reicare R	ate: 1.19E+008 uCiAsec
Offsite Dose Projection (1 mile TEDE 0.601	em): 2 mHes 0.210	5 miles 0.050	10 miles 6.017	
CDE 1099	L762	6411	0.135	
Projected duration of release	:: LObours			
A General Emer	gency Requires :	a Protective	Action Reco	mmendation
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REV.1

DRILL STAMPEDE User Supplied Information DRILL Revision 7.0.3.3 928/2011 DRILL

Date/Time: 12/17/2013 15:30 Comments: User Name: SteamLine General

Ther Supplied Information

Meteorological Data Ingusts: Ground level wind velocity: 13.2 mi/hr Ground level wind from: 130 degrees User-selected Stability Class Stability Class: "D - Neutral"

Manihared S/G Tube Raphare Release: Steam Activity: 4.568+062 a CaArt Steam Flow Rate: 1.650 mH/hr Reactor Shutdewn Date/Time: 12/17/2013 14:30 Release Stort Date/Time: 12/17/2013 15:30

Release Short Date/Time: 12/17/2013 15 Estimated Release Noration: 1.80 Lours

Nuclide Mixture: Noble Gas + Iodine Luñus as percent af noble gas: 8.246

Calculated NOBLE GAS release rate: 1.2019-069 oCl/sec

NOBL	EGAS	IOD	NE	PARTIC	ULATE
Nuclida	nCi/sec	Nuclide	wC3/sec	Nuclis	aCi/sec
ET-BILL	1.148+006	I-131:	3.0700-005	Cs-134;	0.00E+000
Kr-85:	3:45E+007	I-132 :	3.2493+005	Cs-137:	0.008+000
KP-85M	5.826+006	I-133:	4.905+005	Ce/Pr-144:	0.0089+000
Kr-87:	2.568+006	I-134:	4245+004	Ce-141:	0.00B+000
Kr-88:	9:92E+006	I-135 :	1.243+006	La-140:	0,008+000
Kr-89:	4.136-001			Ma-99 :	0:008+000
Xe-Blim	1.27E+007			Rn/Rh-106:	0.00E+600
Xz-133:	1.088+009			Ra-168:	0.008+000
X2-13334	1365+007			Sr/Y-90:	0.00E+000
Xp-135:	3.198+007			Sr-89:	0.005+000
Xe-1353£	1.238+005			Te-132:	0.0022+000
Xz-137:	1.245+001			Za-95:	9.00E+000
Xr-138	1.3SE+005				

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Radiological Release Thresholds for Emergency Action Levels Attachment 3

CALC. NO. STPNOC013-CALC-002 REV. 1

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Method of Pre STAMP		RESU	ous Completed	
		Wind Velocity: Wind Direction:	13.2 mi/hr	Release Rate: 1.20E+009 uCièse
Offite Dose Pr				
EDE	1. mile 0.727	I miles 0.254	5 miles 8.963	10 miles 0.022
DE	5.058	1.747	D.407	0.134
rojected duration	ofrelezse: 1.0 b	0183		
A Genera	l Emergenc	y Requires a l	Protective A	ction Recommendation
EVACUA	TE ZONE(S):	1, 2	•	
SHELTER	R IN PLACE ZO	ONE(S): 6,11		
AFFECT	ED DOWNWI	ND SECTORS:	-R. A. B	· ·
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The particulate channel is used as part of the Reactor Coolant Pressure Boundary (RCPB) leakage detection system. The sensitivity and response time of this part of the leakage detection system, which is used for monitoring unidentified leakage to the Containment, are sufficient to detect an increase in leakage rate of the equivalent of one gal/min within one hour. Elements of this monitor, including the indicator mounted in the RMS CR cabinet, are designed and qualified to remain functional following a Safe Shutdown Earthquake (SSE), in compliance with RG 1.45. Further information on the RCPB leakage detection system is presented in Section 5.2.5.

11.5.2.3.3 <u>Unit Vent Monitor</u>: The unit vent monitor samples the plant vent stack prior to discharge to the environment and monitor for particulates, iodine, and noble gases.

The unit vent particulate and iodine monitor draws representative air samples from the plant vent stack via isokinetic nozzles in the stack, and directs them through a moving filter paper monitored by a shielded beta-sensitive scintillation detector. The sample stream then passes through a charcoal collector, where collected iodine is monitored by a shielded gamma-sensitive scintillation detector. The sample is then returned to the vent stack.

A separate wide-range gas monitor is provided for the unit vent. The monitor has two isokinetic nozzles for sampling during both normal and accident conditions. The stack samples pass first through a sample conditioning unit which filters particulates and iodine and may be used to take grab samples. The samples then pass through the shielded detector assembly, which uses three detectors to cover the complete range required. The low range detector uses a beta-sensitive plastic scintillator-photomultiplier (PM) tube. The mid-range and high-range detectors use cadmium telluride (CdTe), chlorine-doped, solid-state sensors. This wide-range gas monitor satisfies the requirements of NUREG-0737, Item II.F.1 for provisions for sampling plant effluents for iodines and particulates and for noble gas effluents from the plant vent.

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11.5.2.3.4 <u>Control Room Electrical Auxiliary Building Ventilation Monitors</u>: The CR/EAB ventilation monitors are Class 1E monitors which continuously assess the intake air to the CR for indication of abnormal airborne radioactivity concentration. Each monitor assembly is powered from a separate electrical power source. In the event of high radiation CR emergency ventilation operation is initiated (Section 7.3.2). Failure of a monitor is alarmed in the CR.

Each monitor assembly is comprised of a recirculation pump, beta-sensitive scintillation detector, four-pi lead shielding, oheck source, stainless steel sample gas receiving chamber, and associated electronics.

11.5.2.3.5 <u>Condenser Vacuum Pump Monitor</u>: Gaseous samples are drawn through an offline-system by a pump from the discharge of the vacuum pump exhaust header of the condenser. This channel monitors the gaseous sample for radioactivity which would be indicative of an SG tube leak, allowing reactor coolant to enter the secondary side fluid; this monitor complements the SGBD monitors in indication of a SG tube leak. The gaseous radioactivity levels are monitored by a single detector in a manner similar to the unit vent wide range gas monitor.

11.5.2.3.6 <u>Spent Fuel Pool Exhaust Monitors</u>: The SFPE monitors are Class 1E and are identical to the CR/EAB ventilation monitors described in Section 11.5.2.3.4 except that they sample the exhaust from the FHB. In the event of high radiation the monitors initiate emergency operation

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11.5.2.5.1 <u>Gaseous Waste Processing System Inlet Monitor</u>: The GWPS inlet monitor employs a gamma (NaI crystal) sointillator/photomultiplier tube combination to measure the radioactivity level of the waste gases entering the GWPS. The monitor is used in conjunction with the GWPS discharge monitor to measure overall effectiveness of the GWPS.

11.5.2.5.2 <u>GWPS Discharge Monitor</u>: This monitor is similar to the GWPS inlet monitor and is installed upstream of the GWPS discharge valve. Upon detection of high radioactivity or monitor failure, the GWPS discharge valve, FV-4671, is automatically closed.

11.5.2.5.3 <u>Main Steam Line Monitors</u>: Each MS line is monitored by an ATL monitor consisting of a Geiger Mueller (GM) tube detector and an ion chamber detector with overlapping ranges. The detectors are shielded by 3 in. of lead.

The monitors are designed to monitor gross gamma activity in the steam line and provide a basis for determining possible atmospheric releases from the MS power-operated relief valve (PORV), SG safety valves, and/or auxiliary feedwater pump turbine.

The monitors provide a dose rate range equivalent to 10^{-1} to $10^{3} \,\mu$ Ci/cm³ xenon-133. Based upon core inventory, the ratio of xenon-133 to other nuclides in the fuel can be determined. In order to obtain the above concentrations of xenon-133 in the main steam line, a large primary-to-secondary leak must be present coincident with a large amount of fuel failure. The presence of xenon-133 indicates other radioactive isotopes are present.

Using the relative ratios of isotopes present in the MS line, a computer model for determination of dose rates from these isotopes, detector response curves, the thickness of the MS line, and the geometry of the MS line relative to the detector, the dose rate equivalent to MS line concentration is obtained. The quantity of radioactive effluents released is obtained by multiplying the xenon-133 equivalent MS line concentrations by the isotope ratio times the steam release rate.

These detectors are safety-related Class 1E and meet the requirements of RG 1.97 and NUREG-0737.

11.5.2.5.4 <u>Steam Generator Blowdown Monitors</u>: These monitors are identical to the MS line monitors and are adjacent to the SG blowdown lines in the Isolation Valve Cubicle (IVC).

The monitors are used as an aid in determining the source of SG blowdown radioactivity due to SG tube rupture or a large primary-to-secondary leak.

These-detectors are safety-related-Class-1E-and-meet the requirements-of-RG-1-97.-

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11.5.2.5.5 <u>Main Steam Line High Energy Gamma (N-16) Monitors</u>: Each main steam line is monitored by an ATL NaI scintillation detector. These detectors were installed to monitor the status of steam generator primary to secondary tube leaks and to provide a diagnostic tool for all individuals concerned with steam generator condition. These detectors are designed to detect high energy gamma activity in the 6 to 7.2 MEV energy range. High energy gamma activity in the main steam lines indicates the presence of N-16. The level of N-16 in the main steam lines is used to

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The new fuel assemblies are transported to the new fuel storage pit or to the new fuel elevator by the 15/2-ton, dual-service FHB crane. The 2-ton hoist of this crane is designed to handle new fuel assemblies. New fuel handling is discussed in detail in Section 9.1.4. Use of the 2-ton hoist of the 15/2-ton crane or of the fuel-handling machine to handle new fuel ensures that the design uplift of the racks will not be exceeded.

The new fuel storage pit is situated in the approximate center of each FHB. The floor of the new fuel storage pit is at El. 50 ft-3 inches. The new fuel storage pit access hatch is provided with a three-section protective cover at El. 68 ft. The fuel assemblies are loaded into the new fuel storage racks through the top and stored vertically.

9.1.1.3 <u>Safety Evaluation</u>. Units 1 and 2 of the STPEGS are each provided with separate and independent fuel handling facilities.

Flood protection of each FHB is discussed in Section 3.4.1. Flooding of the new fuel storage pit from fluid sources inside either FHB is not considered credible since all fluid systems components are located well below the elevation of the new fuel storage pit access hatch. A floor drain is provided in the new fuel storage pit to minimize collection of water.

The applicable design codes and the ability of the FHB to withstand various external loads and forces are discussed in Section 3.8.4. Details of the seismic design and testing are presented in Section 3.7. Missile protection of the FHBs is discussed in Section 3.5. Failure of nonseismic systems or structures will not decrease the degree of subcriticality provided in the new fuel storage pit.

In accordance with American National Standards Institute (ANSI) N18.2, the design of the normally dry new fuel storage racks is such that the effective multiplication factor will not exceed 0.98 with fuel of the highest anticipated enrichment in place, assuming optimum moderation (under dry or fogged conditions). For the unborated flooded condition, assuming new fuel of the highest anticipated enrichment in place, the effective multiplication factor does not exceed 0.95. Credit may be taken for the inherent neutron-absorbing effect of the materials of construction.

The new fuel assemblies are stored dry, the 21-in. spacing ensuring a safe geometric array. Under these conditions, a criticality accident during refueling and storage is not considered credible. Consideration of criticality safety analysis is discussed in Section 4.3.

Design of the facility in accordance with RG 1.13 ensures adequate safety under both normal and postulated accident conditions. The new fuel storage racks also meet the requirements of General Design Criterion (GDC) 62.

9.1.2 Spent Fuel Storage

9.1.2.1 <u>Design Bases</u>. The spent fuel pool (SFP) is a stainless steel-lined reinforced concrete pool and is an integral part of each FHB. All spent fuel racks are classified as seismic Category I, as defined by RG 1.29, and as ANS SC 3.

The spent fuel storage facility provides storage capacity for 1,969 high density absorber spent fuel racks in a honeycomb array in each unit. Two storage regions are provided in the SFP. Two of the

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Region 2 rack modules on the south end of the pool (modules #12 and #16) have not been installed. A Fuel Ultrasonic Cleaning system may be used in the open space designated for modules #12 and #16. The Fuel Ultrasonic Cleaning system is freestanding and is seismically qualified. It has no adverse effect on the fuel assemblies that are selected for cleaning; nor does it have an adverse effect on the design function of the spent fuel pool or its associated support systems. Figure 9.1.2-2 shows the pool layout for both Units 1 and 2. The six Region 1 rack modules are located in the northwest corner of the spent fuel pool.

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The Region 1 racks have 10.95-in. nominal center-to-center spacing between the cells. This region is conservatively designed to accommodate unirradiated fuel at enrichments to 4.95 weight percent. Region 1 storage cells are each bounded on four sides by a water box except on the periphery of the pool. The Region 1 spent fuel racks include a lead-in-guide to assist in depositing fuel assemblies into the fuel cell. Figure 9.1.2-3 shows a typical Region 1 spent fuel rack.

The reactivity characteristics of fuel assemblies which are to be placed in the spent fuel storage racks are determined and the assemblies are categorized by reactivity. Alternately, if necessary, all assemblies may be treated as if each assembly is of the highest reactivity class until the actual assembly reactivity classification is determined. Section 5.6 of the Technical Specifications provides the definitions of the reactivity classifications and the allowed storage patterns. Fuel assemblies are loaded into the racks in a geometrically safe configuration to ensure rack subcriticality.

Fuel assembly reactivity requirements for close packed storage and checkerboard storage are specified in the Technical Specifications. The boron concentration of the water in the spent fuel pool is maintained at or above the minimum value needed to ensure that the rack K_{eff} is less than or equal to 0.95 in the event of misplaced assemblies in the close packed storage areas or in checkerboard storage areas. Consideration of criticality safety is discussed in Section 4.3.

The Region 2 racks have a 9.15-in. nominal center-to-center spacing with fixed absorber material surrounding each cell. A sheet of neutron absorber material is captured between the side walls of all adjacent boxes. To provide space for the absorber sheet between boxes, a double row of matching flat round raised areas are coined into the side walls of all boxes. The raised dimension of these locally formed areas on each box wall is half the thickness of the absorber sheet. The boxes are fusion welded together at all these local areas. The absorber sheets are scalloped along their edges to clear these areas. Figure 9.1.2-4 shows a typical Region 2 spent fuel rack.

The axial location of the absorber with respect to the active fuel region is provided and maintained by the structure of each box. At the outside periphery of each rack, a sheet of absorber material is captured under thin stainless sheets which are intermittently welded all around to the box.

All rack surfaces that come into contact with fuel assemblies are made of annealed austenitic stainless steel. These materials are resistant to corrosion during normal and emergency water quality conditions. The racks are designed to withstand normal operating loads as well as to remain functional with the occurrence of an SSE. The racks are designed with adequate energy absorption capabilities to withstand the impact of a dropped spent fuel assembly from the maximum lift height of the spent fuel pit bridge hoist. The racks are designed to withstand a maximum uplift force equal to the uplift force of the bridge hoist. The 14-in. and 16-in. racks also meet the requirements of ASME Code, Section III, Appendix XVII. The high-density spent fuel racks meet the criteria of Appendix D to Standard Review Plan (SRP) 3.8.4.

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Shielding for the SFP is adequate to protect plant personnel from exposure to radiation in excess of published guideline values as stated in Section 12.1. A minimum depth of approximately 13 ft of water over the top of an array of 193 (full core) assemblies with 42 hours of decay is required to limit radiation from the assemblies to 2.5 mR/hr. or less.

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The FHB Ventilation Exhaust System is designed to limit the offsite dose in the event of a significant release of radioactivity from the fuel, as discussed in Sections 12.3.3, 15.7.4, and 9.4.2. However, no credit for the FHB Ventilation Exhaust System is taken in the LOCA and Fuel Handling accident in Chapter 15.

The FHB is designed to prevent missiles from contacting the fuel. A more detailed discussion on missile protection is given in Section 3.5.

In addition, space is provided for storage of fuel during refueling inside the RCB for 64 fuel assemblies in four 4 x 4 modules having 16-in. center-to-center spacing (Figure 9.1.2-1A). These modules are firmly bolted in the floor.

9.1.2.2 <u>Facilities Description</u>. The FHB abuts the south side of the RCB and is adjacent to the west side of the MEAB of each unit. The locations of the two FHBs are shown in the station plot plan on Figure 1.2-3. For general arrangement drawings of the spent fuel storage facilities, refer to Figures 1.2-39 through 1.2-48 as listed in Table 1.2-1.

The spent fuel storage facilities are designed for the underwater storage of spent fuel assemblies and control rods after their removal from the reactor vessel. The spent fuel is transferred to the FHB and handled and stored in the spent fuel pool underwater. The fuel is stored to permit some decay, then transferred offsite. For a detailed discussion of spent fuel handling, see Section 9.1.4.

The SFP is located in the northwest quadrant of each FHB. The floor of the pool is at El. 21 ft-11 in., with normal water level at El. 66 ft-6 inches. The top of a fuel assembly in a storage rack does not extend above the top of the storage rack which is El. 39 ft-10 in. maximum. The fuel assemblies are loaded into the spent fuel racks through the top and are stored vertically.

9.1.2.3 <u>Safety Evaluation</u>. Units 1 and 2 of the STPEGS are each provided with separate and independent fuel handling facilities. Flood protection of each FHB is discussed in Section 3.4.1. A detailed discussion of missile protection is provided in Section 3.5.

The applicable design codes and the various external loads and forces considered in the design of the FHB are discussed in Section 3.8.4. Details of the seismic design and testing are presented in Section 3.7.

Design of this storage facility in accordance with GDC 62 and RG 1.13 ensures a safe condition under normal and postulated accident conditions. The K_{eff} of the spent fuel storage racks is maintained less than or equal to 1.00, even if unborated water is used to fill the spent fuel storage pool, by both the designs of the fuel assemblies and the storage rack and the use of administrative procedures to control the placement of burned and fresh fuel and control rods.

Under accident conditions, the K_{eff} is maintained well below 0.95 assuming 2200 ppm borated water. The boron concentration of the water in the spent fuel pool is maintained at or above the minimum

NRC ORDER EA-12-051 (SFP LEVELS)

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REQUIREMENTS FOR RELIABLE SPENT FUEL POOL LEVEL INSTRUMENTATION AT OPERATING REACTOR SITES AND CONSTRUCTION PERMIT HOLDERS

All licensees identified in Attachment 1 to this Order shall have a reliable indication of the water level in associated spent fuel storage pools capable of supporting identification of the following pool water level conditions by trained personnel: (1) level that is adequate to support operation of the normal fuel pool cooling system, (2) level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck, and (3) level where fuel remains covered and actions to implement make-up water addition should no longer be deferred.

- 1. The spent fuel pool level instrumentation shall include the following design features:
 - 1.1 Instruments: The instrumentation shall consist of a permanent, fixed primary instrument channel and a backup instrument channel. The backup instrument channel may be fixed or portable. Portable instruments shall have capabilities that enhance the ability of trained personnel to monitor spent fuel pool water level under conditions that restrict direct personnel access to the pool, such as partial structural damage, high radiation levels, or heat and humidity from a boiling pool.
 - 1.2 Arrangement: The spent fuel pool level instrument channels shall be arranged in a manner that provides reasonable protection of the level indication function against missiles that may result from damage to the structure over the spent fuel pool. This protection may be provided by locating the primary instrument channel and fixed portions of the backup instrument channel, if applicable, to maintain instrument channel separation within the spent fuel pool area, and to utilize inherent shielding from missiles provided by existing recesses and corners in the spent fuel pool structure.
 - 1.3 Mounting: Installed instrument channel equipment within the spent fuel pool shall be mounted to retain its design configuration during and following the maximum seismic ground motion considered in the design of the spent fuel pool structure.
 - 1.4 Qualification: The primary and backup instrument channels shall be reliable at temperature, humidity, and radiation levels consistent with the spent fuel pool water at saturation conditions for an extended period. This reliability shall be established through use of an augmented quality assurance process (e.g., a process similar to that applied to the site fire protection program).
 - 1.5 Independence: The primary instrument channel shall be independent of the backup instrument channel.
 - 1.6 Power supplies: Permanently installed instrumentation channels shall each be powered by a separate power supply. Permanently installed and portable instrumentation channels shall provide for power connections from sources independent of the plant ac and dc power distribution systems, such as portable generators or replaceable batteries. Onsite generators used as an alternate power source and replaceable batteries used for instrument channel power shall have sufficient capacity to maintain the level indication function until offsite

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Attachment 2

resource availability is reasonably assured.

- 1.7 Accuracy: The instrument channels shall maintain their designed accuracy following a power interruption or change in power source without recalibration.
- 1.8 Testing: The instrument channel design shall provide for routine testing and calibration.
- 1.9 Display: Trained personnel shall be able to monitor the spent fuel pool water level from the control room, alternate shutdown panel, or other appropriate and accessible location. The display shall provide on-demand or continuous indication of spent fuel pool water level.
- . The spent fuel pool instrumentation shall be maintained available and reliable through appropriate development and implementation of the following programs:
 - 2.1 Training: Personnel shall be trained in the use and the provision of alternate power to the primary and backup instrument channels.
 - 2.2 Procedures: Procedures shall be established and maintained for the testing, calibration, and use of the primary and backup spent fuel pool instrument channels.
 - 2.3 Testing and Calibration: Processes shall be established and maintained for scheduling and implementing necessary testing and calibration of the primary and backup spent fuel pool level instrument channels to maintain the instrument channels at the design accuracy.

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NEI 12-02 (Revision 1) August 2012

The three critical levels that must be monitored in a spent fuel pool are discussed below. It should be noted that continuous indication from a single instrument over the entire span from level 1 to level 3 is not required but could be used. If more than one instrument is used to monitor the entire span, that set of instruments constitutes a single channel satisfying either the primary or backup instrument channel requirement (refer to Figure 1 below).

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A visual representation of monitoring levels 1, 2 and 3 and the associated requirements for monitoring between the points are presented in Figure 1. The minimum requirements apply to the separation distance between level indications and support development of appropriate response procedures. These requirements are separate from the instrument channel design accuracy discussed in section 3, which apply to either discrete or to continuous instruments.

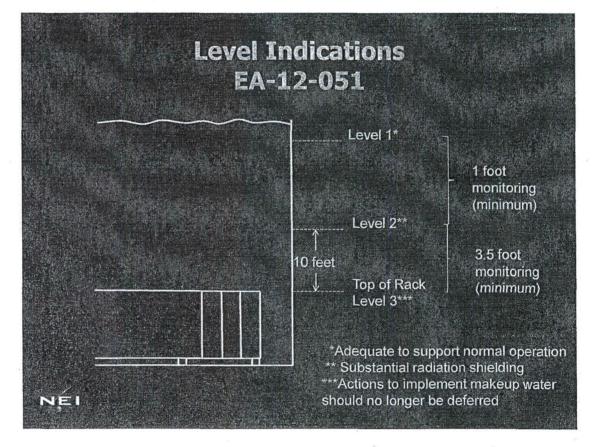


Figure 1

2.3.1. Level 1 – level that is adequate to support operation of the normal fuel pool cooling system

A typical fuel pool cooling system design includes a combination of weirs and/or vacuum breakers that prevent siphoning of the pool water level, below a minimum level, in the event of a piping rupture that can affect the SFP level. Level 1 represents the HIGHER of the following two points:

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- The level at which reliable suction loss occurs due to uncovering of the coolant inlet pipe, weir or vacuum breaker (depending on the design), or
- The level at which the water height, assuming saturated conditions, above the centerline of the cooling pump suction provides the required net positive suction head specified by the pump manufacturer or engineering analysis.

This level will vary from plant to plant and the instrument designer will need to consult plant-specific design information to determine the actual point that supports adequate cooling system performance.

2.3.2. Level 2 – level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck

Level 2 represents the range of water level where any necessary operations in the vicinity of the spent fuel pool can be completed without significant dose consequences from direct gamma radiation from the stored spent fuel. Level 2 is based on either of the following:

- 10 feet (+/- 1 foot) above the highest point of any fuel rack seated in the spent fuel pools, or
- a designated level that provides adequate radiation shielding to maintain personnel radiological dose levels within acceptable limits while performing local operations in the vicinity of the pool. This level shall be based on either plant-specific or appropriate generic shielding calculations, considering the emergency conditions that may apply at the time and the scope of necessary local operations, including installation of portable SFP instrument channel components. Additional guidance can be found in EPA-400 (Reference 4), USNRC Regulatory Guide 1.13 (Reference 5) and ANSI/ANS-57.2-1983 (Reference 6).

Designation of this level should not be interpreted to imply that actions to initiate water make-up must be delayed until SFP water levels have reached or are lower than this point.

2.3.3. Level 3 – level where fuel remains covered and actions to implement make-up water addition should no longer be deferred.

Level 3 corresponds nominally (i.e., +/- 1 foot) to the highest point of any fuel rack seated in the spent fuel pool. Level 3 is defined in this manner to provide the maximum range of information to operators, decision makers and emergency response personnel. Designation of this level should not be interpreted to imply that actions to initiate water make-up must or should be delayed until this level is reached.



South Texas Project Electric Generating Station RO, Box 289 Wadsworth, Texas 77483

February 28, 2013 NOC-AE-13002959 10 CFR 50.4 10 CFR 2.202

33650640

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

South Texas Project Units 1&2

Docket Nos. STN 50-498, STN 50-499

Overall Integrated Plan Regarding Commission Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)

References:

- Letter, Eric Leeds to E. D. Halpin, "Issuance of Order to Modify Licenses with Regard to Requirements for Reliable Spent Fuel Pool Instrumentation," March 12, 2012 (EA-12-051)
- NRC Interim Staff Guidance JLD-ISG-2012-03, Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation, Revision 0, August 29, 2012
- Letter D. W. Rencurrel to NRC, "Initial Status Report in Response to March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)", dated October 24, 2012

On March 12, 2012, the Nuclear Regulatory Commission (NRC) issued an order (Reference 1) to STP Nuclear Operating Company (STPNOC). Reference 1 directs STP Nuclear Operating Company to provide a reliable indication of the water level in associated spent fuel storage pools. Specific requirements are outlined in Attachment 2 of Reference 1.

Reference 1 required submission of an overall integrated plan, including how compliance will be achieved. The final interim staff guidance (Reference 2) was issued August 29, 2012 providing licensees an acceptable approach for complying with the order. The purpose of this letter is to provide the overall integrated plan, including a description of how compliance will be achieved pursuant to Section IV, Condition C.1.a, of Reference 1 in accordance with the guidance in Attachment 2 to Reference 1 and the guidance in Reference 2. See the Enclosure for STPNOC's response to the requested information.

There are no new commitments in this letter.

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If there are any questions regarding this letter, please contact Robyn Savage at (361) 972-7438.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: N

Dennis L. Koehl President and CEO/CNO

Enclosure:

South Texas Project (STP) Overall Integrated Plan for Implementation of Unit 1 & Unit 2 Spent Fuel Pool Level Instrumentation to Meet NRC Order EA-12-051

NOC-AE-13002959 Page 3 of 3

cc: (paper copy)

Regional Administrator, Region IV U. S. Nuclear Regulatory Commission 1600 East Lamar Boulevard Arlington, TX 76011-4511

Balwant K. Singal Senior Project Manager U.S. Nuclear Regulatory Commission One White Flint North (MS 8 B1) 11555 Rockville Pike Rockville, MD 20852

NRC Resident Inspector U. S. Nuclear Regulatory Commission P. O. Box 289, Mail Code: MN116 Wadsworth, TX 77483

C. M. Canady City of Austin Electric Utility Department 721 Barton Springs Road Austin, TX 78704

U. S. Nuclear Regulatory Commission Director of Office of Nuclear Regulation One White Flint North (MS 13 H 16M) 11555 Rockville Pike Rockville, MD 20852-2738

(electronic copy)

A. H. Gutterman, Esquire Morgan, Lewis & Bockius LLP

Balwant K. Singal U. S. Nuclear Regulatory Commission

John Ragan Chris O'Hara Jim von Suskil NRG South Texas LP

Kevin Pollo Richard Pena City Public Service

Peter Nemeth Crain Caton & James, P.C.

C. Mele City of Austin

Richard A. Ratliff Texas Department of State Health Services

Alice Rogers Texas Department of State Health Services

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ENCLOSURE NOC-AE-13002959

South Texas Project (STP) Overall Integrated Plan for Implementation of Unit 1 & Unit 2 Spent Fuel Pool Level Instrumentation

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to Meet NRC Order EA-12-051

Revision: 00

1.0 OVERALL INTEGRATED PLAN INTRODUCTION

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This document provides the overall Integrated Plan (the "Plan") which the STP Nuclear Operating Company ("STPNOC") will implement for Units 1 and 2 to comply with the requirements of NRC Order EA-12-051, "Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation" (Ref.2), (the "ORDER"), NRC Interim Staff Guidance JLD-ISG-2012-003 [Rev.0] (Ref.3), (the "ISG"), and NEI Report 12-02 [Rev.1] ("NEI 12-02").

This Plan follows the format and provides all of the information on the STP 1 & 2 Integrated Plan that is required in NEI 12-02 [Rev.1] (Ref.1), Section A-2-2. Throughout this Plan, any reference to NEI 12-02 and the ISG will be based on the revisions above. Any reference to NEI 12-02 will include compliance to the clarifications and exceptions to NEI 12-02 required by the Interim Staff Guidance, Rev. 0.

In response to the NRC requirements, STPNOC will provide two channels of independent, permanently-installed, wide-range spent fuel pool level instrumentation ("SFPLI"), for the spent fuel pool ("SFP") of each unit. The spent fuel pool for each unit is independent and not interconnected in any way. For each SFP, the instrumentation provided for each channel will utilize the same technology, as permitted by the NEI 12-02 [Rev.1]. The spent fuel pool level instrumentation for each SFP on both the Primary and Backup Channels.

Both the Primary and Backup Channel/Instrument location and display of the SFP level will be independently mounted in each unit's Radwaste Control Room in the Mechanical Electrical Auxiliary Building (MEAB), which is an accessible post-event location. Other locations are still being considered.

Both the Primary and Backup Channel remote, non-safety related indication of the SFP level will also be provided in each unit's Control Room via input to the Plant Computer.

The instrumentation systems will not be safety-related, but will meet the requirements for augmented quality in accordance with NEI 12-02 [Rev.1] and the ISG as described below.

Since all of the potential suppliers have not completed development, the information in this Plan is based on the overall strategy and on information which, based on current information from potential suppliers, is thought to envelope the systems being developed for this application.

If there are any changes to the requirements in NRC JLD-ISG-2012-003 [Rev.0] and NEI 12-02 [Rev.1], relief from the requirements and schedule documented in this Plan may be required, in accordance with Section 12.0. Any required changes to this Plan will be defined in the periodic status reports submitted to the NRC.

2.0 APPLICABILITY:

This Plan applies to the spent fuel pools for South Texas Project Unit 1 and Unit 2.

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3.0 SCHEDULE:

The installation of reliable spent fuel pool level instrumentation for the spent fuel pool associated with Unit 1 is scheduled for completion prior to 10/28/2015, which is the end of the second refueling outage (1RE19) following submittal of this Plan.

The installation of reliable spent fuel pool level instrumentation for the spent fuel pool associated with Unit 2 is scheduled for completion prior to 4/29/2015, which is the end of the second refueling outage (2RE17) following submittal of this Plan.

Unit 1 Milestones are as follows:

- Design/Engineering September of 2014
- Purchase of instruments & equipment February of 2015
- Receipt of equipment June of 2015
- Unit 1 Installation & Functional Testing October of 2015

Unit 2 Milestones are as follows:

- Design/Engineering December of 2013
- Purchase of instruments & equipment August of 2014
- Receipt of equipment November of 2014
- Installation & Functional Testing April of 2015

Consistent with the requirements of the ORDER and the guidance from NEI 12-02 [Rev.1], status reports will be generated in six (6) month intervals from the submittal of this Plan.

4.0 IDENTIFICATION OF SPENT FUEL POOL WATER LEVELS:

The STP Unit 1 and 2 spent fuel pools are essentially identical. The following SFP elevations are identified:

- The bottom of the pool is at Plant El. 21 ft. 11 in.
- The top of the SFP racks is approximately at Plant El. 39 ft. 10 in.
- The minimum Limiting Condition for Operation SFP level is Plant El. 62 ft.
- Normal SFP water level is at Plant El. 66 ft. 6 in.
- Non-safety related level switch alarms are activated at Plant El. 67 ft. on high level and Plant El. 66 ft. on low level.
- The spent fuel pool deck is at Plant El. 68 ft.

The required key SFP water levels, per guidance of NEI 12-02 [Rev.1] and ISG (with clarifications and exceptions), are as follows:

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4.1 LEVEL 1: Level adequate to support operation of the normal fuel pool cooling system.

LEVEL 1 represents the HIGHER of either the level at which reliable suction loss to the cooling pump occurs, or, the required NPSH (Nominal Pump Suction Head) of the cooling pump.

Loss of reliable suction to SFP cooling pumps. For the purposes of this Plan, this level will conservatively be placed at Plant El. 64 ft. 2 in. This allows for just over 1 ft. of SFP water level above the top of the suction inlet flange (SFP Cooling Pump 14 in. suction line with centerline of suction inlet flange at Plant El. 62 ft. 6 in.) which will be sufficient for NPSH. (Ref. 9)

Therefore, considering the top of SFP fuel storage rack is at Plant El. 39 ft. 10 in., the indicated level on either the Primary or Backup Instrument Channel of greater than 24 ft. 4 in. above the top of the SFP fuel storage racks based on the design accuracy for the instrument channel per NEI 12-02 [Rev.1], for both the Primary and Backup Instrument Channels, is adequate for normal SFP cooling system operation.

LEVEL 1 = Plant El. 64 ft. 2 in or 24 ft. 4 in. water level above the top of the SFP fuel storage rack

4.2 LEVEL 2: Level adequate to provide substantial radiation shielding for a person standing on the SFP operating deck.

Indicated level on either the Primary or Backup Instrument Channel of greater than 10 ft, above the top of SFP stored fuel assemblies based on current guidance in NRC RG 1.13 [Rev.2] (Ref. 4) will achieve substantial radiation shielding. Requirements on substantial SFP radiation shielding is also given in ANSI/ANS-57.2-1983 (Ref. 5), and states that radiation shall not exceed 2.5 mRem/hr, but the minimum water depth to achieve this is undefined. NRC RG 1.13 [Rev.2] took exception to using dose rates as design input for minimum SFP water level, and instead defined the minimum level as 10 ft. above the stored fuel assemblies.

STPNOC elects to use the conservative approach of defining the top of the fuel rack as a basis for measurement. Therefore, indicated level on either the Primary or Backup Instrument Channel of greater than 10 ft. above the top of the SFP fuel storage rack, based on the design accuracy of the instrument channel per NEI 12-02 [Rev.1], for both the Primary and Backup Instrument Channels, ensures there is adequate water level to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck.

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LEVEL 2 = Plant El 49 ft. 10 in. or 10 ft. water level above the top of the SFP fuel storage rack.

4.3 LEVEL 3: Level where the fuel remains covered.

As stated above, STPNOC elects to use the conservative approach of defining the top of the fuel rack as a basis for measurement. The installation of the SFPLI sensor will be such that it will measure as close as possible to the top of the SFP fuel rack. Indicated level on either the Primary or Backup Instrument Channel of greater than $\frac{1}{2}$ ft. above the top of SFP fuel storage racks based upon the design accuracy of the instrument channel per NEI 12-02 [Rev.1], for both the Primary and Backup Instrument Channels, satisfies the NEI 12-02 [Rev.2] requirement of \pm 1 ft. from the top of the fuel rack. This monitoring level ensures there is adequate water level above the stored fuel seated in the SFP fuel storage rack.

LEVEL 3 = Plant El 40 ft. 4 in. or 6 in. water level above the top of the SFP fuel storage rack.

5.0 INSTRUMENTS:

Both the Primary and Backup Instrument Channels will utilize permanently-installed instruments. The design of the primary and backup instruments will be consistent with the requirements by NEI 12-02 [Rev.1], the ISG, and this Plan.

The current plan is for both channels to utilize Guided Wave Radar, which functions according to the principle of Time Domain Reflectometry (TDR). A generated pulse of electromagnetic energy travels down the probe. Upon reaching the liquid surface the pulse is reflected and based upon reflection times level is inferred. The measured range will be continuous from the high pool level elevation (67') to the top of the spent fuel racks (Ref. 8). However, STP is still evaluating other designs for this application. Any changes to the current plan will be reported in the 6 month update letter.

The supplier for the SFP instrumentation will be chosen by a competitive bidding process completed after submittal of this Plan, so the information in this Plan is based on the overall strategy and on available information from potential supplier's information on systems being developed for this application.

5.1 Primary (fixed) Instrument Channel

The Primary Instrument Channel level sensing components will be located in the northeast corner of the Spent Fuel Pool, as shown in Attachment 1. The primary instrument channel will provide continuous level indication over a range from Plant El. 40 ft. 4 in. (LEVEL 3) to Plant El. 67 ft. (SFP high level alarm) or a range of 26 ft. 8 in. In addition, the capability for discrete level indications at LEVEL1, LEVEL 2 and LEVEL 3, as described in Section 4.0, will be available.



South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

June 25, 2013 NOC-AE-13003008 File No.: G25 10 CFR 2.202

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

South Texas Project Unit 1 & 2 Docket Nos. STN 50-498, STN 50-499 Response to Request for Additional Information Regarding the Overall Integrated Plan in Response to Order EA-12-051, <u>"Reliable.Spent Fuel Pool Instrumentation" (TAC Nos. MF0827 and MF0828)</u>

References:

- 1. Letter, Eric Leeds to E. D. Halpin, "Issuance of Order to Modify Licenses with Regard to Requirements for Reliable Spent Fuel Pool Instrumentation," March 12, 2012 (EA-12-051) (ST-AE-NOC-12002271) (ML12054A679)
- 2. Letter, D. L. Koehl to NRC Document Control Desk, "Overall Integrated Plan Regarding Commission Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)," dated February 28, 2013 (NOC-AE-13002959) (ML13070A006)
- NRC letter dated June 7, 2013, "South Texas Project, Units 1 and 2 Request for Additional Information RE: Overall Integrated Plan in Response to Order EA-12-051, "Reliable Spent Fuel Pool Instrumentation" (TAC Nos. MF0827 and MF0828) (ST-AE-NOC-13002439) (ML131149A09)

On March 12, 2012, the Nuclear Regulatory Commission (NRC) issued an Order (Reference 1) modifying licenses with regard to requirements for reliable spent fuel pool instrumentation. On February 28, 2013, STP Nuclear Operating Company (STPNOC) submitted an Overall Integrated Plan (OIP) (Reference 2) in response to the NRC Order. By a letter (Reference 3) dated June 7, 2013, the NRC staff determined that additional information is needed to complete their review of the OIP. The STPNOC response to Reference 3 is provided in the attachment to this letter.

There are no regulatory commitments in this letter.

STI: 33704694

If there are any questions, please contact Ken Taplett at 361-972-8416.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: _______ June 25, 2013

kjt

G J. Powell

G. T. Powell Site Vice President

Attachment: Response to Request for Additional Information Regarding Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)

NOC-AE-13003008 Page 3

cc: (paper copy)

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Regional Administrator, Region IV U. S. Nuclear Regulatory Commission 1600 East Lamar Boulevard Arlington, TX 76011-4511

Balwant K. Singal Senior Project Manager U.S. Nuclear Regulatory Commission One White Flint North (MS 8 B1) 11555 Rockville Pike Rockville, MD 20852

NRC Resident Inspector U. S. Nuclear Regulatory Commission P. O. Box 289, Mail Code: MN116 Wadsworth, TX 77483

C. M. Canady City of Austin Electric Utility Department 721 Barton Springs Road Austin, TX 78704

U.S. Nuclear Regulatory Commission Director, Office of Nuclear Reactor Regulation One White Flint North (MS 13 H 16M) 11555 Rockville Pike Rockville, MD 20852-2738 (electronic copy)

A. H. Gutterman, Esquire Morgan, Lewis & Bockius LLP

Balwant K. Singal U. S. Nuclear Regulatory Commission

John Ragan Chris O'Hara Jim von Suskil NRG South Texas LP

Kevin Pollo Richard Peña City Public Service

Peter Nemeth Crain Caton & James, P.C.

C. Mele City of Austin

Richard A. Ratliff Texas Department of State Health Services

Robert Free Texas Department of State Health Services

Response to Request for Additional Information Regarding Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)

References:

- Letter, D. L. Koehl to NRC Document Control Desk, "Overall Integrated Plan Regarding Commission Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)," dated February 28, 2013 (NOC-AE-13002959) (ML13070A006)
- Letter, Eric Leeds to E. D. Halpin, "Issuance of Order to Modify Licenses with Regard to Requirements for Reliable Spent Fuel Pool Instrumentation," March 12, 2012 (EA-12-051) (ST-AE-NOC-12002271) (ML12054A679)
- 3. NRC Japan Lessons-Learned Project Directorate Interim Staff Guidance JLD-ISG-2012-03, "Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation, Revision 0, August 29, 2012 (ML12221A339)
- NEI 12-02, Industry Guidance for Compliance with NRC Order EA-12-051, "To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," Revision 1, dated August 2012 (ML122400399)

Reference 1 provided the Overall Integrated Plan (OIP) which the STP Nuclear Operating Company ("STPNOC") will implement for Units 1 and 2 to comply with the requirements of NRC Order EA-12-051, "Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation" (Reference 2), NRC Interim Staff Guidance JLD-ISG-2012-003, Revision 0, (Reference 3) and NEI Report 12-02, Revision 1 (Reference 4).

As discussed in Reference 1, any changes to the requirements in NRC JLD-ISG-2012-003 or NEI 12-02 may require relief from the requirements and schedule documented in the OIP.

As provided in the OIP, the Milestones for completing the design and engineering work for Unit 1 are September 2014 and for Unit 2 is December 2013.

The following responses to the request for additional information are based on information developed to date. Any changes to the following information that occur after completing and approving the final design for reliable spent fuel pool instrumentation will be provided in the periodic 6-month status reports submitted to the NRC required by Order EA-12-051.

Attachment NOC-AE-13003008 Page 2 of 23

REQUEST FOR ADDITIONAL INFORMATION

OVERALL INTEGRATED PLAN IN RESPONSE TO

ORDER EA-12-051, "RELIABLE SPENT FUEL POOL INSTRUMENTATION"

STP NUCLEAR OPERATING COMPANY

SOUTH TEXAS PROJECT, UNITS 1 AND 2

DOCKET NOS. 50-498 AND 50-499

1.0 <u>Introduction</u>

By letter dated February 28, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13070A006), STP Nuclear Operating Company (STPNOC, the licensee), submitted an Overall Integrated Plan (OIP) in response to the March 12, 2012, U.S. Nuclear Regulatory Commission (NRC), Commission Order modifying licenses with regard to requirements for Reliable Spent Fuel Pool (SFP) Instrumentation (Order Number EA-12-051; ADAMS Accession No. ML12054A679) for South Texas Project (STP), Units 1 and 2. The NRC staff endorsed Nuclear Energy Institute (NEI) 12-02, "Industry Guidance for Compliance with NRC Order EA-12-051, to Modify Licenses with Regard to Reliable SFP Instrumentation," Revision 1, dated August 2012 (ADAMS Accession No. ML12240A307), with exceptions as documented in Interim Staff Guidance (ISG) 2012-03, "Compliance with Order EA-12-051, Reliable SFP Instrumentation," Revision 0, dated August 29, 2012 (ADAMS Accession No. ML12221A339).

The NRC staff has reviewed the February 28, 2013, response by the licensee and determined that the following request for additional information (RAI) is needed to complete its technical review. Please provide the response to the following RAIs.

Attachment NOC-AE-13003008 Page 3 of 23

2.0 Levels of Required Monitoring

The OIP states, in part, that

LEVEL 1: Level adequate to support operation of the normal fuel pool cooling system.

Plant El. 64 ft. 2 in or 24 ft. 4 in. water level above the top of the SFP fuel storage rack.

LEVEL 2: Level adequate to provide substantial radiation shielding for a person standing on the SFP operating deck.

Plant El 49 ft. 10 in. or 10 ft. water level above the top of the SFP fuel storage rack.

LEVEL 3: Level where the fuel remains covered.

Plant El 40 ft. 4 in, or 6 in. water level above the top of the SFP fuel storage rack.

...The installation of the SFPLI [spent fuel pool level instrumentation] sensor will be such that it will measure as close as possible to the top of the SFP fuel rack. Indicated level on either the Primary or Backup Instrument Channel of greater than $\frac{1}{2}$ ft. above the top of SFP fuel storage racks based upon the design accuracy of the instrument channel per NEI 12-02 [Rev.1], for both the Primary and Backup Instrument Channels, satisfies the NEI 12-02 [Rev.2] requirement of ±1 ft. from the top of the fuel rack. This monitoring level ensures there is adequate water level above the stored fuel seated in the SFP fuel storage rack.

NRC RAI-1a

Please provide the following:

a) For Level 1, please specify how the identified location represents the HIGHER of the two points described in the NEI 12-02 guidance for this level.

STPNOC Response

LEVEL 1 represents the HIGHER of either the level at which reliable suction loss to the spent fuel pool (SFP) cooling pump occurs, or the required net positive suction head (NPSH) of the SFP cooling pump

Required NPSH.

The SFP cooling pumps were analyzed for the conservative worst case operation of the SFP cooling pumps. Maximum values for line resistance, fluid temperature, suction flow

Attachment NOC-AE-13003008 Page 4 of 23

and static head were used to calculate NPSH parameters for both required and available NPSH (NPSH_R and NPSH_A). It was determined that for the worst case scenario, the NPSH_A was significantly higher than NPSH_R. The NPSH_A was calculated to be 42.67 feet (ft) and NPSH_R was calculated to be 18.75 ft.

Therefore, NPSH_R is not the determining value to be used for LEVEL 1.

Loss of reliable suction to SFP cooling pumps.

For the purposes of the OIP, this level is conservatively placed at Plant elevation (El.) 64 ft, 2 inches (in). This level provides for more than one foot of water above the top of the SFP cooling pump suction inlet flange (the centerline of the 14 inch suction line flange to the pump is at Plant El. 62 ft. 6 in.) which will be sufficient for NPSH.

A vortex calculation shows 0.134% air entrainment at an elevation one foot above the suction pipe centerline. Level 1 at 64 ft. 2 in. is adequate for normal SFP cooling system operation. Therefore, Level 1 represents the HIGHER of the two points described in the NEI 12-02 guidance.

NRC RAI-1b

b) A clearly labeled sketch depicting the elevation view of the proposed typical mounting arrangement for the portions of instrument channel consisting of permanent measurement channel equipment (e.g., fixed level sensors and/or stilling wells, and mounting brackets). Please indicate on this sketch the datum values representing Level 1, Level 2, and Level 3 as well as the top of the fuel. Indicate on this sketch the portion of the level sensor measurement range that is sensitive to measurement of the fuel pool level, with respect to the Level 1, Level 2, and Level 3 datum points.

STPNOC Response

See Figures 1 and 2 of this Attachment.

3.0 Instrumentation and Design Features

3.1 Instruments and Arrangement

The OIP states, in part, that

Both the Primary and Backup Instrument Channels will utilize permanentlyinstalled instruments....

The Primary Instrument Channel level sensing components will be located in the northeast corner of the Spent Fuel Pool, as shown in Attachment 1....

The Backup Instrument Channel level sensing components will be located in the northwest corner of the Spent Fuel Pool, as shown in Attachment 1....

The current Plan is to mount the supporting electronic instruments outside of the spent fuel pool area, to provide a more benign radiation and environmental conditions, and also provide for reasonable and accessible locations for operators.

SFP Primary and Backup Channel Level Instruments are currently planned to be located in Radwaste Control Room of the Mechanical Auxiliary Building (MAB); however, STPNOC is still evaluating other possible locations (i.e. relay room).

NRC RAI-2

Please provide a clearly labeled sketch or marked-up plant drawing of the plan view of the SFP area, depicting the SFP inside dimensions, the planned locations/ placement of the primary and back-up SFP level sensor, and the proposed routing of the cables that will extend from the sensors toward the location of the read-out/display device.

STPNOC Response

See Figure 3 of this Attachment.

3.2 Mounting

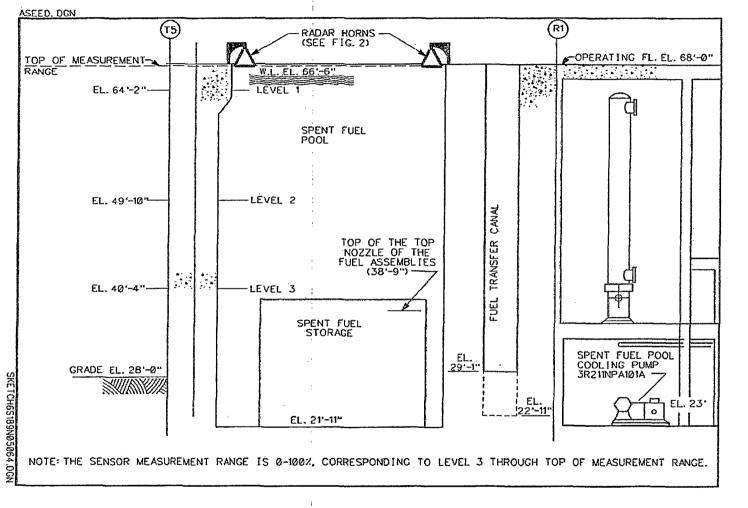
The OIP states, in part, that

Consideration will be given to the maximum seismic ground motion that occurs at the installation location for the permanently installed equipment which is documented in the UFSAR [Updated Final Safety Analysis Report] Section 3.7. The mountings shall be designed consistent with the highest safety or seismic classification of the SFP. The level sensors will be mounted on seismically gualified brackets.

NRC RAI-3a

Please provide the following:

a) The design criteria that will be used to estimate the total loading on the mounting device(s), including static weight loads and dynamic loads. Please describe the methodology that will be used to estimate the total loading, inclusive of design basis maximum seismic loads and the hydrodynamic loads that could result from pool sloshing or other effects that could accompany such seismic forces.



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Figure 1 Elevation View Orientation of Monitored Levels

> Attachment NOC-AE-13003008 Page 20 of 23

Attachment NOC-AE-13003008 Page 21 of 23

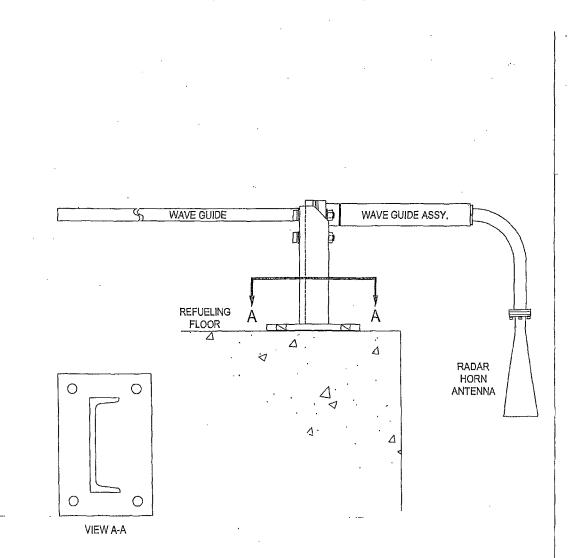


Figure 2 Proposed Mounting Arrangement

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7.1	Unusual Event – RU19
7.0	STAMPEDE CALCULATIONS
6.0	ASSUMPTIONS
5.0	REFERENCES
4.0	INPUTS
3.0	METHOD OF ANALYSIS
2.0	SUMMARY OF RESULTS
1.0	OBJECTIVE/SCOPE



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1.0 OBJECTIVE/SCOPE

The purpose of this calculation is to determine the Emergency Action Level (EAL) threshold values of a radiological release from the Unit Vent or Main Steam Lines for an Unusual Event, Alert, Site Area Emergency, or General Emergency. The calculated threshold values are to be included in the STP EAL Technical Basis document, which implements the new NEI 99-01, Revision 6, Emergency Action Level Scheme and will be submitted to the NRC for approval. Upon NRC approval, the values will be used in 0ERP01-ZV-IN01, Revision 10, Emergency Classification.

Both a hand calculation and the South Texas Assessment Model Projecting Emergency Dose Evaluation (STAMPEDE) software program were used to generate the results. The hand calculation is included as Attachment 1.

Revision 1 of this calculation incorporated decay for a release taking place one hour after reactor shutdown. This was done to create continuity between the two methodologies present.

2.0 SUMMARY OF RESULTS

The results of the calculations for the radiation monitors specified in the STP EAL Basis Document and are listed in Table 2.1, below.

	Table 2.1: Summary o	f Calculation Results	· · · ·
Emergency Action Level		RT-8010B, Unit Vent (μCi/sec)	RT-8046 through 8049, Main Steam Lines (μCi/cc)
RU1	Unusual Event		
	Easternation Sector	40E#05	5:00E:027:57
	STAMPEDE	N/A	NZAL
RA1	Alert		· · ·
	- Bland Calentations	457E+06	4105+007
	SIAMREDE	250E+06	4:50B+00
RS1	Site Area Emergency		
	Hand Calculation		44105E+01
	STAMPEDE	2:50E±07	A150H#01
RG1	General Emergency		
	Hand Galeulation 3	1 57E+08	410H402
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*STAMPEDE was not used to determine the threshold for RU1. Reference 5.10 indicates that the ODCM methodology should be used to determine the threshold value.

This calculation will be used to establish the threshold values for abnormal radiation based emergencies in the STP EAL Technical Basis document.



3.0 METHOD OF ANALYSIS

Previously, STAMPEDE was used to calculate the Emergency Action Level threshold values for effluent releases. A hand calculation will verify the STAMPEDE calculations. The hand calculation is described in Attachment 1 of this document STAMPEDE conforms to the requirements of STP Procedure 0PGP07-ZA-0014, Software Quality Assurance Program. STAMPEDE was run at STP on an STP computer and under the supervision of an ENERCON employee with access to the STP site as a critical worker.

4.0 INPUTS

- 4.1 Per NEI 99-01, Revision 6, Initiating condition AU1, EAL 1, the Notice of Unusual Event initiating condition is a release of gaseous or liquid radioactivity greater than two times the ODCM limit for sixty minutes or longer (Reference 5.10).
- 4.2 The ODCM offsite dose limit is exceeded if the Xe-133 release concentration exceeds 7.41E-04 μCi/cc (Reference 5.6).
- 4.3 The Unit Vent flow rate is 9.4E+07 cc/sec (Reference 5.1).
- 4.4 The main steam line pressure and PORV choke flow rate are 1285 psig and 1.05E+06 lbm/hr, respectively (Reference 5.2).
- 4.5 ---- The specific volume of saturated steam at 1285 psig is 0.338 ft³/lbm (Reference 5.3).
- 4.6 The release concentration is varied to find the release concentration which correlates to each emergency action level. Emergency action levels are taken from NEI 99-01, Revision 6 (Reference 5.10) for initiating conditions AA1, AS1 and AG1. EAL 1 is the EAL of interest in each initiating condition. The doses at the Site Boundary that correlate to the threshold concentrations are listed in Table 4.1.

Table 4.1 EAL Offsite Dose Initiating Conditions

	Alert	Site Area	General
TEDE	10 miem	isl00 mrem	1000 inrem 🔬 🛼
Thyroid CDE	50 mrem	500 mrem	5000 mrem

5.0 REFERENCES

- 5.1 Offsite Dose Calculation Manual, Revision 17, March 2011
- 5.2 Main Steam PORV Capacity Verification MC05591, Revision: 1
- 5.3 NIST Steam Tables, 2011
- 5.4 0ERP01-ZV-IN01, Emergency Classification Draft Revision 10
- 5.5 0ERP01-ZV-TP01, Offsite Dose Calculations, Revision 21
- 5.6 STP Calculation NC-9012, CRMS Rad Monitor Setpoints, Revision 7
- 5.7 STP Calculation NC-9011, Revision 2
- 5.8 STAMPEDE Computer Program, Revision 7.0.3.3
- 5.9 STAMPEDE User's Manual
- 5.10 NEI 99-01, Revision 6, Development of Emergency Action Levels for Non-Passive Reactors
- 5.11 0PGP07-ZA-0014 Quality Assurance Program
- 5.12 ITWMS Call Number 1000010987 Design Document, Revision 0



Radiological Release Thresholds for Emergency Action Levels

CALC. NO. STPNOC013-CALC-002 REV. 1 PAGE NO. 8 of 49

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6.0 ASSUMPTIONS

6.1 Unit Vent Noble Gas Monitor

To be consistent with the ODCM methodology, the unit vent release is assumed to be entirely Xe-133. The unit vent noble gas monitor is calibrated to Xe-133 (Reference 5.1) therefore, the monitor reading accurately reflects the Xe-133 release magnitude.

To be consistent with ODCM methodology, the main steam line release is assumed to be entirely Xe-133. The noble gas monitor is calibrated to Xe-133 (Reference 5.6).

6.2 Release Duration

Per Reference 5.10, Sections IC AA1, AS1, and AG1 developer notes, the release should be assumed to last one hour.

6.3 Release following Reactor Shutdown

The release initiates one hour after reactor shutdown. While a release initiating at reactor shutdown is likely, significant decay of short lived nuclides occurs during the migration time. A release at reactor shutdown would have a significantly higher activity at the monitor location than at the reception site. It is important for the threshold to not be calculated at shutdown as this ------would create a very high threshold which would not be appropriate for releases which occur shortly after shutdown. One hour after reactor shutdown is sufficient time to decay short lived nuclides and create a conservative threshold.

6.4 Source Term

Per Reference 5.1, any unit vent release with increased RCS activity and no core melt should be calculated using the gap inventory. Therefore, the gap inventory is used for all unit vent releases. Per Reference 5.1, for a main steam line release following a steam generator tube rupture it is appropriate to use an inventory of noble gases plus 0.2% iodine. A steam generator tube rupture is the only scenario which would create significant offsite doses through a main steam line release.

6.5 Default STAMPEDE Input Values

Reference 5.10 developer notes for initiating conditions AA1, AS1 and AG1 suggest using the ODCM or the site's emergency dose assessment methodology. STAMPEDE is used for emergency dose assessment. Per Reference 5.1, when actual meteorology is not available, the default STAMPEDE values should be used. Had the ODCM methodology been used, the 500 hour peak χ/Q value would be used which is less conservative than the χ/Q value produced by STAMPEDE using default meteorological conditions. Therefore, the use of STAMPEDE default values provides a more conservative estimate than that of the alternative method outlined in Reference 5.10.

6.6 Average Effluent Concentration (χ/Q)

The same χ/Q is used for the unit vent and main steam line release. Reference 5.1 applies the same unit vent χ/Q to Units 1 and 2 which would also be applicable to the main steam line. All releases are considered to be ground level releases.



7.0 STAMPEDE CALCULATIONS

- 7.1 Unusual Event RU1
 - 7.1.1 Unit Vent Monitor

AU1 recommends declaring an unusual event due to a release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer (Reference 5.10).

STP sets the ODCM limit at 7.41E-04 μ Ci/cc (Reference 5.6, pg. 16). Two times the limit would be 1.48E-03 μ Ci/cc. The threshold is listed in μ Ci/sec so that variations in flow rate do not change the threshold. The normal flow rate from the unit vent is 9.4E+07 cc/sec (Reference 5.1).

$$Concentration\left(\frac{\mu Ci}{cc}\right) * Flow Rate\left(\frac{cc}{sec}\right) = Release Rate\left(\frac{\mu Ci}{sec}\right)$$
$$(1.48E - 03)\left(\frac{\mu Ci}{cc}\right) * (9.4E + 07)\left(\frac{cc}{sec}\right) = 1.4E + 05\left(\frac{\mu Ci}{sec}\right)$$
$$Equation 7.1.1.1$$

7.1.2 Main Steam Line Monitor

The ODCM does not calculate a release corresponding to allowable limits for the main steam line monitors. Since the unit vent release calculated in the ODCM was assumed to be primarily Xe-133, the assumption is made in the ODCM that other noble gases and iodine may be ignored in the calculation. This assumption is equally justifiable for the main steam line and the same limiting release will be used.

The magnitude of the release calculated for the unit vent Unusual Event applies to the main steam lines as well. The main steam line PORV's will create a dose exceeding two times the ODCM limit by releasing 1.4E+05 μ Ci/sec of activity which is equivalent to the release from the unit vent.

The steam lines hold saturated steam at 1285 psig, per Reference 5.2, which has a specific volume of 0.338 ft³/lbm (Reference 5.3). The PORVs will release the steam at 1.05E+06 lbm/hr per Reference 5.2. This creates a set flow rate of steam from the main steam lines of 2.79E+06 cc/sec as shown below.

$$F\left(\frac{lbm}{hr}\right) * Density\left(\frac{ft^3}{lbm}\right) * 28316.846\left(\frac{cc}{ft^3}\right) \div 3600\left(\frac{sec}{hr}\right) = \frac{cc}{sec}$$

$$1.05E + 06\left(\frac{lbm}{hr}\right) * 0.338\left(\frac{ft^3}{lbm}\right) * 28316.846\left(\frac{cc}{ft^3}\right) \div 3600\left(\frac{sec}{hr}\right) = 2.79E + 06\frac{cc}{sec}$$
Equation 7.1.2.1

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Since the flow rate is set, the concentration will determine the limit. Equation 7.1.1.1 solves for the limiting concentration of $5.00E-02 \ \mu Ci/cc$ as shown below.

$$\frac{\text{Limiting Release}\left(\frac{\mu\text{Ci}}{\text{sec}}\right)}{\text{Release Rate}\left(\frac{cc}{sec}\right)} = \text{Limiting Concentration}\left(\frac{\mu\text{Ci}}{cc}\right)$$
$$\frac{1.40 * 10^{5}\left(\frac{\mu\text{Ci}}{sec}\right)}{2.79 * 10^{6}\left(\frac{cc}{sec}\right)} = 5.00E - 02\left(\frac{\mu\text{Ci}}{cc}\right)$$

iation 7.1.2.2

Alert, Site Area and General Emergencies – RA1, RS1, RG1 7.2

7.2.1 Unit Vent Monitor

Input

1.1.1

The Alert BAL is set to 10 mrem TEDE and 50 mrem Thyroid CDE per Reference 5.10. The emergency offsite dose calculation software STAMPEDE was used to calculate the release which corresponds to this dose. A release concentration correlating to the EAL threshold value was calculated by varying the input. The following assumptions and inputs were used for the calculation as described in Sections 4.0 and 6.0.

Release begins at reactor trip

Release lasts for one hour •

Gap inventory source term

Default STAMPEDE input values

Windspeed = 13.2 mph0

Stability class \mathbf{D}_{ch} , where the transformation of the state of the stat ...0

Results

Given a monitored unit vent release of 2.50E+06 µCi/sec, the Thyroid CDE is 51 mrem/hr at the closest portion of the site boundary and the EAL Initiating Condition is exceeded.

Threshold values for the Site Area Emergency and General Emergencies are multiples of 10 and 100 of the Alert. Since the correlation between release concentration and dose is linear, threshold values for the steam line monitors are 2.50E+07 and 2.50E+08 µCi/sec for the SAE and GE respectively. Both are also limited by Thyroid CDE. Additional STAMPEDE iterations were performed to confirm this and are attached.

The input and output files can be found at the end of this document in Attachment 3.

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7.2.2 Main Steam Line Monitor

Input

A release concentration correlating to the EAL threshold value was calculated by varying the input. The following assumptions and inputs were used for this calculation as described in Sections 4.0 and 6.0.

- Release begins at reactor trip
- Release lasts for one hour
- Noble gas + iodine with 0.2% iodine source term
- Default STAMPEDE input values
 - \circ Windspeed = 13.2 mph
 - o Stability class D

Results

Given a monitored main steam line release of 4.5 μ Ci/cc, the Thyroid CDE is 50 mrem/hr and the BAL Initiating Condition is exceeded.

- The-input and-output-files-can be found at the end of this document in Attachment 3. -

7.3 Threshold values for the Site Area Emergency and General Emergencies are multiples of 10 and 100 of the Alert. Since the correlation between release concentration and dose is linear, threshold values for the steam line monitors are 45 and 450 μ Ci/cc for the SAE and GE respectively. Both are also limited by Thyroid CDE. Additional STAMPEDE iterations were performed to confirm this and are attached.



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Attachment 1 – Hand Calculations

1.0 OBJECTIVE/SCOPE

Each release calculated using STAMPEDE in the main document is calculated by hand in this attachment and the results compared to STAMPEDE.

2.0 SUMMARY OF RESULTS

Table 2.1 is displayed again below showing the results from all the calculations. The minor difference is due to STAMPEDE using decay factors over a one hour period after shutdown. This also accounts for the change in the limiting dose being TEDE in the hand calculations and Thyroid CDE in the STAMPEDE calculations. The accuracy of the hand calculation is considered sufficient and recommended for use in Emergency Action Levels.

Table 2.1 Results

Emergency Action	가운 가슴 가슴 가슴 가슴 가슴 가슴. 가슴 가슴 가슴 가슴 가슴 가슴 가슴 가슴. 전 가슴 가슴 가슴 가슴 가슴 가슴 가슴 가슴.	RT-8010b, Unit Vent	RT-8046 through 8049,
Level		(µCi/sec)	—– Main Steam-Line-—– (μCi/cc)
RU1	Unusual Event		
an a	Hand Calculation	1/210E+05 -/N//S	
RA1	Alert		
	ElantRCalculations	E 57/E#06 2150/E#06	90B±000 + 113 450B±000 + 113
RS1	Site Area Emergency		
	Equilibrium Stra MIRDD	1.57E±07	3790B401
RG1	General Emergency		4750B/t01+x+m+33
	Band Galeulation	1.57EH08+	C 23:90E-027

3.0 METHOD OF ANALYSIS

Using the limiting dose at the site boundary, the release is back calculated using atmospheric dispersion models. The X/Q value used is calculated from Regulatory Guide 1.145, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants. Rather than using the most conservative meteorology, average meteorological conditions are used as inputs



to most closely agree with STP emergency dose assessment methodology per the ODCM and STAMPEDE. Assumed nuclide inventories are taken from Reference 5.4. The dose conversion factors are taken from Reference 5.2. A release concentration is used to find an initial projected dose at the Site Boundary. Using the projected dose at the Site Boundary, the release concentration is scaled to find the limiting dose for each EAL.

4.0 INPUTS

- The Unit Vent flow rate is taken from the Offsite Dose Calculation Manual; Revision 17, March 2011 and is 9.44E+07 cc/sec.
- The main steam line pressure and PORV choke flow rate were taken from Reference 5.5 and are 1285 psig and 1.05E+06 lbm/hr respectively.
- The specific volume of saturated steam at this pressure is taken from the NIST steam tables and is 0.338 ft³/lbm.
- The release concentration is varied to find the release concentration which correlates to each emergency action level dose. Emergency action level doses are taken from NEI 99-01 Revision 6 for initiating conditions AA1, AS1 and AG1. EAL 1 is the EAL of interest in each initiating condition. The limiting doses are listed in Table 4.1. NEI 990-01 Revision 6 states that these
 values are based on fractions of the Environmental Protection Agencies Protective Action
 Guidelines (EPA PAGs) and the General Emergency represents the protective action values recommended by the EPA.

Table 4.1 EAL Thresholds

	Alert	Site Area	General
TEDE		100 mrem	1000 mrem
Thyroid CDE	50 mrem	500 mrem	5000 mrem

- A release lasting one hour is selected per NEI 99-01 Revision 6 developer notes.
- Atmospheric dispersion factors are calculated per Regulatory Guide 1.145 (Reference 5.1). The reactor building dimensions used as inputs for this calculation are taken from Reference 5.13.
- Nuclide inventories are taken from TGX/THX 3-1, (Reference 5.4) which is the source document for the nuclide inventories used in STAMPEDE. The release inventories are a gap release and noble gases plus 0.2% iodine which are listed below. Each nuclide inventory was normalized to one so it could be scaled to various release activities.

ENERCON ExcellenceDeery project. Freisy day	Radiological Release Thresholds for Emergency Action Levels Attachment 1	CALC. NO. STPNOC013-CALC-002 REV. J PAGE NO. 14 of 49
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	· · · · ·					1
i te a se	Nuclide	Activity (µCi/cc)	Normalized	Nuclide	Activity (µCi/cc)	Normalized
	WILLIAM STREET	$(\mu C I C C)$		STATE STATE	The second s	
	EI-131	ALLOE HOSE	\sim 112E $_{*}03$ \sim	Xe-135	——5150B带06。	5562E-02
	I-132	1.50E+05	1.53E-03	Xe-137	1.90E+07	1.94E-01
	1-133 - 2	12.20E+05	2;2;5E-03 T	Xe-138	- 1080E+07~	1,84E-04
	I-134	2.40E+05	2.45E-03	Cs-134	3.70E+01	3.78E-07
4 . <u>5</u> .2	1/185	2.00E+05		C S-137/	2-2-90E+01C	207E-0720
	Kr-83m	1.30E+06	1.33E-02	Te132	4.80E+00	4.91E-08
	-Kw-85m	2590E## 0 6#		M099	1.16.2/2458-01	1.25E-07.05
	Kr-85	3,70E+05	3.78E-03	Ru103	8,80E-03	9.00E-11
	Kr-87* .	~~5 <u>151015</u> 47066	5.62F-02	Rull06 .	290E-03	192097E-10
	Kr-88	7.80E+06	7.98E-02	Zr95	1.10E-02	1.12E-10
1.11	Ki28914		9.72E022	La140.52	1.90E-02	1.941.10
	Xe-131m	1.10E+05	1.12E-03	Ce144	7.40E-03	7.57E-11
- i	Xe_133m	6-80F=05=	6.98E-08	-Ce-141	1.00E-02.	271L02E-10
	Xe-133	2.20E+07	2.25E-01	Sr89	6.40E-02	6.55E-10
	Xe=135m;*	4;2012年06年	4-31015-102	\$590	. S. 2016, 05 2	45.3.27B-11.5

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Table 4.2 Gap Inventory

Table 4.3 Noble Gases+0.2% Iodine Inventory

 $N_{i}^{(1)}$

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Nuclide	Inventory	Normalized
121318 8 1 4		2 26E-04
I-132	8.61E-02	3.19E-04
4-133	10(0)E-108[
I-134	1.86E-02	6.92E-05
4-135 · · ·	217/3E_0104**	
Xe-131m	2.80E+00	The second s
Xe-135.	2.401-102-1	8,90E-018
Xe-133m	4.20E+00	1.56E-02
•Xe=135	7.60日年00>	22.82E-02
Xe-135m	4.00E-01	1.48E-03
Xe-137	1360E-01	\$\$\$\$5,93E-04a-85
Xe-138	5.80E-01	2.15E-03
- Kr-83m.	\$\$3.70E-01	2.2.1337E-03
Kr-85	7.60E+00	2.82E-02
Kr-85m,	-1.60E+00	5.56E-03.
Kr-87	9.80E-01	3.63E-03
Kr-88	2 80E±00	1404E-02-5
Kr-89	8.40E-02	3.12E-04

The dose conversion factors taken from EPA 400R92001 (Reference 5.2) are listed in Tables 4.4 6 and 4.5 below.



	Dose Conversion		Dose Conversion Factor	
Nuclide	Factor (rem per uCi*hr/cc)	Nuclidé	(rem per uCi*hr/cc)	
I-131	5.30E404	Xe-135	1,40E+02	
I-132	4.90E+03	Xe-137	1.10E+02	
·I-133	J 50B+04	Xe-138	》。月20日±02	
I-134	3.10E+03	Cs-134	6.30E+04	
1-135	8.10E#03	Cs-137	4-10E+04	
Kr-83m		Te132	1.20E+04	
'.Kr-85m	9.30E+01-5	Mo99	5-20E#03	
Kr-85	1.30E+00	Ru103	1.30E+04	
Kr-87	0.5.10E±02.	Rul06	<	
Kr-88	1.30E+03	Zr95	3.20E+04	
Kr=89	11-20E-t03	La140	>> 1.10E+04	
<u>Xe-131m</u>		_Ce144	4.50E+05	
Xe-133m	L.70E+01	Ge-141	**************************************	
Xe-133	2.00E+01	Sr89	5.00E+04	
Xe-135m	-2,50E±02	Sr90:	1.60E#06.0	

Table 4.4 TEDE Dose Conversion Factors

Table 4.5 Thyroid CDE Dose Conversion Factors

	Thyroid CDE DCF
Nuclide	(rem per uCi*hr/cc)
<u>(1-13)</u>	Letter LESOE 06
I-132	7.70E+03
1-133	4
I-134	1.30E+03
	3780E-04****

• The unit vent noble gas monitor energy efficiency by nuclide is taken from Offsite Dose Calculation Manual (Reference 5.3). The values are relative to Xe-133 efficiency since the monitor is calibrated to Xe-133. Table 4.6 displays the energy efficiency by nuclide relative to Xe-133.

D	E	N	- 11 - L	. 1			•••	
		Excel	алсе-	-Ever	y proje	nt Eve	y day	

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Table 4.6 Energy Efficiency Relative to Xe-133

Nuclide (uCi/cc) _{etutivalent} Kr.83m. 1.9 Kr.85 2.4 Kr.85 2.4 Kr.85 2.4 Kr.88 3 Kr.88 3 Kr.89 2.8 Ke-133m 0.14 Xe-135m 0.042 Xe-135m 2.8			Efficiency Relative to Xe-133
Kr-85m 1.9 Kr-85 24 Kr-87 2.8 Kr-89 2.8 Xe-131m 0.012 Xe-135m 0.042 Xe-135s 0.042		Nuclide	(uCi/cc) _{equivalent}
Kr.85 24 Kr.87 2.8 Kr.88 3 Kr.89 2.8 Xe-133m 0.14 Xe-135m 0.042 Xe-135 0.042	11 1	Kr-83m	
Kr-87 2.8 Kr-89 2.8 Kr-89 2.8 Xc-133m 0.14 Xc-135m 0.042 Xc-135s 0.042		Kr-85m	1.9
Kr.88 2.8 Kr.89 2.8 Xe-133m 0.014 Xe-135m 0.042 Xe-135m 0.042 Xe-135m 0.042		Kr-85	244
Kr-89 2.8 Xc-131m 0.0015 Xc-133m 0.14 Xc-135m 0.042 Xc-135s 0.042	·	Kr-87	2.8
Xe-131m 0.015 Xe-133m 0.14 Xe-133 0.042 Xe-135 0.042		Kr-88	
Xe-133m 0.14 Xe-133 1 Xe-135m 0.042 Xe-1355 2	i i i i	Kr-89	2.8
Xe-133 Xe-135m 0.042 Xe-135		exe-131m.	0:015
Xe-135m 0.042 Xe-135 2		Xe-133m	0.14
XE135		Xe4133	
		Xe-135m	0.042
Xo 137 28		Xe-135	25.46.55
	· 	Xe-137	2.8

Xe-138 *There is no relative efficiency available for Kr-83m. Assumption 6.4 further justifies the omission. ν. γ.

n (m. 1916) - Angela	Table 4.	7 Nuclide Half	Lives
Nuclide	Half Life	Nuclide	Half Life
1-131	(hr) 93E+022=	Xe-135	(hr) 9.08E+000*
I-132	2.38E+00	Xe-137	6.38E-02
11-133	C 2 OBEE014	Xe-138	236E01.0
I-134 1-135	8.77E-01	Cs-134 Cs-137	1.80E+04
Kr-83m	1.83E+00	Te132	7.79E+01
Kr-85m	4.485400	M099	1.46.62E401
Kr-85	9.40E+04	Ru103	9.44E+02
Kr-88	2.84E+00	Ra106.	28.84E#03
K1-80	5710H-02	La140	4.03E+01
Xe-131m	2.83E+02	Ce144	6.82E+03
Xe-133m.	3715 42E401	Ce-141	1.21E+03
Xe-133 Xe-135m	1.27E+02	Sr89 Si90	1.21E+03
			and the second secon

The half-lives are taken from Reference 5.15 which lists the input data used by STAMPEDE. 6

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5.0 REFERENCES

- 5.1 Regulatory Guide 1.145, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants, Revision 1, November 1982.
- 5.2 EPA 400R92001, Manual of Protective Action Guides and Protective actions for Nuclear Incidents, Revision 1, May 1992.
- 5.3 Offsite Dose Calculation Manual, Revision 17, March 2011.
- 5.4 TGX/THX 3-1, Revision 5, Westinghouse Radiation Analysis Manual.
- 5.5 MC05591, Main Steam PORV Capacity Verification, Revision 1.
- 5.6 NIST Steam Tables, 2011.
- 5.7 0ERP01-ZV-IN01, Emergency Classification, Revision 10.
- 5.8 0ERP01-ZV-TP01, Offsite Dose Calculations, Revision 21.
- 5.9 STP Calculation NC-9012, Process and Effluent Radiation Monitor Set Points, Revision 7
- 5.10 STP Calculation NC-9011, CRMS Rad Monitor Setpoints, Revision 2.
- 5.11 STAMPEDE Computer Program, Revision 7.0.3.3.
- 5.12 STAMPEDE User's Manual
- 5.13 STP Drawing 6C189N5007, General Arrangement Reactor Containment Building, Revision 6
- 5.14 NEI 99-01, Revision 6, Development of Emergency Action Levels for Non-Passive Reactors
- _5.15 _ITWMS Call Number 1000010987 Design Document, Revision 0_____

6.0 ASSUMPTIONS

6.1 Release lasts for one hour

Per NEI 99-01 (Reference 5.14), IC AA1, AS1, AG1 developer notes, the release should be assumed to last one hour.

For this to be true for the main steam line, it is assumed that the PORV is open for one hour. To calculate the most limiting case, it is assumed that the maximum flow possible is being released from the PORV.

6.2 Nuclide mix

<u>Per 0ERP01-ZV-TP01, Offsite Dose Calculations (Reference 5.8) any unit vent release with</u> increased RCS activity and no core melt should be calculated using a gap inventory. It is conservative to assume an increased RCS activity and not within the intended scope of the relevant initiating conditions to assume core melt. Therefore, a gap inventory is used for all unit vent releases.

Per 0ERP01-ZV-TP01, Offsite Dose Calculations (Reference 5.8) for a main steam line release following a steam generator tube rupture it is appropriate to use an inventory of 100 percent noble gases plus 0.2 percent iodine. Since a steam generator tube rupture releasing through the PORVs is the only steam generator tube rupture scenario which would create offsite doses large enough to meet or exceed the EALs, this assumption is made.



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6.3 Atmospheric Dispersion

NEI 99-01 (Reference 5.14) developer notes for initiating conditions AA1, AS1 and AG1 suggest using the ODCM or the site's emergency dose assessment methodology. Per 0ERP01-ZV-TP01, Offsite Dose Calculations (Reference 5.8), when actual meteorology is not available, the default STAMPEDE values should be used. The default STAMPEDE values assume a stability class D for atmospheric dispersion and a windspeed of 13.2 mph. These values were used as inputs for the atmospheric dispersion calculation.

It is clear that STAMPEDE uses the same method for calculating atmospheric dispersion factor (X/Q) outlined in section 7.1.1 of this Attachment. However, STAMPEDE does not follow the same logic in selecting the appropriate result from the three calculations. The STAMPEDE value printed in the results found in attachment 3 is consistent with the largest of the three hand calculated X/Q values. This suggests that STAMPEDE simply selects the largest of the three X/Q values resulting in a much more conservative estimate. This calculation will deviate from the recommendations of Regulatory Guide 1.145 and conform to the methodology STAMPEDE uses.

The close proximity of all release points allows for a single atmospheric dispersion coefficient to be used. This assumption is also made by STAMPEDE.

6.4 Exposure Pathways

The dose conversion factors used in table 4.4 and 4.5 represent a summation of dose conversion factors for external plume exposure, inhalation from the plume, and external exposure from deposition. Because the dose estimations are used for implementing early phase protective actions, conversion factors using limited pathways are appropriate.

The EPA does not provide a dose conversion factor for Kr-83m. Because the PAGs are based on EPA dose calculations, it is appropriate to only use the nuclides for which dose conversion factors are provided. Additionally, Kr-83m represents only 1.33% of the nuclide inventory activity and its exclusion would not significantly affect the final dose.

6.5

The release initiates one hour after reactor shutdown. While a release initiating at reactor shutdown is likely, significant decay of short lived nuclides occurs during the migration time. A release at reactor shutdown would have a significantly higher activity at the monitor location than at the reception site. It is important for the threshold to not be calculated at shutdown as this would create a very high threshold which would not be appropriate for releases which occur shortly after shutdown. One hour after reactor shutdown is sufficient time to decay short lived nuclides and create a conservative threshold.

Decay is incorporated for one hour from reactor shutdown as well as migration time. Half-lives are taken from Reference 5.15. Migration time is assumed to be the reciprocal of the wind speed.

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7.0 HAND CALCULATIONS

7.1 Unit Vent Monitor

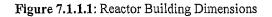
7.1.1 X/Q

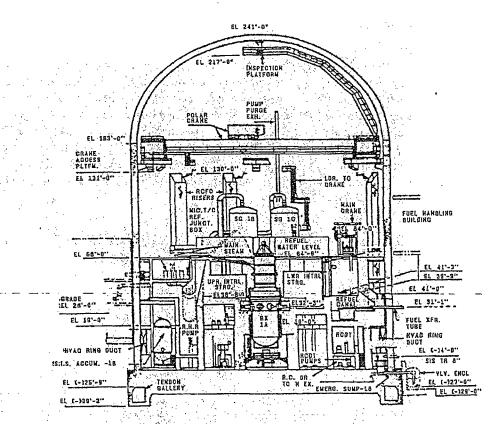
The atmospheric dispersion factor, X/Q, determines the change in concentration between the unit vent discharge and the dose reception site. This value is based on meteorological conditions and will vary with wind speed and stability class. The ODCM uses the highest annual average X/Q value at the site boundary which is 5.3E-06 sec/m³. However, for an accident related release STAMPEDE is used rather than the ODCM. STAMPEDE uses real time, user entered, or default meteorological conditions to calculate the X/Q for a specific accident. Default values will be used as inputs into the Regulatory Guide 1.145 method for calculating X/Q as described below. Default values are identified in section 6.0, Atmospheric Dispersion.

For a neutral atmospheric stability class, which is the default in STAMPEDE, X/Q values can be determined through the following set of equations.

·	. _	
•	$\frac{X}{Q} = \frac{1}{\overline{U}_{10} \left(\pi \sigma_y \sigma_z + \frac{A}{2} \right)}$	
		Equation 7.1.1.1
	$\frac{X}{Q} = \frac{1}{\overline{U}_{10}(3\pi\sigma_y\sigma_z)}$	
	$z = b_{10}(5\pi b_y b_z)$	Equation 7.1.1.2
	$\frac{X}{Q} = \frac{1}{\overline{U}_{10}\pi\Sigma_{12}\sigma_{7}}$	-
	$Q = U_{10}\pi\Sigma_y\sigma_z$	Equation 7.1.1.3
Where		Бушиноп 7.1.1.5
X/Q	= relative concentration (sec/m^3)	
$\pi \overline{U}_{10}$	= 3.14159 = windspeed at 10 meters above plant grade (m/s)	_
$\frac{\sigma_{10}}{\sigma_v}$	= lateral plume spread (m), a function of atmospheric stability	and distance,
-y	determined from Regulatory Guide 1.145 Figure 1	
σ_z	= vertical plume spread (m), a function of atmospheric stability determined from Regulatory Guide 1.145 Figure 2	and distance,
Σ_y	= $(M-1)\sigma_{y800m} + \sigma_y$ = lateral plume spread with meander a	
	effects (m), a function of atmospheric stability, windspeed \overline{U}_{10} is determined from Regulatory Guide 1.145 Figure 3	, and distance; M
Å	= the smallest vertical-plane cross-sectional area of the reactor taken from Reference 5.13 and shown below	or building (m^2),

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Assuming the reactor building cross section to be a perfect rectangle and half sphere, the variables are defined as follows;

101 - 101 -

$$\overline{U}_{10} = 13.2 \text{ mph} = 5.9 \text{ m/s}$$

 $\sigma_{\mathbf{y}} = 1200 \text{ m}$ $\sigma_{\mathbf{z}} = 4.2 \text{ m}$ $\Sigma_{\mathbf{y}} = (M-1)\sigma_{\mathbf{y}800m} + \sigma_{\mathbf{y}}; \text{ M=1} \rightarrow \sigma_{\mathbf{y}} = 1200 \text{ m}$ $A = (135' * 158') + \left(\frac{\pi * 79^2}{2}\right) = 31128.37$ The three equations become;

$$\frac{X}{Q} = \frac{1}{5.9\left(\pi 1200 * 4.2 * \frac{31128.37}{2}\right)} = 5.398 * 10^{-6}$$



$$\frac{X}{Q} = \frac{1}{5.9(3\pi * 1200 * 4.2)} = 3.568 * 10^{-6}$$
$$\frac{X}{Q} = \frac{1}{5.9 * \pi * [(1-1)\sigma_{y800m} + 1200] * 4.2} = 1.07 * 10^{-5}$$

To select the appropriate X/Q value, the first two X/Q values should be compared and the higher value selected. This value is then compared with the third X/Q value and the lower of those two is the appropriate X/Q value. The appropriate X/Q is 5.39E-06 sec/m³ for default meteorological conditions by the methodology recommended in Regulatory Guide 1.145.

This calculated value is very similar to the ODCM highest average value of 5.3E-06 sec/m³ which was not selected for use. Additionally, the value shown in the STAMPEDE output file at one mile is 1.032E-05 sec/m³. This suggests that STAMPEDE uses the same methodology and simply selects the largest atmospheric dispersion value to remain conservative. This methodology will be replicated and 1.07E-05 will be used as the X/Q.

7.1.2 Nuclide Inventory

As previously stated, a gap inventory is appropriate for this problem. The gap inventory is taken from TGX/THX 3-1 (Reference 5.4) which is used as the source term for STAMPEDE inventories. The concentrations were then normalized so they could be scaled to the varying emergency classifications. The values for the normalized inventory can be found in Table 4.2.

7.1.3 Dose Conversion Factors

As stated in NEI99-01 (Reference 5.14) developer notes, the purpose of dose projections is to check if the Environmental Protection Agencies Protective Action Guidelines (EPA PAGs) have been exceeded. The dose conversion factors provided by the EPA in EPA 400R92001 are used. These dose conversion factors account for external plume exposure, inhalation from the plume, and external exposure from deposition and are listed Tables 4.4 and 4.5, and taken from tables 5-1, 5-2 in EPA 400R92001 (Reference 5.2).

The EPA does not provide a dose conversion factor for Kr-83m. This nuclide contributes 1.33% of the inventory activity. The lack of this nuclide's contribution to the final dose will not significantly affect the outcome.

7.1.4 Decay Time

One hour of decay is incorporated to the monitor response due to the release initiating one hour after reactor shutdown. Decay is also incorporated for the duration of the migration time. The total decay time is one hour plus the reciprocal of wind speed, or 1.07575 hours.



7.1.5 Dose Calculations

The dose rate at the site boundary is calculated using Equation 7.1.5.1.

$$\dot{D} = \frac{X}{Q}F\sum_{i}^{n}C_{i} * 0.5^{\frac{1.07575}{T_{1/2}i}} * DCF_{i}$$

Equation 7.1.5.1

Where

 \dot{D} = dose rate per hour at the site boundary

= atmospheric dispersion coefficient as calculated in section 7.1.1

F =unit vent flow rate

 C_i = concentration of nuclide i at the time of shutdown

1.07575 = the total decay time of interest from section 7.1.4

 $T_{1/2i}$ = the half-life of nuclide i

 DCF_i = the dose conversion factor for nuclide i listed in tables 4.4 and 4.5

The total concentration of the nuclides is varied to find the dose rate of interest. Beginning with an arbitrary release concentration of 1 μ Ci/cc, the dose rate is calculated. Since the dose is linearly correlated to concentration, the release concentration may be scaled to find the dose rate of interest.

The Alert EAL is 10 mrem TEDE or 50 mrem Thyroid CDE. Using the above method to calculate TEDE with the appropriate conversion factors, a limiting release rate of $2.33E+06 \ \mu Ci/sec$ from the unit vent results in 5.7 mrem TEDE. Using the calculated release rate to find Thyroid CDE with the appropriate conversion factors, the same release results in a 50 mrem Thyroid CDE at the site boundary. Thus, $2.33E+06 \ \mu Ci/sec$ is the limiting release rate based on the 50 mrem Thyroid CDE EAL initiating condition.

The limiting release rate threshold values for the Site Area Emergency and General Emergencies are multiples of 10 and 100 of the Alert release rate threshold value.

These calculations can be found in Attachment 2.

7.1.6 Monitor Response

The unit vent noble gas monitor is calibrated to Xe-133. Monitor efficiencies relative to Xe-133 by nuclide are listed in ODCM Table B3-2. To find the monitor reading associated with each limiting release, the noble gas concentrations must be multiplied by the monitor response and summed. Table 4.6 shows the indicated response of the unit



vent noble gas monitor by nuclide and Equation 7.1.5.1 shows how the monitor response was calculated.

Monitor Response =
$$\sum_{i}^{n} C_{i} * Re_{i}$$

Equation 7.1.5,1

Where

 C_i = concentration of nuclide i (µCi/cc)

 Re_i = monitor response to nuclide i (μ Ci/cc)_{Xe-133} equivalent</sub>

In the case of an Alert, the 2.33E+06 μ Ci/sec release rate will read as 1.57E+06 μ Ci/sec on the monitor. Kr-83m does not have an indicated monitor response coefficient. Because Kr-83m is only 1.34% of the noble gases and does not contribute to the dose calculation, its exclusion is acceptable.

This again is a linear correlation and the SAE and GE scale by factors of 10 and 100 respectively.

These calculations can be found in Attachment 2.

7.2 Main Steam Line Monitors

7.2.1 X/Q

Since the atmospheric dispersion is independent of nuclide inventory or release rate and the close proximity of the releases, the X/Q value will be the same for a main steam line release as it is for a unit vent release. This assumption is also taken by STAMPEDE and outline in Assumption 6.3.

7.2.2 Nuclide Inventory

Per 0ERP01-ZV-TP01, if the release path is the main steam line with a steam generator tube rupture, the nuclide inventory should be 100% noble gas and 0.2% of the iodine from the reactor coolant.

The secondary steam concentration for noble gases and iodine after a steam generator tube rupture are taken from TGX/THX 3-1 (Reference 5.4). Values for the reactor coolant inventory are listed in table 4.3. All of the noble gases are used and the iodine concentration from the coolant inventory is scaled to total 0.2% of iodine in the total coolant inventory. These inventories are then normalized to one. These values are listed in Table 4.3.

7.2.3 Dose Conversion Factors

The dose conversion factors used are found in Tables 4.4 and 4.5, taken from tables 5-1, 5-2 in EPA 400R92001.



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7.2.4 Decay Time

One hour of decay is incorporated to the monitor response due to the release initiating one hour after reactor shutdown. Decay is also incorporated for the duration of the migration time. The total decay time is one hour plus the reciprocal of wind speed, or 1.07575 hours.

7.2.5 Dose Calculations

Equation 7.1.5.1 applies to the release from the main steam lines. The main steam line flow rate is used instead of the unit vent flow rate for the value F. The main steam line flow rate was calculated in Equation 7.1.2.2 of the STAMPEDE CALCULATIONS section of this document as 2.79E+06 cc/sec.

The Alert EAL threshold is 10 mrem TEDE or 50 mrem Thyroid CDE at the site boundary (Table 4.2). Using the method in Equation 7.1.5.1 to calculate TEDE with the appropriate conversion factors, a concentration at time of shutdown of 4.10 μ Ci/cc would result in 6.89 mrem TEDE at the site boundary if the steam line PORV was open for an hour. Using the same steam line concentration to calculate Thyroid CDE results in 50 mrem Thyroid CDE at the site boundary.

The steam line concentrations at the time of shutdown for the Site Area Emergency and General Emergencies are multiples of 10 and 100 of the Alert. Since the correlation between release concentration and dose is linear, values for the steam line concentration at time of shutdown are 41.0 and 410 μ Ci/cc for the SAE and GE respectively. Both are also limited by Thyroid CDE.

These calculations can be found in Attachment 2.

7.2.6 Monitor Response

Because the main steam line monitor is adjacent to the main steam line, significant shielding takes place between the source and monitor. STP calculation NC-9011 Revision 2 calculates a conversion factor for the main steam lines for a noble gas inventory which is incorporated into the monitor readout. No monitor response needs to be calculated.

The concentration of the main steam line one hour after shutdown given a concentration of 4.10 μ Ci/cc at the time of shutdown is 3.90 μ Ci/cc. This calculation is also found in Attachment 2. Additionally, the monitor readings for the SAE and GE one hour after shutdown are 39.0 and 390 μ Ci/cc respectively. These values are the thresholds for the main steam line monitor.

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Table A2-1: Unusual Event Emergency Calculations

Unit Non			and the interview of the second se
	1.48E-03	9.44E+07	1.40E+05
Misir	imiting Release Kate	Flow Rate	Emp <u>ina</u> timentation
	1.40E+05	2.79E+06	5.00E-02

Table A2-2: Input Values for Calculations

Santa Maria Anglas	dimation	Rolease Refe	Releases orstant St.	Unit Conversion disc Refersor Constant	als tomic Release	Trean Thure
5.40E-06	3600	9.44E+07	1.83E+06	5.10E-04	1.79E-02	1.07575
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Table A2-3: Calculations for Boundary Concentrations and TEDE dose due to Unit Vent Release

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		NATA AND AND A		Rolesse	Conception	Phate Lore	Decayed	LING	Dose
Nucida	Intentory	Normalized	NARA CONTRACTOR	Constant	- AND SOT TERMS A		Ann cent internal	Conversion	Commission
e contensi rec			A REAL PROPERTY.						
	Cile Cile		ne (LErce) de	Corect 1	(uCir înzice)			uCr Intyder	14 (10) - 7 (11) 14 (10) - 7 (11) 14 (11) - 7 (11)
5 101	1.10E+05	1.12E-03	2.76E-05	1.01E-03	2.79E-08	1.93E+02	2,78E-08	5.30E+04	1.47E-03
. I-131		1.53E-03	3.77E-05	1.01E-03	3.81E-08	2.38E+00	2.79E-08	4.90E+03	1.37E-04
I-132	1.50E+05		5.55E-05	1.01E-03	5.61E-08	2.03E+01	5.40E-08	1.50E+04	8.11E-04
I-133	2.20E+05	2,25E-03	동안 제 이 가는 것같이 같이 같이 같이 같이 같이 같이 같이 않는 것이 같이 않는 것이 같이 많이	1.01E-03	6.11É-08	8.77E-01	2.61E-08	3.10E+03	8.09E-05
I-134	2.40E+05	2.45E-03	6 04E-05	an a	5.11E-08	6.61E+00	4.56E-08	8.10E+03	3.70E-04
I-135	2.00E+05	2.05E-03	5.06E-05	1.01E-03	No. Carla Company	1.83E+00	2.21E-07		0.00E+00
Kr-83m	1.30E+06	1.33E-02	3 28E-04	1.01E-03	3.31E-07	A LAND LAND AND A REAL AND A	a state in the second	9.30E+01	5.83E-05
Kr-85m	2.90E+06	2.97E-02	7.33E-04	1.01E-03	7,40E-07	4.48E+00	6.27E-07		1.22E-07
Kr-85	3.70E+05	3.78E-03	9.33E-05	1.01E-03	9.42E-08	9.40E+04	9.42E-08	1.30E+00	3.97E-04
Kr-87	5.50E+06	5.62E-02	1.39E-03	1.01E-03	1.40E-06	1.27E+00	7.79E-07	5.10E+02	
Kr-88	7.80E+06	7.98E-02	1.97E-03	1.01E-03	1.99E-06	2.84E+00	1.53E-06	1.30E+03	1.99E-03
Kr-89	9.50E+06	9.72E-02	2.40E-03	1.01E-03	2.42E-06	5.10E-02	1.08E-12	1.20E+03	1.30E-09
Xe-131m	1.10E+05	1.12E-03	2.76E-05	1.01E-03	2.79E-08	2.83E+02	2.78E-08	2.50E+02	1.36E-07
	6.80E+05	6.95E-03	171E-04	1.01E-03	1.73E-07	5.42E+01	1.71E-07	1.40E+02	2.90E-06
Xe-133m		2.25E-01	5.55E-03	1.01E-03	5.61E-06	1.27E+02	5.57E-06	1.10E+02	1.11E-04
Xe-133	2.20E+07		1.06E-03	1.01E-03	1.07E-06	2.60E-01	6.09E-08	7.20E+02	1.52E-05
Xe-135m	4.20E+06	4.30E-02	이 가지 않는 것 같아요.	1.01E-03	1.40E-06	9.08E+00	1.29E-06	5.30E+04	1.81E-04
Xe-135	5.50E+06	5.62E-02	1 39E-03	a 11日 - 11日 - 11日 - 11日本部計:		6.38E-02	4.06E-11	4.90E+03	4.47E-09
Xe-137	1.90E+07	1.94E-01	4 79E-03	1.01E-03	4.83E-06	방송 영상 영상 영상 영상	1.95E-07	1.50E+04	1.40E-04
Xe-138	1.80E+07	1.84E-01	4 54E-03	1.01E-03	4.59E-06	2.36E-01	1.95E-07	1.502104	1.100 01

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Nielide	Interiory	Normalized		ionstanti g	Manalan		ik (Constitution	in <u>Alfactor</u>	Contribution).
	CHO CHO			he core	Contract in (SS)	- CHANN	Sales (Inclusion) (Se)	as are no per dore the	CONTRACTOR MUSICINE AND STORE
Cs-134	3.70E+01	3.78E-07		1.01E-03	9.42E-12	1.80E+		6.30E+04	5.93E-07
Cs-137	2.90E+01	2.97E-07	7.33E-09	1.01E-03	7.40E-12	2.60E+	05 7.40E-12	4.10E+04	3.03E-07
Te132	4.80E+00	4.91E-08	1.2 E-09	1.01E-03	1.22E-12	7.79E+	01 1.21E-12	1.20E+04	1.45E-08
Mo99	1.22E+01	1.25E-07	3.08E-09	1.01E-03	3.11E-12	6.62E+	01 3.08E-12	5.20E+03	1.60E-08
Ru103	8.80E-03	9.00E-11	2.22E-12	1.01E-03	2.24E-15	9.44E+	02 2.24E-15	1.30E+04	2.91E-11
Ru106	2.90E-03	2.97E-11	7.33E-13	1.01E-03	7.40E-16	8.84E+	03 7.40E-16	5.70E+05	4.22E-10
Zr95	1.10E-02	1.12E-10		1.01E-03	2.79E-15	1.55E+	03 2.79E-15	3.20E+04	8.93E-11
La140	1.90E-02	1.94E-10	1	1.01E-03	4.83E-15	4.03E+	01 4.75E-15	1.10E+04	5.22E-11
Ce144	7.40E-03	7.57E-11		1.01E-03	1.89E-15	6.82E+	03 1.89E-15	4.50E+05	8.49E-10
Ce-141	1.00E-02	1.02E-10		1.01E-03	2.54E-15	7.77E+		1.10E+04	2.79E-11
Sr89	6.40E-02	6.55E-10		1.01E-03	1.63E-14	1.21E+		5.00E+04	8.16E-10
Sr90	3.20E-03	3.27E-11		1.01E-03	8.15E-16	2.50E+		1.60E+06	1.30E-09
2120	5.201-05	J.212 11				1	1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 -	Total TEDE Do	and a second a second second second
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Table A2-4: Thyroid Dose Calculation for Unit Vent Release

Nutchicke 71	Charged Content	Thyroid DCT.	Mineratii Dosa
	the (mattice incore) as the me	manendication	
I-131	2.78E-08	1.30E+06	3.61E-02
I-132	2.79E-08	7.70E+03	2.15E-04
I-133	5.40E-08	2.20E+05	1.19E-02
I-134	2.61E-08	1.30E+03	3.39E-05
I-135	4.56E-08	3.80E+04	1.73E-03
		Total Rhabid	
		Doxe	-1.5001-02

Table A2-5: Unit Vent Monitor Response to Nuclide Inventory

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- Nitalider - C	oncentration.	AANTI MC	Concentration R	esponse . Cocii	NESTONS
	(uestree)	Holins		fic.v.cc) - P	
Kr-83m	\$.28E-04	1.83E+00	2.25E-04		0.00E+00
Kr-85m	7.33E-04	4.48E+00	6.28E-04	1.9	1.19E-03
Kr-85	9.33E-05	9.40E+04	9.33E-05	2.4	2.24E-04
Kr-87	1.39E-03	1.27E+00	8.03E-04	2.8	2.25E-03
Kr-88	1.97E-03	2.84E+00	1.54E-03	2.3	3.55E-03
Kr-89	2.40E-03	5.10E-02	3.00E-09	2.8	8.40E-09
Xe-131m	2.76E-05	2.83E+02	2.76E-05	0.015	4.13E-07
Xe-133m	1.71E-04	5.42E+01	1.69E-04	0.14	2.37E-05
Xe-133	5.55E-03	1.27E+02	5.52E-03	1 - 1	5:52B-03
Xe-135m	1.06E-03	2.60E-01	7.38E-05	0.042	3.10E-06
Xe-135	1.39E-03	9.08E+00	1.28E-03	2.5	3.21E-03
Xe-137	4.79E-03	6.38E-02	9.15E-08	2.8	2.56E-07
Xe-138	4.54E-03	2.36E-01	2.41E-04	2.8	6.74E-04
			Monitor	Reading:	Har Har Ester Charles I in

(uCi/cc)

(uCi/sec)

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Table A2-6: Input for Main Steam Line Release Calculation

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	Relex		luma aya aya			nversion st	n (na star da star star star star References en star star star star star star star star		
1.05E+06	3.37E-	+06 0	338	3.55E+05	2.79	E+06	4.05		1.07575
1.050 00	5.575				17 7) 10				
					i.				
:: (and table at 10 a court of the group region of the court of the	Tab	le A2-7: Calcula				dose due to M	and the second	Release	
Ser Nitching	Sieom.	Normalized	Varied menuisiion		Concentration : 10/Brandary	Hallo	Concentration	CODER	Doscusa
	ALINA COCORA		nreenuenuen.		(a) b) an carry		ACTORICAL MULTIPLE		
							to (n.C.t. mee) or	Curch hater is	
I-131	6.10E-02	2.26E-04	9.27E-04	2.9853E-05	2.77E-08	1.93E+02	2.76E-08	5.30E+04	1.46E-03
I-132	8.61E-02	3.19E-04	1.31E-03	2.9853E-05	3.90E-08	2.38E+00	2.85E-08	4.90E+03	1.40E-04
I-133	1.00E-01	3.72E-04	1.53E-03	2.9853E-05	4.55E-08	2.03E+01	4.39E-08	1.50E+04	6.58E-04
I-134	1.86E-02	6.92E-05	2.84E-04	2.9853E-05	8.47E-09	8.77E-01	3.62E-09	3.10E+03	1.12E-05
I-135	2.73E-01	1.01E-03	4.14E-03	2.9853E-05	1.24E-07	6.61E+00	1.10E-07	8.10E+03	8.95E-04
Xe-131m	2.80E+00	1.04E-02	4.26E-02	2.9853E-05	1.27E-06	2.83E+02	1.27E-06	4.90E+00	6.22E-06
Xe-133	2.40E+02	8.90E-01	3.65E+00	2.9853E-05	· 1.09E-04	5.42E+01	1.07E-04	2.00E+01	2.15E-03
Xe-133m	4.20E+00	1.56E-02	6.40E-02	2.9853E-05	1.91E-06	1.27E+02	1.90E-06	1.70E+01	3.23E-05
Xe-135	7.60E+00	2.82E-02	1.16E-01	2.9853E-05	3.45E-06	2.60E-01	1.96E-07	1.40E+02	2.75E-05
Xe-135m	4.00E-01	1.48E-03	6.07E-03	2.9853E-05	1.81E-07	9.08E+00	1.67E-07	2.50E+02	4.17E-05
Xe-137	1.60E-01	5.93E-04	2.43E-03	2.9853E-05	7.26E-08	6.38E-02	6.10E-13	1.40E+02	8.53E-11
Xe-138	5.80E-01	2.15E-03	8.82E-03	2,9853E-05	2.63E ₇ 07	2.36E-01	1.12E-08	7.20E+02	8.04E-06
Kr-83m	3.70E-01	1.37E-03	5.62E-03	2.9853E-05	1.68E-07	1.83E+00	1,12E-07		0.00E+00
Kr-85	7.60E+00	2.82E-02	1,16E-01	2.9853E-05	3.45E-06	4.48E+00	2.92E-06	1.30E+00	3.80E-06
Kr-85m	1.50E+00	5.56E-03	2.28E-02	2.9853E-05	6.81E-07	9.40E+04	6.81E-07	9.30E+01	6.33E-05
Kr-87	9.80E-01	3.63E-03	1.49E-02	2.9853E-05	4.44E-07	1.27E+00	2.47E-07	5.10E+02	1.26E-04
Kr-88	2.80E+00	1.04E-02	4.26E-02	2.9853E-05	1.27E-06	2.84E+00	9.79E-07	1.30E+03	1.27E-03
Kr-89	8.40E-02	3.12E-04	1.28E-03	2.9853E-05	3.82E-08	5.10E-02	1.71E-14	1.20E+03	2.05E-11
								Total Dose	6.89E-03

*Release Constant = X/Q * duration * release rate

Section and the

	Radiological Release Thresholds	CALC. NO. STPNOC013-CALC-002
FNERCOI	for Emergency Action Levels	REV. 1
Excellence-Every project Every	Kay Attachment 2	PAGE NO. 30 of 49

Table A2-8: Main Steam Line Release Thyroid Dose Calculation

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	relide 1	and oncentration and an	Priving DCF 2016	e fluyeoù Er Dosene
Not State		and realities and a strengthe	m delau Criffin de	Alema -
I-131		2.76E-08	1.30E+06	3.58E-02
I-132		2.85E-08	7.70E+03	2.20E-04
I-133		4.39E-08	2.20E+05	9.66E-03
I-134		_3.62E-09	1.30E+03	4.71E-06
I-135		1.10E-07	3.80E+04	4.20E-03
		A State of the second sec		4.991-02

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Radiological Release Thresholds for Emergency Action Levels Attachment 2

CALC. NO. STPNOC013-CALC-002 REV. 1 PAGE NO. 31 of 49

Table A2-9: Main Steam Line Reading at Release

Nordinie	Concentitation	THAT LAIR	Concentration - 1 Bonzatter - Shartava
I-131	9.27E-04	1.93E+02	9.23E-04
I-132	1.31E-03	2.38E+00	9.77E-04
I-133	1.53E-03	2.03E+01	1.47E-03
I-134	2.84E-04	8.77E-01	1.29E-04
I-135	4.14E-03	6.61E+00	3.73E-03
Xe-131m	4.26E-02	2.83E+02	4.25E-02
Xe-133	3.65E+00	5.42E+01	3.60E+00
Xe-133m	6.40E-02	1.27E+02	6.36E-02
Xe-135	1.16E-01	2.60E-01	8.04E-03
Xe-135m	6.07E-03	9.08E+00	5.62E-03
Xe-137	2.43E-03	6.38E-02	4.65E-08
Xe-138	8.82E-03	2.36E-01	4.67E-04
Kr-83m	5.62E-03	1.83E+00	3.85E-03
Kr-85	1.16E-01	4.48E+00	9.90E-02
Kr-85m	2.28E-02	9.40E+04	2.28E-02
Kr-87	1.49E-02	1.27E+00	8.62E-03
Kr-88	4.26E-02	2.84E+00	3.34E-02
Kr-89	1.28E-03	5.10E-02	1.60E-09
		Though Active y	3, 201E +0(0).



CALC. NO. STPNOC013-CALC-002

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REV 1 PAGE NO. 32 of 49

STAMPEDE User Supplied Information Revision 7.033 9/28/2011 Uber Manne: Unit Vent Alert Date/Time: 12/17/2013 15:24 Connacuts ther Supplied Information Meteorological Data Ioputa: Graundlevel undvelocity: 13.2 mi/hr Groundlevel windfrom: 159 degrees 2.32 ther-selectedStability Class "D - Neutral" Stability Class: Monifored Unit Vest Release: Unit Vent Release Rate enterest 2.50E+606 BCDisec 12/17/2013 14:24 Reartor Shutdown Eater Time: 12/17/2013 15:24 Referse Start Date Time: 1.00 hours Estimated Release Duration: Confinentery Nuclide Mature: Canadated NOHLE GAS release rate: 1.199-006 aClises NOHLEGAS IODINE PARTICULATE uCi/sec Nuclide **nCi/sec** Nucâde nCi/sec Norlide Kr-8334 2535-1004 1111: 3128-003 Cs-134. 1.05E+000 I-131: 3.188+008 Cs-137: 2.25E-001 1052+004 Kr-85: T-133: 6.0584008 CePr-144: 110E-004 7068+604 KT-8514 Ce-I4I: 2.34B-004 Kr-87: 943E+804 1-134. BUZEHUB La-140: 531E-004 5.128+083 Kr-88: 1.74E+005 1-135: 100-300.E Mo-99: 3.438-001 Kr-89: 8256-005 Ra Rh-106. Xe-33114 3.12E+003 2.538-004 Ra-103: 622E+005 Xe-133: 9.108-005 Sr/¥-90: 191E+604 1.82E-003

Sr 89:

Te-132:

Zr-95:

135E-001

3.138-004

Xe-133M Xe-135: 145E+C05 Xz-135M 814E+COB

9.535-4600

262+004

Xe-137:

λe-198:

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Radiological Release Thresholds for Emergency Action Levels Attachment 3

CALC. NO. STPNOC013-CALC-002 REV.1

PAGE NO. 33 of 49

DRILL	STAMPEDE Result Residon 7.03.3 9.08/2011	Page1of2	DRILI
Date/Time: 12/17/2013 15:24	iker Manne: Unit Ve		taa noone attain, aan taaaqaada
Consusents:			
	Linne Inform		enal same an and a sec
Distance	Plane Travel Line	CHIQ Value	CHUQ BEPL
(milos)	(hears:minutes)	(sector)	(3.8 <i>0¹0</i> 1 ²)
0.5	0:02	2.6862-005	2,4368-005
1.0	0:05	1.032E-005	9.110E-D05
20	6:09	3.7558-005	3.151E-005
5.0	023	1.0048-005	7.373E-007
75	0:34	5.7042-007	3.845E-007
10.0	0:45	3.8S1E-007	2.441E-007
20.0	131	1.541E-007	9.1098-008
	Messurable Dose Rafes	PAC	Dose Rates
	Immersion Whole Body	TRUE	Indina CDE
Bistanco	Rahls yes gamma	ortorasi + interasi	Thyreid
(wiles)	(rom.Gr)	(78 m/br)	(rom/hr)-
05	0.009	0.016	0.137
1.0	0.0033	0.005	0.051
20	0.001	0.002	20018
50	0.000	1001	0.004
73	0.000	0.007	0.002
10.0	0.000	0.000	0.001
20.0	0.000	0.009	0.060
	Measurable Doses	PAC	Dases
		TESE	lutine CBR
Birburg	Inspersion Whole Hody	oriornal + fatural	Thyroid
(miløs)	nopie ärz Lemma (1916)	(rem)	(rem)
0.5	0.009	0.016	0.137
1.0	0.003	0.006	0.051
20	0.001	0.002	0.013
50	0.350	0.001	0.004
75	0.000	0.003	0.062
100	0.000	0.000	0.001
20.0	0.000	0.000	0.000

12/17/2013 3:24:28384



CALC. NO. STPNOC013-CALC-002

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REV. 1

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		ulations Complet	ed framework and the second		
Method of Projecti	and the stand second	SULTS diyi 13.2 millar	Rolanao R	nte: 1.198+006uCidse	·· -
STAMPEDE	Wind Direct				
Offsite Dose Project			an de la serie de la serie La serie de la s	· · · ·	
	nile 3 miles 0.006 0.002	5 miles 0.001	10 miles 0.000		
DE l	1051 0.018	0.004	0.001		·
rojected duration of re	lesse: 1.0 hours				
A General E	mergency Requires	a Protectiv	e Action Reco	mmendation	
EVACUATE	그는 집에서 동안을 하는 것이다.				
14 (M. 16)	PLACE ZONE(S): 2				
		5. TE & D			
	DOWNWIND SECTOR			i sakar Sakar	
All Remaining Zo	nes Go Indoors And Monit	of eas radio 3	Lauon		
Based on a Dose R Site Boundary (I M	ate Projection of > 3 mren file) for 15 minutes or longe	a/hr (Immersion er the Emergenc	Whole Body Noble v Classification Init	e Gas Gamma) at th	e 1
(ALERT) has been	n met.			want constants was	
	n met .				
(ALERT) has been				12/17/2613 3:24:44 PM	
(ALERT) has been	n met. Name				
(ALERT) has been PERFORMED BY: REVIEWED BY:				12/17/2613 3:24:44 PM	
(ALERT) has been PERFORMED BY: REVIEWED BY:	Name			12/17/2013 3:24:44 PM Date/Time	

12/17/2013 3:24:28 PM



CALC. NO. STPNOC013-CALC-002

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REV.1

DRILL STAMPEDE User Supplied Information DRILL Revision 7.033 9/28/2011 DRILL

Date/Time: 12/18/2013 07:54. The Name: SteamLine Site Alert Comments:

User Supplied Information

Meteorological Data Inguis: Ground level wind velocity: 13.2 mi/hr Ground level wind fram: 138 degrees Uker-selected Stability Class Stability Class: "D - Neutral"

Manitared S/G Tabe Rapture Release: Steam Activity: 4502+000 aC3/cc Steam Flow Rate: 1.050 mB/hr

 Reactor Shutdown Date/Time:
 12/18/2013 06:54

 Release Start Date/Time:
 12/18/2013 07:54

 Edimated Release Duration:
 1.90 hours

Nuclide Matters: Noble Cas + Lookue Lookue as percent of noble gas: 0.2%

Calculated NOBLE CAS release rate: 1.1989-087 oCUser

NOBL	EGAS	IODI	NE	PARTIC	ulate
Nuclide	aCi/sec	Naclide	BCE SEC	Nachde	oCi/sec
KF-SALL	1.142+004	I-131:	3.052+003	Co-134;	0.008+000
Kr-85:	3.438+005	1-132:	3225+008	Cs-137 :	0.008+000
Kr-85M	5.798+004	1-133:	4.888+003	CePr-144:	0.00E+000
Kr-87:	255E+004	I-134:	4228+002	Ce-141:	0.00E+000
Kr-88:	9.298+004	I-135 :	1.238+004	La-140:	0.0JE+000
Kr-89;	4.25E-003			Lin-99:	0.0XE+000
Xe-131Mf	1.202+005			Rn/Rh-106	0.00E+000
Xr-133:	1.038+007			RD-IRI:	0.00E+000
Xx-13334	1.87E+005			Sr/Y-99:	0.00E+000
Xr-135:	3.188+005			Se-59:	0.00E+000
Xe-135M	1.238+003			Te-132:	0.60E+000
X2-137:	1.27E-001			Zr-95:	0.00E+000
Xr-138:	1.362+003				

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CALC. NO. STPNOC013-CALC-002

REV.1 1.1 x PAGE NO. 36 of 49

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	an an an Andrewski († 1937) 1970 - Nicker Stater, filozof († 1937)	• • • • • • • • • • • • • • • • • • • •	
DRIL	STAMPEDE Results Restant 7.03.3 \$/28/2011	: Information Pagelat2	DRILL
Bale/Time: 12/15/20130 Comments:	754 Iker Name: SwamLu	ie Site Alert	
	Simoe Information	ora www.	And the second second second second
	CANAL STREET, SALES	and the second	CHINO DEPL
Distance	Plumo Travel Amo	CHLQ Value	(sector)
(solim)	(hours:minules)	('m')	
05	0.02	2.6865-005	2.436E-005 9.110E-005
1.0	0.05	17558-005	3.151E-006
20	0.09 0.23	10015-005	7.3738-007
5.0 7.5	034	5704E-007	1845E-007
10.0	645	3.8518-007	2441E-007
20.0	131	15418-007	9.100E-00E
e e e e e		1047	Dose Rates
	Measurable Bose Rates	- 2011년 1947년 - 1977년 - 1977년 - 1977년 - 1977년	the second second second second
÷.,	Inemersian Whole Body	TELE	ledine CDE
Distance	stople Ear farmers	external + inversal (remfar)	Thyreid frem.hr.)
(miles)	(rom.ār)		
0.5	0.011	0.019	<u>0135</u>
1.0	0.0042	0.007	0.050 0.017
20	0.682	0.008 0.001	0.077
50	0.080 0.080	0.000	0.002
75	0.050	0.000	0.001
100 200	0.000	0.009	0.000
2000	S. LONG		
	Manurable Dozes	124	Dases
	Ale maintaine areas		Indine CDE
Distance	Immerzian Whole Nody	erierail † internal	Thursday
[miles]	noble gus persons (ram)	(ram)	(nem)
	요즘은 이 사람들은 물질 것이 있는 것을 많을 것이다.	0019	0.135
0.5	0.011 0.604	0.007	0.050
10 20	0.082	0.005	8.017
20 50	fineo	0.001	6.004
75	0.000	0,000	9.002
100	0.000	0.000	6.003
200	0000	0.000	6000

12/18/2013 7:54/42 AM



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REV.1

DR	ILL	STAMPEDE R Revision 7.0.3.3 9/28/2		nformation Page2of2	DRILL
e stationerse			ous Comul	ted because and the second	
		RESU	JTS		
Method of Projection		Wind Velocity:	13.2 mi/hr	Release Rates	1.19E+007 uCidsec
,	STAMPEDE	Wind Directions	150		
Offsite 2	Done Projection (rem	ðr			
	I mile	1 miles	5 miles	10 miles	
TEDE	0.007	0.003	0.001	0.000	
CDE	0.050	0.017	0.004	0.001	
Projected	duration of release:	0 hours			
4 G.	maral Finara	nev Requires a I	ratacti	ve Action Recomm	aendation

EVACUATE ZONE(S): 1

SHELTER IN PLACE ZONE(S): 2

AFFECTED DOWNWIND SECTORS: -R, A, B -- --

All Remaining Zones Go Indoors And Munitor EAS Radio Station

Based on a Dose Rate Projection of \geq 3 mem/hr (Immersion Whole Body Noble Gas Gamma) at the Site Boundary (1 Mile) for 15 minutes or longer the Emergency Classification Initiating Condition RA1 (ALERT) has been met.

PERFORMED BY:

12/18/2013 7:55:14 AM Date/Time

Date Time

REVIEWED BY:

-Rad Manager/Radiological Director

Name

12/18/2013 7:54:42 AM



CALC, NO. STPNOC013-CALC-002

REV. 1 PAGE NO. 38 of 49



ie A

DRILL STAMPEDE User Supplied Information DRILL Nevision 7013 9/28/2011 Date/Time: 12/17/2013 15/25 Ther Name: Unit Vent Sile Area Comments:

User Supplied Information

5.00

12.1

Méřeskological Data Inpuis: Groundlevel windvelosity: 13.2 mi/hr Groundlevel windrenn: 150 degrees User-selectod/Stabiliy Class Stability Class: "D- Newical"

Manifered Unit Vent Release: Unit Vent Release Rale entryid: 2.505+007 mCl/nec

Reactar Shuidova Dale/Time: 12/17/003 14-25 Release Start Date/Time: 12/17/2013 15-25 Estimated Release Datation: 1.00 hours

Nachde Mathere:

Calculated NOBLE GAS release rate: 1.190-007 aCives

Cap Inventory

NOHL	elad	KODE	12	PARTH	THATE
Muchde	nCi/sec	Nuclide	BCUSEL	Nuclisie	wCi/sec
Kr-SIM	2.52E+605	I-131:	3.128+004	Cr-134:	1.05B+601
Kr-85.	1056+005	1-132:	3.1284004	C5-137	82/E+000
KT-85M	7.068+005	1-133:	60-18+004	Ce/Pr-144:	2.10E-003
Kr-97:	9.0384005	I-13 4:	3.020+004	Ce-141:	2.BIE 003
Kr-88:	1.738+005	1-135:	5.128+004	Lz-140:	5.31E-005
K5-89:	3.14E+000			Ma 99:	3.43B+CO0
Xe 13114	3.125+004		an a	Ra/Rb-165	824E-004
Xe-131:	6.228+005			R4-163:	2.508-003
X2-13314	191E=005			St/Y-90:	9.10E-004
Xe-135:	1.452##006		the state of the second se	Sr-89:	1.82E-002
Xe-13534	8.152+004			Te-132:	1.35E+000
Xz-J37:	9.746+001			21-35.	3.13E-003
X:e-138.	2676+005	· · ·		-	

12/17/2013 3:25:33 PM



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PAGE NO. 39 of 49

DRILL	STAMPEDE Resul Revision 7.03.3 9/28/2011	Page lof 2	DRILI
Dz/e/Time: 12/17/2013 15:25 Comments:	Uter Name: Unit Ve		
	Fluence Inform	SINKI COMPANY STATE	an a
Distance	Plane Travel Time	CHEQ Value	CHIQ DIPL.
(milos)	(hours:minutes)	(sasc/m*)	(110/11)
0.5	0:02	2,6362-005	2.436E-005
10	0:05	1.6828-005	9.110E-006
2.0	0:09	3.755E-005	3.151B-005
5.0	0:23	1.004E-005	7.373B-007
7.5	0:34	5.704B-007	3.845E-007
10.0	0:45	3.8SIE-007	2.441B-007
20.0	131	1.541E-007	9.1098-008
	Measurable Dose Rates	PAC	Dose Rates
	Immerzian Whole Body	THE	Iodian CBR
Distance	noble gas games	orternal + internal	Thyroid
(miles)	(гон.б.т)	(rom/kr)	(rem/hr)
0.5	0.033	0.160	1.364
1.0	0.0333	0.060	0.510
2.0	0.012	0.021	0.176
5.0	0.003	0.005	8.041
75	0.002	0.003	0.021
10.0	0.001	0.003	0.014
20.0	0.000	0.001	0.005
	Mersuralla Boses	PAC	Dozes
an a' .		TEDE	ladias CDE
Bittance	Immersion Whole Hody	STITUTAL + Sutarasi	Thyraid
(miløs)	noble yr: yrmen (ram)	(mm)	(rem)
0.5	0.088	0.169	1.364
1.0	0.083	0.050	0.510
20	0.012	0.021	0.175
5.0	0.003	0.005	0.041
75	0.092	0.003	0.021
10.0	100.0	0.002	0.014
20.0	0.000	0.001	0.005

12/17/2013 3:25:21 PM

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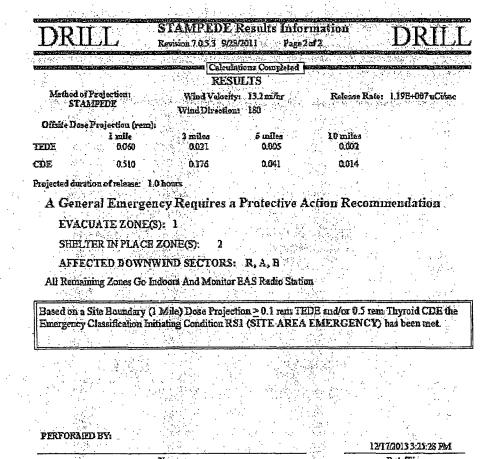
and the second



CALC. NO. STPNOC013-CALC-002

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REVIEWED BY	Mame		Date Time	
KLWMAN MARINE			· · · · · ·	
Ra	d&Inuager/Kadiological Director		Date Time	-
		· · ·		

12/17/2013 3:25:21 PM



CALC, NO. STPNOC013-CALC-002

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DRILL STAMPEDE User Supplied Information DRILI Revision 70.3.3 9/28/2011 DRILI

Date/Time: 12/17/2013 15:28 Comments: ther Name: Steam Line Site Area

Ler Supplied Information and

Meteorological Data Inputs: Granul level wind velocity: 13.2 mi/hr Granul level wind fram: 130 degrees User-selected Stahility Class Stability Class: "D - Neufral"

Monikared S/GTube Regime Release: Steam Actinity: 4.50E+001 uCk/cr Steam Flow Rate: 1.650 mB/kr Reactor Shutdown Date/Tione: 12/17/2013 14/28

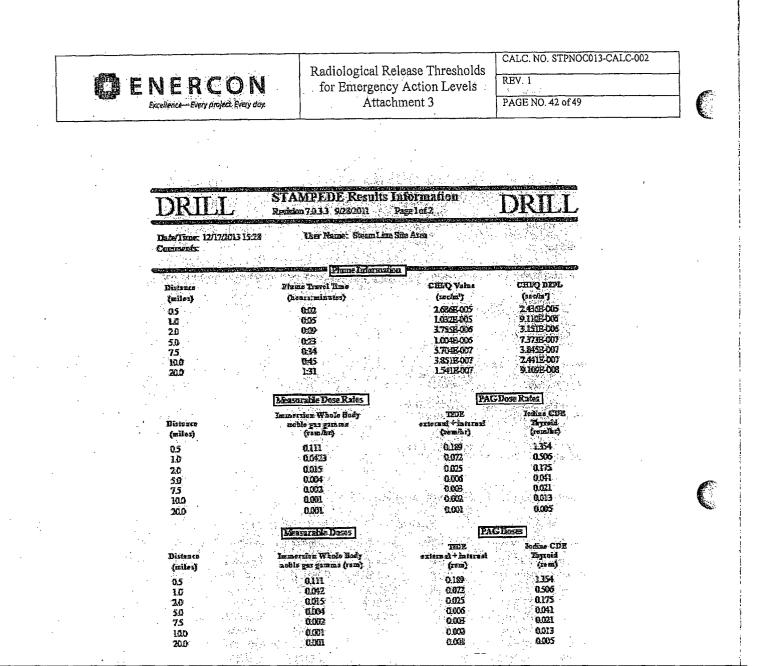
Release Start Date/Time: 12/17/2013 15:28 Estimated Release Buration: 1.00 hepra

Nuclide Minime: Noble Gas + Iodine Indiae as pertent of noble gas: 8,296

Calculated NOBLE CAS release rate: 1.20E-088 aCilsec

NOHLE GAS		RODI	NE	PARTIC	ULATE
Nachda	uCi/sec	Nachde	BC äsec	Nachde	oCi/sec
KF-SILL	1.142+005	3-131:	3.078+004	Cs-134:	0.00E+600
Kr-85:	3.458+006	1-132 :	3235+004	Cs-137:	0.008+000
Kr-S5M	5.81B+005	1-133 :	490E+004	Ce/Fr-144:	0.00E+000
Kr-87:	2.55E+005	I-134 :	422E+0B	Ce-141:	0.002+000
Ki-88:	991E+005	I-135 :	1248+005	La-140:	8.00E+000
Kr-29:	3.928-002			Ma-99;	0.0000+000
Xe-13114	1.278+006			Ra/Rh-105:	0.0XE+000
X2-133;	1.082+008			En-103:	0.00E+000
Xe-1334f	1.388+006			Sr/¥-90:	0.00E+000
Xr-135:	3.198+005			Sr-89:	0.00E+000
Xe-135M	1218+004			Te-132:	0.00E+900
Xz-137:	1.198+000			Zz-95;	0.00000000
Xe-138:	13450+004				

12/17/2013 3:29:03 PM



12/17/2013 3:28:53 PM

1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 -1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 -1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 -



CALC. NO. STPNOC013-CALC-002

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REV.1

DR	ILL	STAMPEDE R Revision 7.0.3.3 9/28/2		e2of2	DRILL		
RESULTS							
Method of Projection: STAMPEDE		Wind Velority:	13.2 mi/hr	Release Rate:	1.20E+008 aCi/sec		
		Wind Direction:	180				
Offsite Dase Projection (rem):							
	l mile	2 miles	5 miles	10 miles			
TEDE	0.072	0.025	0.006	0.002			
CDE	0,505	0.175	0.041	0.013			
Projected duration of release: 1.0 hours							

A General Emergency Requires a Protective Action Recommendation

EVACUATE ZONE(S): 1

SHELTER IN PLACE ZONE(S): 2

AFFECTED DOWNWIND SECTORS: R, A, B

All Remaining Zones Go Indoors And Monitor EAS Radio Station

Based on a Site Boundary (1 Mile) Dose Projection > 0.1 rem TEDE and/or 0.5 rem Thyroid CDE the Emergency Classification Initiating Condition RSI (SITE AREA EMERGENCY) has been met.

PERFORMED BY:

12/17/2013 3:29:00 PM Date/Time

REVIEWED BY:

Rad Manager Radiological Director

Name

------Date/Thue

12/17/2013 3:28:53 PM

Radiological Release Thresholds for Emergency Action Levels Attachment 3 CALC. NO. STPNOC013-CALC-002

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REV, I

3.436+001.

825B-003

2.502-002

9.10E 003

1.82E-001

1358+001

3.13E-002

Mo-99:

Rn/Rb-106:

RI-IIIS:

Scry-90:

Sr-89:

Zr-95:

Te-132:

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DRILL SI	AMPEDE Us Revision 703	er Supplied L 3 928/2011	iformation	DRILL
Bals/Fine: 12A7/2013 1526 Comments:	Tert Nam	e: Unit Veni General		
				an shakarar
	user S	upplied Information		
Meteorological Data Inputs: Ground Level wind velocity:	13.2 milar	a ga	and the second	· · ·
Grandlevel viadirena:	130 degrees		8 (L. 1997) 19 - Alexandre Alexandr 19 - Alexandre A	· · · ·
User-selected Stability Class	S	A 7		14 - 14 - 14 - 14 - 14 - 14 - 14 - 14 -
Statistity Class:	"D - Neutral"			1
Monitored Unit Vent Release.				
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Radiological Release Thresholds for Emergency Action Levels Attachment 3

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Radiological Release Thresholds for Emergency Action Levels Attachment 3

CALC. NO. STPNOC013-CALC-002 REV.1

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DRILL STAMPEDE User Supplied Information DRILL Revision 7.0.3.3 9/28/2011 DRILL

Date/Time: 12/17/2013 15:30 Comments: ther Manne: StremLine General

User Supplied Information

Meteorological Data Ingests: Ground lovel wind relocity: 13.2 mi/hr Ground lovel wind from: 180 degrees User-selected Stability Class Stability Class: "D - Neutral"

Minilared S/GTube Rupture Release: Steam Activity: 4.505+062 uCi&c Steam Flow Rale: 1.050 mills/hr

 Reactor Shutdown Date/Time:
 12/17/2013 14:38

 Release Start Date/Time:
 12/17/2013 15:30

 Edimated Release Maration:
 1.90 hears

 Nuclide Minister:
 Noble Gas + Iodiae

 Lotine as percent of noble gas:
 8.244

Calculated NOHLE GAS release rate: 1.20E-009 aCi/sec

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	K)-8514	5.8215+006	I-133:	4908+005	Ce/Pr-144:	0.038+000
	Kr-87:	2.565+006	I-134 :	42454004	Ce-141:	0.00E+000
	Kr-28:	9:92E+006	I-135 :	1.248+006	La-140:	0.008+000
	Kr-89:	4.135-001			Map-99;	0.008+000
	Xe-131M	1.27E+007			En/Rh-166	0.00B+000
	Xz-133:	1.088+009			En-163:	0.00E+000
-	Xr-13314	1.38E+007		·····	St/¥-99;	0.00E+000
	Xe-135:	3.192+007			Sr-89:	0.00E+000
	X1-135M	1.238+005			Te-132:	0.00E+000
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Radiological Release Thresholds for Emergency Action Levels Attachment 3

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Radiological Release Thresholds for Emergency Action Levels Attachment 3

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The particulate channel is used as part of the Reactor Coolant Pressure Boundary (RCPB) leakage detection system. The sensitivity and response time of this part of the leakage detection system, which is used for monitoring unidentified leakage to the Containment, are sufficient to detect an increase in leakage rate of the equivalent of one gal/min within one hour. Elements of this monitor, including the indicator mounted in the RMS CR cabinet, are designed and qualified to remain functional following a Safe Shutdown Earthquake (SSE), in compliance with RG 1.45. Further information on the RCPB leakage detection system is presented in Section 5.2.5.

11.5.2.3.3 <u>Unit Vent Monitor</u>: The unit vent monitor samples the plant vent stack prior to discharge to the environment and monitor for particulates, iodine, and noble gases.

The unit vent particulate and iodine monitor draws representative air samples from the plant vent stack via isokinetic nozzles in the stack, and directs them through a moving filter paper monitored by a shielded beta-sensitive scintillation detector. The sample stream then passes through a charcoal collector, where collected iodine is monitored by a shielded gamma-sensitive scintillation detector. The sample is then returned to the vent stack.

A separate wide-range gas monitor is provided for the unit vent. The monitor has two isokinetic nozzles for sampling during both normal and accident conditions. The stack samples pass first through a sample conditioning unit which filters particulates and iodine and may be used to take grab samples. The samples then pass through the shielded detector assembly, which uses three detectors to cover the complete range required. The low range detector uses a beta-sensitive plastic scintillator-photomultiplier (PM) tube. The mid-range and high-range detectors use cadmium telluride (CdTe), chlorine-doped, solid-state sensors. This wide-range gas monitor satisfies the requirements of NUREG-0737, Item II.F.1 for provisions for sampling plant effluents for iodines and particulates and for noble gas effluents from the plant vent.

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11.5.2.3.4 <u>Control Room Electrical Auxiliary Building Ventilation Monitors</u>: The CR/EAB ventilation monitors are Class 1E monitors which continuously assess the intake air to the CR for indication of abnormal airborne radioactivity concentration. Each monitor assembly is powered from a separate electrical power source. In the event of high radiation CR emergency ventilation operation is initiated (Section 7.3.2). Failure of a monitor is alarmed in the CR.

Each monitor assembly is comprised of a recirculation pump, beta-sensitive scintillation detector, four-pi lead shielding, check source, stainless steel sample gas receiving chamber, and associated electronics.

11.5.2.3.5 <u>Condenser Vacuum Pump Monitor</u>: Gaseous samples are drawn through an offline system by a pump from the discharge of the vacuum pump-exhaust header of the condenser. This channel monitors the gaseous sample for radioactivity which would be indicative of an SG tube leak, allowing reactor coolant to enter the secondary side fluid; this monitor complements the SGBD monitors in indication of a SG tube leak. The gaseous radioactivity levels are monitored by a single detector in a manner similar to the unit vent wide range gas monitor.

11.5.2.3.6 <u>Spent Fuel Pool Exhaust Monitors</u>: The SFPE monitors are Class 1E and are identical to the CR/EAB ventilation monitors described in Section 11.5.2.3.4 except that they sample the exhaust from the FHB. In the event of high radiation the monitors initiate emergency operation

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11.5.2.5.1 <u>Gaseous Waste Processing System Inlet Monitor</u>: The GWPS inlet monitor employs a gamma (Nal crystal) scintillator/photomultiplier tube combination to measure the radioactivity level of the waste gases entering the GWPS. The monitor is used in conjunction with the GWPS discharge monitor to measure overall effectiveness of the GWPS.

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11.5.2.5.2 <u>GWPS Discharge Monitor</u>: This monitor is similar to the GWPS inlet monitor and is installed upstream of the GWPS discharge valve. Upon detection of high radioactivity or monitor failure, the GWPS discharge valve, FV-4671, is automatically closed.

11.5.2.5.3 <u>Main Steam Line Monitors</u>: Each MS line is monitored by an ATL monitor consisting of a Geiger Mueller (GM) tube detector and an ion chamber detector with overlapping ranges. The detectors are shielded by 3 in. of lead.

The monitors are designed to monitor gross gamma activity in the steam line and provide a basis for determining possible atmospheric releases from the MS power-operated relief valve (PORV), SG safety valves, and/or auxiliary feedwater pump turbine.

The monitors provide a dose rate range equivalent to 10^{-1} to $10^{3} \,\mu\text{Ci/cm}^{3}$ xenon-133. Based upon core inventory, the ratio of xenon-133 to other nuclides in the fuel can be determined. In order to obtain the above concentrations of xenon-133 in the main steam line, a large primary-to-secondary leak must be present coincident with a large amount of fuel failure. The presence of xenon-133 indicates other radioactive isotopes are present.

Using the relative ratios of isotopes present in the MS line, a computer model for determination of dose rates from these isotopes, detector response curves, the thickness of the MS line, and the geometry of the MS line relative to the detector, the dose rate equivalent to MS line concentration is obtained. The quantity of radioactive effluents released is obtained by multiplying the xenon-133 equivalent MS line concentrations by the isotope ratio times the steam release rate.

These detectors are safety-related Class 1E and meet the requirements of RG 1.97 and NUREG-0737.

11.5.2.5.4 <u>Steam Generator Blowdown Monitors</u>: These monitors are identical to the MS line monitors and are adjacent to the SG blowdown lines in the Isolation Valve Cubicle (IVC).

The monitors are used as an aid in determining the source of SG blowdown radioactivity due to SG tube rupture or a large primary-to-secondary leak.

These detectors are safety-related Class IE and meet the requirements of RG-1.97.-

11.5.2.3.5 <u>Main Steam Line High Energy Gamma (N-16) Monitors</u>: Each main steam line is monitored by an ATL NaI solntillation detector. These detectors were installed to monitor the status of steam generator primary to secondary tube leaks and to provide a diagnostic tool for all individuals concerned with steam generator condition. These detectors are designed to detect high energy gamma activity in the 6 to 7.2 MEV energy range. High energy gamma activity in the main steam lines indicates the presence of N-16. The level of N-16 in the main steam lines is used to

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Shielding for the SFP is adequate to protect plant personnel from exposure to radiation in excess of published guideline values as stated in Section 12.1. A minimum depth of approximately 13 ft of water over the top of an array of 193 (full core) assemblies with 42 hours of decay is required to limit radiation from the assemblies to 2.5 mR/hr. or less.

The FHB Ventilation Exhaust System is designed to limit the offsite dose in the event of a significant release of radioactivity from the fuel, as discussed in Sections 12.3.3, 15.7.4, and 9.4.2. However, no credit for the FHB Ventilation Exhaust System is taken in the LOCA and Fuel Handling accident in Chapter 15.

The FHB is designed to prevent missiles from contacting the fuel. A more detailed discussion on missile protection is given in Section 3.5.

In addition, space is provided for storage of fuel during refueling inside the RCB for 64 fuel assemblies in four 4 x 4 modules having 16-in. center-to-center spacing (Figure 9.1.2-1A). These modules are firmly bolted in the floor.

9.1.2.2 <u>Facilities Description</u>. The FHB abuts the south side of the RCB and is adjacent to the west side of the MEAB of each unit. The locations of the two FHBs are shown in the station plot plan on Figure 1.2-3. For general arrangement drawings of the spent fuel storage facilities, refer to Figures 1.2-39 through 1.2-48 as listed in Table 1.2-1.

The spent fuel storage facilities are designed for the underwater storage of spent fuel assemblies and control rods after their removal from the reactor vessel. The spent fuel is transferred to the FHB and handled and stored in the spent fuel pool underwater. The fuel is stored to permit some decay, then transferred offsite. For a detailed discussion of spent fuel handling, see Section 9.1.4.

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The SFP is located in the northwest quadrant of each FHB. The floor of the pool is at El. 21 ft-11 in., with normal water level at El. 66 ft-6 inches. The top of a fuel assembly in a storage rack does not extend above the top of the storage rack which is El. 39 ft-10 in. maximum. The fuel assemblies are loaded into the spent fuel racks through the top and are stored vertically.

9.1.2.3 <u>Safety Evaluation</u>. Units 1 and 2 of the STPEGS are each provided with separate and independent fuel handling facilities. Flood protection of each FHB is discussed in Section 3.4.1. A detailed discussion of missile protection is provided in Section 3.5.

The applicable design codes and the various external loads and forces considered in the design of the FHB are discussed in Section 3.8.4. Details of the seismic design and testing are presented in Section 3.7.

Design of this storage facility in accordance with GDC 62 and RG 1.13 ensures a safe condition under normal and postulated accident conditions. The K_{eff} of the spent fuel storage racks is maintained less than or equal to 1.00, even if unborated water is used to fill the spent fuel storage pool, by both the designs of the fuel assemblies and the storage rack and the use of administrative procedures to control the placement of burned and fresh fuel and control rods.

Under accident conditions, the K_{eff} is maintained well below 0.95 assuming 2200 ppm borated water. The boron concentration of the water in the spent fuel pool is maintained at or above the minimum

Revision 16

NRC ORDER EA-12-051 (SFP LEVELS)

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REQUIREMENTS FOR RELIABLE SPENT FUEL POOL LEVEL INSTRUMENTATION AT OPERATING REACTOR SITES AND CONSTRUCTION PERMIT HOLDERS

All licensees identified in Attachment 1 to this Order shall have a reliable indication of the water level in associated spent fuel storage pools capable of supporting identification of the following pool water level conditions by trained personnel: (1) level that is adequate to support operation of the normal fuel pool cooling system, (2) level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck, and (3) level where fuel remains covered and actions to implement make-up water addition should no longer be deferred.

- 1. The spent fuel pool level instrumentation shall include the following design features:
 - 1.1 Instruments: The instrumentation shall consist of a permanent, fixed primary instrument channel and a backup instrument channel. The backup instrument channel may be fixed or portable. Portable instruments shall have capabilities that enhance the ability of trained personnel to monitor spent fuel pool water level under conditions that restrict direct personnel access to the pool, such as partial structural damage, high radiation levels, or heat and humidity from a boiling pool.
 - 1.2 Arrangement: The spent fuel pool level instrument channels shall be arranged in a manner that provides reasonable protection of the level indication function against missiles that may result from damage to the structure over the spent fuel pool. This protection may be provided by locating the primary instrument channel and fixed portions of the backup instrument channel, if applicable, to maintain instrument channel separation within the spent fuel pool area, and to utilize inherent shielding from missiles provided by existing recesses and corners in the spent fuel pool structure.
 - 1.3 Mounting: Installed instrument channel equipment within the spent fuel pool shall be mounted to retain its design configuration during and following the maximum seismic ground motion considered in the design of the spent fuel pool structure.
 - 1.4 Qualification: The primary and backup instrument channels shall be reliable at temperature, humidity, and radiation levels consistent with the spent fuel pool water at saturation conditions for an extended period. This reliability shall be established through use of an augmented quality assurance process (e.g., a process similar to that applied to the site fire protection program).
 - 1.5 Independence: The primary instrument channel shall be independent of the backup instrument channel.
 - 1.6 Power supplies: Permanently installed instrumentation channels shall each be powered by a separate power supply. Permanently installed and portable instrumentation channels shall provide for power connections from sources independent of the plant ac and dc power distribution systems, such as portable generators or replaceable batteries. Onsite generators used as an alternate power source and replaceable batteries used for instrument channel power shall have sufficient capacity to maintain the level indication function until offsite

----Attachment 2

resource availability is reasonably assured.

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- 1.7 Accuracy: The instrument channels shall maintain their designed accuracy following a power interruption or change in power source without recalibration.
- 1.8 Testing: The instrument channel design shall provide for routine testing and calibration.
- 1.9 Display: Trained personnel shall be able to monitor the spent fuel pool water level from the control room, alternate shutdown panel, or other appropriate and accessible location. The display shall provide on-demand or continuous indication of spent fuel pool water level.
- 2. The spent fuel pool instrumentation shall be maintained available and reliable through appropriate development and implementation of the following programs:
 - 2.1 Training: Personnel shall be trained in the use and the provision of alternate power to the primary and backup instrument channels.
 - 2.2 Procedures: Procedures shall be established and maintained for the testing, calibration, and use of the primary and backup spent fuel pool instrument channels.
 - 2.3 Testing and Calibration: Processes shall be established and maintained for scheduling and implementing necessary testing and calibration of the primary and backup spent fuel pool level instrument channels to maintain the instrument channels at the design accuracy.

The three critical levels that must be monitored in a spent fuel pool are discussed below. It should be noted that continuous indication from a single instrument over the entire span from level 1 to level 3 is not required but could be used. If more than one instrument is used to monitor the entire span, that set of instruments constitutes a single channel satisfying either the primary or backup instrument channel requirement (refer to Figure 1 below).

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A visual representation of monitoring levels 1, 2 and 3 and the associated requirements for monitoring between the points are presented in Figure 1. The minimum requirements apply to the separation distance between level indications and support development of appropriate response procedures. These requirements are separate from the instrument channel design accuracy discussed in section 3, which apply to either discrete or to continuous instruments.

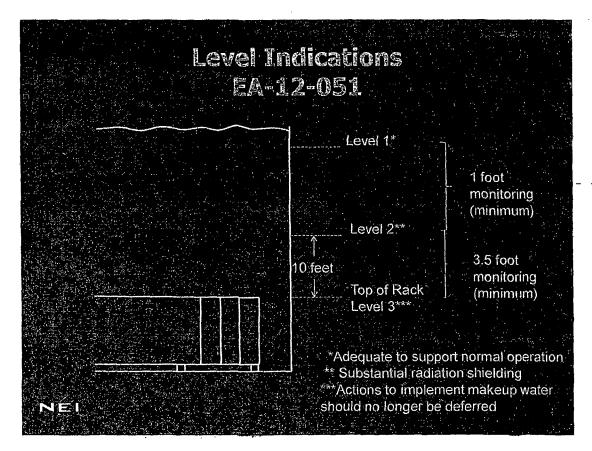


Figure 1

2.3.1. Level 1 – level that is adequate to support operation of the normal fuel pool cooling system

A typical fuel pool cooling system design includes a combination of weirs and/or vacuum breakers that prevent siphoning of the pool water level, below a minimum level, in the event of a piping rupture that can affect the SFP level. Level 1 represents the HIGHER of the following two points:

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- The level at which reliable suction loss occurs due to uncovering of the coolant inlet pipe, weir or vacuum breaker (depending on the design), or
- The level at which the water height, assuming saturated conditions, above the centerline of the cooling pump suction provides the required net positive suction head specified by the pump manufacturer or engineering analysis.

This level will vary from plant to plant and the instrument designer will need to consult plant-specific design information to determine the actual point that supports adequate cooling system performance.

2.3.2. Level 2 – level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck

Level 2 represents the range of water level where any necessary operations in the vicinity of the spent fuel pool can be completed without significant dose consequences from direct gamma radiation from the stored spent fuel. Level 2 is based on either of the following:

- 10 feet (+/- 1 foot) above the highest point of any fuel rack seated in the spent fuel pools, or
- a designated level that provides adequate radiation shielding to maintain personnel radiological dose levels within acceptable limits while performing local operations in the vicinity of the pool. This level shall be based on either plant-specific or appropriate generic shielding calculations, considering the emergency conditions that may apply at the time and the scope of necessary local operations, including installation of portable SFP instrument channel components. Additional guidance can be found in EPA-400 (Reference 4), USNRC Regulatory Guide 1.13 (Reference 5) and ANSI/ANS-57.2-1983 (Reference 6).

Designation of this level should not be interpreted to imply that actions to initiate water make-up must be delayed until SFP water levels have reached or are lower than this point.

2.3.3. Level 3 – level where fuel remains covered and actions to implement make-up water addition should no longer be deferred.

Level 3 corresponds nominally (i.e., +/- 1 foot) to the highest point of any fuel rack seated in the spent fuel pool. Level 3 is defined in this manner to provide the maximum range of information to operators, decision makers and emergency response personnel. Designation of this level should not be interpreted to imply that actions to initiate water make-up must or should be delayed until this level is reached.



South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

February 28, 2013 NOC-AE-13002959 10 CFR 50.4 10 CFR 2.202

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

South Texas Project Units 1&2

Docket Nos. STN 50-498, STN 50-499 Overall Integrated Plan Regarding Commission Order Modifying Licenses with Regard to

Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)

References:

- 1. Letter, Eric Leeds to E. D. Halpin, "Issuance of Order to Modify Licenses with Regard to Requirements for Reliable Spent Fuel Pool Instrumentation," March 12, 2012 (EA-12-051)
- NRC Interim Staff Guidance JLD-ISG-2012-03, Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation, Revision 0, August 29, 2012
- Letter D. W. Rencurrel to NRC, "Initial Status Report in Response to March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)", dated October 24, 2012.

On March 12, 2012, the Nuclear Regulatory Commission (NRC) issued an order (Reference 1) to STP Nuclear Operating Company (STPNOC). Reference 1 directs STP Nuclear Operating Company to provide a reliable indication of the water level in associated spent fuel storage pools. Specific requirements are outlined in Attachment 2 of Reference 1.

Reference 1 required submission of an overall integrated plan, including how compliance will be achieved. The final interim staff guidance (Reference 2) was issued August 29, 2012 providing licensees an acceptable approach for complying with the order. The purpose of this letter is to provide the overall integrated plan, including a description of how compliance will be achieved pursuant to Section IV, Condition C.1.a, of Reference 1 in accordance with the guidance in Attachment 2 to Reference 1 and the guidance in Reference 2. See the Enclosure for STPNOC's response to the requested information.

There are no new commitments in this letter.

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If there are any questions regarding this letter, please contact Robyn Savage at (361) 972-7438.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on:

Dennis L. Koehl President and CEO/CNO

Enclosure:

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South Texas Project (STP) Overall Integrated Plan for Implementation of Unit 1 & Unit 2 Spent Fuel Pool Level Instrumentation to Meet NRC Order EA-12-051

NOC-AE-13002959 Page 3 of 3

cc: (paper copy)

Regional Administrator, Region IV U. S. Nuclear Regulatory Commission 1600 East Lamar Boulevard Arlington, TX 76011-4511

Balwant K. Singal Senior Project Manager U.S. Nuclear Regulatory Commission One White Flint North (MS 8 B1) 11555 Rockville Pike Rockville, MD 20852

NRC Resident Inspector U. S. Nuclear Regulatory Commission P. O. Box 289, Mail Code: MN116 Wadsworth, TX 77483

C. M. Canady City of Austin Electric Utility Department 721 Barton Springs Road Austin, TX 78704

U. S. Nuclear Regulatory Commission Director of Office of Nuclear Regulation One White Flint North (MS 13 H 16M) 11555 Rockville Pike Rockville, MD 20852-2738 (electronic copy)

A. H. Gutterman, Esquire Morgan, Lewis & Bocklus LLP

Balwant K. Singal U. S. Nuclear Regulatory Commission

John Ragan Chris O'Hara Jim von Suskil NRG South Texas LP

Kevin Pollo Richard Pena City Public Service

Peter Nemeth Crain Caton & James, P.C.

C. Mele City of Austin

Richard A. Ratliff Texas Department of State Health Services

Alice Rogers Texas Department of State Health Services

ENCLOSURE NOC-AE-13002959

South Texas Project (STP)

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Overall Integrated Plan for Implementation of Unit 1 & Unit 2 Spent Fuel Pool Level Instrumentation

to Meet NRC Order EA-12-051

Page 1 of 12

Revision: 00

1.0 OVERALL INTEGRATED PLAN INTRODUCTION

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This document provides the overall Integrated Plan (the "Plan") which the STP Nuclear Operating Company ("STPNOC") will implement for Units 1 and 2 to comply with the requirements of NRC Order EA-12-051, "Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation" (Ref.2), (the "ORDER"), NRC Interim Staff Guidance JLD-ISG-2012-003 [Rev.0] (Ref.3), (the "ISG"), and NEI Report 12-02 [Rev.1] ("NEI 12-02").

This Plan follows the format and provides all of the information on the STP 1 & 2 Integrated Plan that is required in NEI 12-02 [Rev.1] (Ref.1), Section A-2-2. Throughout this Plan, any reference to NEI 12-02 and the ISG will be based on the revisions above. Any reference to NEI 12-02 will include compliance to the clarifications and exceptions to NEI 12-02 required by the Interim Staff Guidance, Rev. 0.

In response to the NRC requirements, STPNOC will provide two channels of independent, permanently-installed, wide-range spent fuel pool level instrumentation ("SFPLI"), for the spent fuel pool ("SFP") of each unit. The spent fuel pool for each unit is independent and not interconnected in any way. For each SFP, the instrumentation provided for each channel will utilize the same technology, as permitted by the NEI 12-02 [Rev.1]. The spent fuel pool level instrumentation for each SFP on both the Primary and Backup Channels.

Both the Primary and Backup Channel/Instrument location and display of the SFP level will be independently mounted in each unit's Radwaste Control Room in the Mechanical Electrical Auxiliary Building (MEAB), which is an accessible post-event location. Other locations are still being considered.

Both the Primary and Backup Channel remote, non-safety related indication of the SFP level will also be provided in each unit's Control Room via input to the Plant Computer.

The instrumentation systems will not be safety-related, but will meet the requirements for augmented quality in accordance with NEI 12-02 [Rev.1] and the ISG as described below.

Since all of the potential suppliers have not completed development, the information in this Plan is based on the overall strategy and on information which, based on current information from potential suppliers, is thought to envelope the systems being developed for this application.

If there are any changes to the requirements in NRC JLD-ISG-2012-003 [Rev.0] and NEI 12-02 [Rev.1], relief from the requirements and schedule documented in this Plan may be required, in accordance with Section 12.0. Any required changes to this Plan will be defined in the periodic status reports submitted to the NRC.

2.0 APPLICABILITY:

This Plan applies to the spent fuel pools for South Texas Project Unit 1 and Unit 2.

Revision: 00

3.0 SCHEDULE:

The installation of reliable spent fuel pool level instrumentation for the spent fuel pool associated with Unit 1 is scheduled for completion prior to 10/28/2015, which is the end of the second refueling outage (1RE19) following submittal of this Plan.

The installation of reliable spent fuel pool level instrumentation for the spent fuel pool associated with Unit 2 is scheduled for completion prior to 4/29/2015, which is the end of the second refueling outage (2RE17) following submittal of this Plan.

Unit 1 Milestones are as follows:

- Design/Engineering September of 2014
- Purchase of instruments & equipment February of 2015
- Receipt of equipment June of 2015
- Unit 1 Installation & Functional Testing October of 2015

Unit 2 Milestones are as follows:

- Design/Engineering December of 2013
- Purchase of instruments & equipment August of 2014
- Receipt of equipment November of 2014
- Installation & Functional Testing April of 2015

• Consistent with the requirements of the ORDER and the guidance from NEI 12-02 [Rev.1], status reports will be generated in six (6) month intervals from the submittal of this Plan.

4.0 IDENTIFICATION OF SPENT FUEL POOL WATER LEVELS:

The STP Unit 1 and 2 spent fuel pools are essentially identical. The following SFP elevations are identified:

- The bottom of the pool is at Plant El. 21 ft. 11 in.
- The top of the SFP racks is approximately at Plant El. 39 ft. 10 in.
- The minimum Limiting Condition for Operation SFP level is Plant EI. 62 ft.
- Normal SFP water level is at Plant El. 66 ft. 6 in.
- Non-safety related level switch alarms are activated at Plant El. 67 ft. on high level and Plant El. 66 ft. on low level.
- The spent fuel pool deck is at Plant El. 68 ft.

The required key SFP water levels, per guidance of NEI 12-02 [Rev.1] and ISG (with clarifications and exceptions), are as follows:

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4.1 LEVEL 1: Level adequate to support operation of the normal fuel pool cooling system.

LEVEL 1 represents the HIGHER of either the level at which reliable suction loss to the cooling pump occurs, or, the required NPSH (Nominal Pump Suction Head) of the cooling pump.

Loss of reliable suction to SFP cooling pumps. For the purposes of this Plan, this level will conservatively be placed at Plant El. 64 ft. 2 in. This allows for just over 1 ft. of SFP water level above the top of the suction inlet flange (SFP Cooling Pump 14 in. suction line with centerline of suction inlet flange at Plant El. 62 ft. 6 in.) which will be sufficient for NPSH. (Ref. 9)

Therefore, considering the top of SFP fuel storage rack is at Plant El. 39 ft. 10 in., the indicated level on either the Primary or Backup Instrument Channel of greater than 24 ft. 4 in. above the top of the SFP fuel storage racks based on the design accuracy for the instrument channel per NEI 12-02 [Rev.1], for both the Primary and Backup Instrument Channels, is adequate for normal SFP cooling system operation.

LEVEL 1 = Plant El. 64 ft. 2 in or 24 ft. 4 in. water level above the top of the SFP fuel storage rack

4.2 LEVEL 2: Level adequate to provide substantial radiation shielding for a person standing on the SFP operating deck.

Indicated level on either the Primary or Backup Instrument Channel of greater than 10 ft. above the top of SFP stored fuel assemblies based on current guidance in NRC RG 1.13 [Rev.2] (Ref. 4) will achieve substantial radiation shielding. Requirements on substantial SFP radiation shielding is also given in ANSI/ANS-57.2-1983 (Ref. 5), and states that radiation shall not exceed 2.5 mRem/hr, but the minimum water depth to achieve this is undefined. NRC RG 1.13 [Rev.2] took exception to using dose rates as design input for minimum SFP water level, and instead defined the minimum level as 10 ft. above the stored fuel assemblies.

STPNOC elects to use the conservative approach of defining the top of the fuel rack as a basis for measurement. Therefore, indicated level on either the Primary or Backup Instrument Channel of greater than 10 ft. above the top of the SFP fuel storage rack, based on the design accuracy of the instrument channel per NEI 12-02 [Rev.1], for both the Primary and Backup Instrument Channels, ensures there is adequate water level to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck.

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Revision: 00

LEVEL 2 = Plant EI 49 ft. 10 in. or 10 ft. water level above the top of the SFP fuel storage rack.

4.3 LEVEL 3: Level where the fuel remains covered.

As stated above, STPNOC elects to use the conservative approach of defining the top of the fuel rack as a basis for measurement. The installation of the SFPLI sensor will be such that it will measure as close as possible to the top of the SFP fuel rack. Indicated level on either the Primary or Backup Instrument Channel of greater than $\frac{1}{2}$ ft. above the top of SFP fuel storage racks based upon the design accuracy of the instrument channel per NEI 12-02 [Rev.1], for both the Primary and Backup Instrument Channels, satisfies the NEI 12-02 [Rev.2] requirement of \pm 1 ft. from the top of the fuel rack. This monitoring level ensures there is adequate water level above the stored fuel seated in the SFP fuel storage rack.

LEVEL 3 = Plant El 40 ft. 4 in. or 6 in. water level above the top of the SFP fuel storage rack.

5.0 INSTRUMENTS:

Both the Primary and Backup Instrument Channels will utilize permanently-installed instruments. The design of the primary and backup instruments will be consistent with the requirements by NEI 12-02 [Rev.1], the ISG, and this Plan.

The current plan is for both channels to utilize Guided Wave Radar, which functions according to the principle of Time Domain Reflectometry (TDR). A generated pulse of electromagnetic energy travels down the probe. Upon reaching the liquid surface the pulse is reflected and based upon reflection times level is inferred. The measured range will be continuous from the high pool level elevation (67') to the top of the spent fuel racks (Ref. 8). However, STP is still evaluating other designs for this application. Any changes to the current plan will be reported in the 6 month update letter.

The supplier for the SFP instrumentation will be chosen by a competitive bidding process completed after submittal of this Plan, so the information in this Plan is based on the overall strategy and on available information from potential supplier's information on systems being developed for this application.

5.1 Primary (fixed) Instrument Channel

The Primary Instrument Channel level sensing components will be located in the northeast corner of the Spent Fuel Pool, as shown in Attachment 1. The primary instrument channel will provide continuous level indication over a range from Plant EI. 40 ft. 4 in. (LEVEL 3) to Plant EI. 67 ft. (SFP high level alarm) or a range of 26 ft. 8 in. In addition, the capability for discrete level indications at LEVEL1, LEVEL 2 and LEVEL 3, as described in Section 4.0, will be available.

STPEGS UFSAR

Shielding for the SFP is adequate to protect plant personnel from exposure to radiation in excess of published guideline values as stated in Section 12.1. A minimum depth of approximately 13 ft of water over the top of an array of 193 (full core) assemblies with 42 hours of decay is required to limit radiation from the assemblies to 2.5 mR/hr. or less.

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The FHB Ventilation Exhaust System is designed to limit the offsite dose in the event of a significant release of radioactivity from the fuel, as discussed in Sections 12.3.3, 15.7.4, and 9.4.2. However, no credit for the FHB Ventilation Exhaust System is taken in the LOCA and Fuel Handling accident in Chapter 15.

The FHB is designed to prevent missiles from contacting the fuel. A more detailed discussion on missile protection is given in Section 3.5.

In addition, space is provided for storage of fuel during refueling inside the RCB for 64 fuel assemblies in four 4 x 4 modules having 16-in. center-to-center spacing (Figure 9.1.2-1A). These modules are firmly bolted in the floor.

9.1.2.2 <u>Facilities Description</u>. The FHB abuts the south side of the RCB and is adjacent to the west side of the MEAB of each unit. The locations of the two FHBs are shown in the station plot plan on Figure 1.2-3. For general arrangement drawings of the spent fuel storage facilities, refer to Figures 1.2-39 through 1.2-48 as listed in Table 1.2-1.

The spent fuel storage facilities are designed for the underwater storage of spent fuel assemblies and control rods after their removal from the reactor vessel. The spent fuel is transferred to the FHB and handled and stored in the spent fuel pool underwater. The fuel is stored to permit some decay, then transferred offsite. For a detailed discussion of spent fuel handling, see Section 9.1.4.

The SFP is located in the northwest quadrant of each FHB. The floor of the pool is at El. 21 ft-11 in., with normal water level at El. 66 ft-6 inches. The top of a fuel assembly in a storage rack does not extend above the top of the storage rack which is El. 39 ft-10 in. maximum. The fuel assemblies are loaded into the spent fuel racks through the top and are stored vertically.

9.1.2.3 <u>Safety Evaluation</u>. Units 1 and 2 of the STPEGS are each provided with separate and independent fuel handling facilities. Flood protection of each FHB is discussed in Section 3.4.1. A detailed discussion of missile protection is provided in Section 3.5.

The applicable design codes and the various external loads and forces considered in the design of the FHB are discussed in Section 3.8.4. Details of the seismic design and testing are presented in Section 3.7.

Design of this storage facility in accordance with GDC 62 and RG 1.13 ensures a safe condition under normal and postulated accident conditions. The K_{eff} of the spent-fuel storage racks is maintained less than or equal to 1.00, even if unborated water is used to fill the spent fuel storage pool, by both the designs of the fuel assemblies and the storage rack and the use of administrative procedures to control the placement of burned and fresh fuel and control rods.

Under accident conditions, the K_{eff} is maintained well below 0.95 assuming 2200 ppm borated water. The boron concentration of the water in the spent fuel pool is maintained at or above the minimum

Revision 16

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	0POP04-RC-0003	Excessive RCS	Leakage	Rev. 18	Page 53 of 127
			Leanage		
Γ	Addendum 9	Basi		Ba	asis Page 5 of 77
	STEP	DESCRIPTION FOR 0PO	P04-RC-0003	STEP 3.0	
		For Any Of The Following			
		or RT8011 Particulate – Risin			
	Reactor Coo	lant Drain Tank Level – Rising	-		
	• Pressurizer I	Relief Tank Level – Rising			
	RCB Norma	l Sump Level – Rising			
		-			
	PURPOSE: To determ	ine if leakage is from RCS ar	nd not CVCS.	•	
		T8011, RCDT, PRT or RCE RCS and not CVCS which			vill confirm
		ends from RT8011, RCDT, F	<u>.</u>		
	·	<u>N</u> : Level indications located of		-	maita
Ĺ		ntrol room. Radiation Monite			mpater
	CONTROL/EQUIPMI	ENT: N/A			
	KNOWLEDGE: N/A				
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This Procedure is Applicable in Modes 1, 2, 3, and 4

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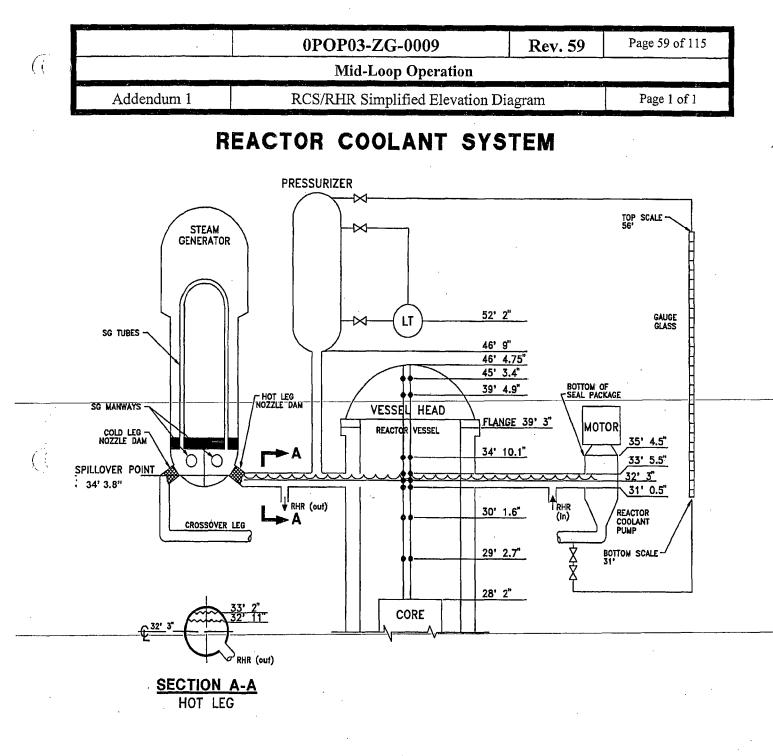
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Plant Cooldown

3.57	Minimize the time at lowered RCS inventory (fuel in the reactor with level at or below the reactor vessel flange). Controls for Infrequently Performed Evolution per 0PGP03-ZA-0506, Tests or Evolutions Requiring Additional Controls, and 0PGP03-ZO-0049, Conduct of Tests or Evolutions Requiring Additional Controls, SHALL be in place prior to lowering RCS level below 0% Pressurizer Cold Calibration Level elevation (elevation 52 ft 2 in) at Step 9.30.
3.58	<u>WHEN</u> Steam Generator (SG) temperature is lowered, <u>THEN</u> SG narrow range level indication will indicate higher than actual level.
3.59	Addendum 7 contains a list of conditions that should be met prior to taking credit for using the Steam Generators as a decay heat removal means while in Mode 5.
3.60	During plant cooldown, all SGs will normally be connected to the steam header to assure a uniform cooldown of the RCS. (UFSAR 5.2.2.11.3)
3.61	The Main Steam lines upstream of the MSIVs may require periodic blowdown for moisture control. This can be accomplished by performing Addendum 13. MONITOR the following "MAIN STEAM OUTLET DRIP LEG LEVEL SWITCH" Plant Computer — points for indications of moisture buildup in the Main Steam Lines:
	•
	• LD7900, S/G 1A(2A) MS LN DRN FROM MS-2001
	• LD7901, S/G 1B(2B) MS LN DRN FROM MS-2002
	• LD7902, S/G 1C(2C) MS LN DRN FROM MS-2003
	• LD7903, S/G 1D(2D) MS LN DRN FROM MS-2004
3.62	Deaerator Storage Tank Level SHALL be maintained in normal band of 60% to 80% when condenser vacuum is established. Going below 60% level may affect condenser vacuum. (Ref. 2.111)
3.63	The principles of 0PGP03-ZO-0042, Reactivity Management Program, are in effect at all
	times during Operations in this procedure.
3.64	Shutdown margin SHALL be verified adequate based on the RCS boron concentration.
3.65	IF planned to place the RCS in MODE 5 with reactor coolant loops NOT filled or MODE 6 <u>AND</u> planned to swap the CVCS Bed Demineralizers in service during RCS in MODE 5 with reactor coolant loops <u>NOT</u> filled or MODE 6 <u>THEN</u> FLUSH the oncoming Demineralizers per 0POP02-CV-0004, Chemical and Volume Control System Subsystem <u>PRIOR TO</u> entering RCS in MODE 5 with reactor coolant loops <u>NOT</u> filled and MODE 6 conditions. (Ref 2.57)
3.66	<u>IF</u> Personnel Air Lock (PAL) doors are open in Mode 5, <u>THEN</u> Addendum 21, Closure of Personnel Air Lock Doors, is available to establish containment closure.



STP D-0794 Rev 2

	0POP03-ZG-0009	Rev. 59	Page 60 of 11
	Mid-Loop Operation		
Addendum 2	RVWL Sensor Elevation		Page 1 of 1

NOTE

• Top of Core is elevation 28 ft 2 inches.

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• SG spillover is elevation 34 ft 3.8 inches.

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	SENSOR UNCOVERED	UPPER HEAD INDICATED LEVEL (%)	PLENUM INDICATED LEVEL (%)	SENSOR	LEVEL DESCRIPTION
	All Covered	100	100	46' 4.75"	Upper Head Full
	1	64	100	45' 3.4"	Upper Head Partially Drained
	2	0	100	39' 4.9"	Plenum Full
(3	0 ·	85	34' 10.1"	Plenum <u>NOT</u> Full (Enter Reduced Inventory)
	4	0	66	33' 5.5"	Top of Hot Leg Nozzle
	5	0	48	32' 3"	Hot Leg Centerline
	6	0	33	31' 0.5"	Bottom of Hot Leg Nozzle
	7	. 0	20	30' 1.6"	Midway between Hot Leg Nozzle and Upper Core Plate
	8	0	0 ·	29' 2.7"	Upper Core Plate

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0POP04-AE-0001

First Response To Loss Of Any Or All 13.8 KV Or 4.16 KV Bus

PROCEDURE PURPOSE

This procedure provides guidelines for the initial response and stabilization of the plant in the event of a loss of any single or all 13.8 KV bus(es) or 4.16 KV Bus(es). This includes all 13.8 KV Auxiliary and Standby buses, and 4.16 KV buses with the exception of Buses 1K(2K), 1L(2L) and 1M. Loss of a 4.16 KV ESF bus is addressed as it indicates at least a partial loss of offsite and onsite AC power (ESF bus power can only be completely lost if both offsite and onsite power sources to the specific bus are lost).

MAJOR ACTION CATEGORIES

- Provide interface with Emergency Operating Procedures and provide the instructions to establish the minimum equipment required to safely stabilize the unit.
- Identify actions associated with commitments to perform the action within a specified time period
 after the initiating event.

DISCUSSION:

The electrical distribution system at STP has by design, a high degree of flexibility and ability to withstand casualties, especially the Class 1E alternating current systems. However throughout the nuclear industry Loss Of Offsite Power (LOOP) events have occurred as well as Station Blackout (loss of all offsite and onsite AC power) events.

When dealing with a loss of offsite AC power, both complete and partial, with the Unit in Modes 1 or 2, the loss of an Auxiliary power bus will result in the loss of a Reactor Coolant Pump requiring a reactor trip because STP is not analyzed for operation with only three Reactor Coolant Pumps. In the event that no ESF bus is available the indication is that all offsite and onsite AC power has been lost requiring transition to the Emergency Operating Procedures. Under these same conditions STP has committed to shed the Channel I Load Sequencer from its power supply within the first 30 minutes after the initiating event, and if the associated battery bank has a jumpered cell then all the loads on DP 1201 and DP 1204 will be shed except for QDPS and SG PORVs.

The initial response provided by this procedure is directed to the stabilization of critical plant parameters and then analyzes the extent of the loss of power. While this procedure does not identify the specific combination of buses that have been lost, it does identify the specific area of the power loss so that a procedure that is more specific to the method for power restoration can be referred to.

This Procedure is Applicable in all Modes

0POP04-AE-0001	First Response To Loss Of Any Or All 13.8 KV Or 4.16 KV Bus	Rev. 44	Page 34 of 5
Addendum 4	Basis	Ba	sis Page 6 of 29
STI	EP DESCRIPTION FOR 0POP04-AE-0001 ST	<u>ГЕР 3.0</u>	
STEP: CHECK 4.16	KV ESF Bus Status:		
	KV ESF Bus NOT energized from offsite power phases of each ESF Bus).	er (VERIFY	the voltage
b. VERIFY A	Applicable STBY DG(s) running		
c. VERIFY A 4.16 KV E	Applicable STBY DG(s) output breaker(s) closed SF bus	l to the assoc	ciated
	nine the status of the 4.16 KV ESF buses and per rformed under the current conditions.	rforms any c	orrective
	mpts-to-start-SDG-for-a-de-energized-bus.—Also- if not determines the cause of the failure and pro		
	OCKOUT" indicating lamp on applicable BSMP rgized until corrective maintenance is complete.		illuminated
	e if the SDG is available to be started by checkin If available then perform the steps to start SD0		
breaker to energize th a fault does exist, the	running at this step, then determine the need to a associated bus and close the breaker in the even in the cause of the fault would have to be corrected be reset and the bus energized.	ent that no fa	ults exist. If
INSTRUMENTATIO		·	
CONTROL/EQUIPM			
KNOWLEDGE: If th	e SDG has a 4.16 KV ESF Bus overcurrent lock r an SDG overspeed lockout then these faults wi		
			·

0	POP04-AE-0004	Loss Of Power To One Or More 4.16 KV ESF Bus	Rev. 15	Page 77 of 95
Sector Sector				
H.	Addendum 14	Basis	Basis I	Page 1 of 18

PROCEDURE PURPOSE

The purpose of this procedure is to restore power to any ESF bus which is not energized. In the case where only one ESF bus is energized by a DG, and another one cannot be energized by the associated DG or offsite power, then steps are taken to operate breakers and disconnects to use the one running DG to supply key loads on another bus.

MAJOR ACTION CATEGORIES

- Tie the operating DG to another bus via the emergency switchgear bus 1L(2L).
- Energize at least one ESF bus from the Emergency Transformer.
- Control and load essential equipment on to the available ESF buses.

DISCUSSION:

STP has committed under specific conditions related to loss of offsite and onsite power to energize at least two ESF buses from a running DG in order to energize specific loads needed to extend station battery life or provide availability of ESF equipment that is electrically powered from one of two specific ESF buses.

This Procedure is Applicable in all Modes

0PSP03-EA-0002

ESF Power Availability

Rev. 32

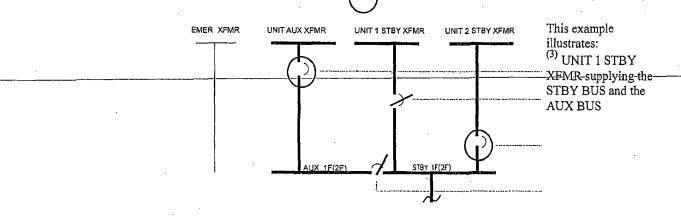
NOTE There are 5 possible lineups on Data Sheet 2, 3, and 4 for 13.8 KV XFMRS in the **DESIGNATED** Class 1E 4160 VAC Bus Power Source Table that meet Technical Specification requirements for being a power source for the 4.16 KV Buses: • (1) UAT supplying the AUX BUS and STBY BUS • (2) UAT supplying the AUX BUS and UNIT 1 STBY XFMR supplying the STBY BUS (2) UDIT 1 STDY VID (D)

• (3) UNIT 1 STBY XFMR supplying the STBY BUS and the AUX BUS

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- (4) UAT supplying the AUX BUS and UNIT 2 STBY XFMR supplying the STBY BUS
- (5) UNIT 2 STBY XFMR supplying the STBY BUS and the AUX BUS
 - 5.2 COMPLETE Required ESF Power Train Data Sheet 2 through 4 by performing the following steps.
 - 5.2.1 RECORD actual breaker/disconnect positions for the 13.8 KV XFMRs, AUX BUS, STBY BUSES and from the 13.8 KV STBY BUS to the 480 VAC BUSES.
 - RECORD "CLOSED" breaker/disconnect positions by drawing a line at an angle through the breaker.
 - RECORD "OPEN" breaker/disconnect positions by drawing a CIRCLE around the breaker.



0PSP03-EA-0002

ESF Power Availability

6.0 <u>Acceptance Criteria</u>

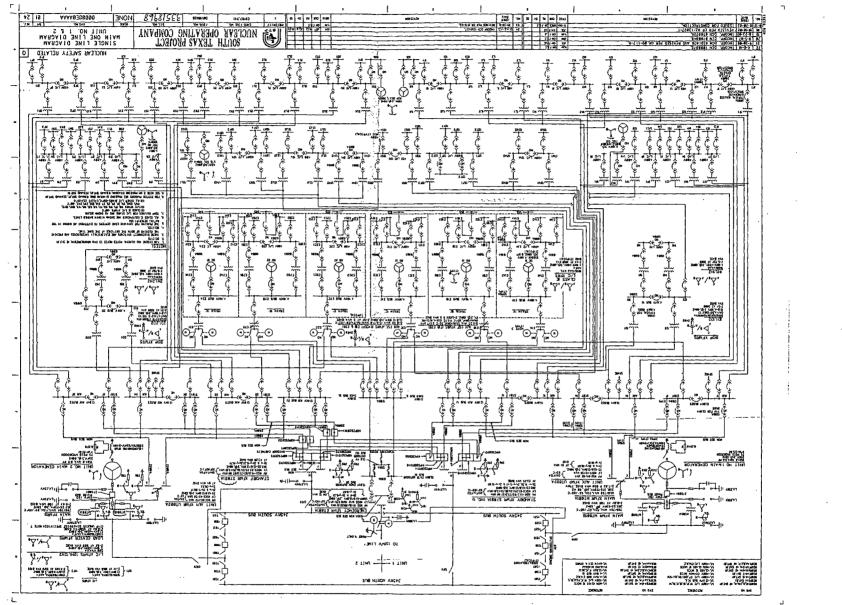
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NOTE							
•	 Addendum 2, Two Physically Independent Circuits, provides a drawing of rights of way and offsite circuits to aide in the definition of "two physically independent circuits". 						
•	 Loss of one 13.8 KV Standby Bus to 4.16 KV ESF bus line constitutes loss of one required offsite source. (Reference 8.2) 						
•	• Loss of two 13.8 KV Standby busses to 4.16 KV ESF bus lines constitutes loss of two required offsite sources. (Reference 8.2)						
•	• The preceding notes also apply when the 4.16 KV ESF bus is not energized by the 13.8 KV XFMR.						
• Step 6.1 applies during standby diesel inoperability.							
• Step 6.2 applies during offsite independent circuits inoperability.							
• Note and Precaution 3.28 should be referred to for additional clarification regarding allowable indication to be utilized when obtaining 345 KV switchyard voltage.							
	6.1 Two physically independent circuits exist between the offsite transmission network and onsite Class 1E Distribution System as determined from Data Sheet 1, 2, 3, 4, and 9. (Technical Specifications 3.8.1.1.b, 3.8.1.1.f, and 4.8.1.1.1.a.)						
• North and South Bus in service with bus voltage:							
\circ ≥ 340 KV" for NORMAL LINEUP							
	OR						
·	\circ ≥ 356 KV for NORMAL LINEUP with UAT or Train B ESF LTC in "MANUAL"						
	OR						
	 ≥ 358 KV for all ALTERNATE LINEUPs OR voltage specified in the "Minimum Voltage for Various Alternate 13.8 KV Bus Alignments" Addendum— of 0POP02-AE-0002, Transformer Normal Breaker and Switch Lineup. 						
	• Two of the following Rights of Way with a 345 KV line are available:						
	• NW Right of Way 1 (White Point 39)						
	• NW Right of Way 2 (Elm Creek 27 <u>OR</u> WA Parish 39 <u>OR</u> Elm Creek 18)						
	 Eastern Right of Way (Dow Velasco 27 <u>OR</u> Dow Velasco 18) 						
	• Two of the following 13.8 KV XFMRs are available:						
	o Unit Aux XFMR						
	o Unit 1 Stby XFMR						
	o Unit 2 Stby XFMR						

• Three 13.8 KV Standby Buses energizing the 4.16 KV ESF bus lines.



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TABLE 1.2

OPERATIONAL MODES

MODE			REACTIVITY CONDITION, Keff	% RATED THERMAL POWER*	AVERAGE CODLANT TEMPERATURE
cu ma'	1.	POWER OPERATION	<u>></u> 0.99	> 5%	≥ 350°F
	2.	STARTUP	<u>></u> 0.99	<u><</u> 5%	≥ 350°F
	3.	HOT STANDBY	< 0.99	0	<u>></u> 350°F
	4.	HOT SHUTDOWN	< 0.99	0	350°F > T > 2D0°F avg
	5.	COLD SHUTDOWN	< 0.99	0	<u><</u> 200°F
	6.	REFUELING**	<u><</u> 0.95	0	≤ 140°F
				,	

*Excluding decay heat.

**Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

SOUTH TEXAS - UNITS 1 & 2

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	OPCP	04-ZA-0307		Rev. 6		0-20		
· ·		ration of C		1				
Form		Calculation C						
1.011								
	CALC	JLATION (OVER	SHEET	•	Page 1		
Calculation]	No.: <u>13-DJ-006</u> U	nit: <u>9</u>	Bldg	g/Area/Sys: <u> </u>	ARIOUS			
		2						
[] Design	Calculation [X] Enginee	ring Calcula	tion	Cog. Org.:	ELECTR	ICAL		
Title: 125 VI	OC BATTERY FOUR HOUR	COPING A	NALYS	SIS				
Additional	Dept: N/A	Si	gnature:		Date:			
Review: Additional	Dept: N/A	 C:	gnature:		Date:	<u></u>		
Review:			gnature:		Date:			
RPE Certific Required:	ation [] Yes	[X] No			. •			
RPE Signatu	re: <u>N/A</u>	Date:		Registration	No.:			
RPE Seal:								
	lation revision contains a chang	e in the meth	odology :	as described in U	JFSAR Section			
Ne	·							
CR actions t	racking documents impacted	by this revi	sion to tl	he calculation:	ORIG	INAL		
	N/A	-						
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	Approval Signature PRINT/SIGN	Date	Rev	Revisio	on Description			
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SOUTH TEXAS PROJECT ELECTRICAL CALCULATION

SUBJECT 125 VDC BATTERY FOUR HOUR COPING ANALYSIS

CALCULATION 13-DJ-006 REV. 0

5.0 ACCEPTANCE CRITERIA

5.1 Battery Size

Page 14

The required battery size, as calculated using the IEEE Standard 485-1978 (Ref. 6.3.2) methodology, must be less than or equal to the installed battery size, including the impact of minimum temperature and aging factors. This is determined by comparing the number of positive plates calculated to the actual number of positive plates for the installed battery.

STP's defense-in-depth strategy requires the four Class 1E DC channels to be "AC-Independent" for a minimum of four (4) hours, to facilitate coping with a postulated loss of AC power event. The results of this calculation show the following:

5.1.1. With no battery cells jumpered out (i.e. 59 cell operation), the Class 1E DC Channel I can operate for a period of four (4) hours without battery charging support by manually de-energizing ESF Load Sequencer "A" within 30 minutes following the loss of Channel I battery charging capability. Class 1E DC Channel II can operate for a period of eight (8) hours without battery charging support and without shedding of any connected loads. Class 1E DC Channels III and IV can operate for a period of four (4) hours without battery charging support and without shedding of any connected loads.

5.1.2. With one battery cell jumpered out (i.e. 58 cell operation), the Class 1E DC Channel I can operate for a period of four (4) hours without battery charging support by manually de-energizing the ESF Load Sequencer "A" and shedding all but three loads on Panel 1201 within 30 minutes following the loss of Channel I battery charging capability. Class 1E DC Channel II can operate for a period of eight (8) hours without battery charging support and without shedding of any connected loads. Class 1E DC Channel III can operate for a period of four (4) hours-without-battery-charging-support and without-shedding of any connected loads. Class 1E DC Channel IV can operate for a period of four (4) hours without battery charging support by manually shedding all but three loads on Panel 1204 within 30 minutes following the loss of Channel IV battery charging capability. The loads on both 1201 and 1204 are breakers 13, 15, and 17 which are QDPS APC A1 (C1) at 7 amps, QDPS APC A2 (C2) at 10 amps and Steam Generator 1A PORV Servo Amplifier at 2 amps, according to EC-5008. This is a total of 19 amps which are then converted to power at 120 AC resulting in a power of 2280 watts. The efficiency losses per EC-5008 are 2511 w. Summing these results in a power of 4791 W which are then converted back to DC amps at 125 VDC. resulting in the loads on the respective batteries EIV1201 and EIV1204 of 38.328 amps after 30 minutes.

The minimum battery voltage that was used in this calculation for all safety batteries to calculate the margin above was 106 volts. As an input to operations Emergency Operating Procedure 0POP05-EO-EC00 'Loss of All AC Power', the

SOUTH TEXAS PROJECT ELECTRICAL CALCULATION

SUBJECT 125 VDC BATTERY FOUR HOUR COPING ANALYSIS

CALCULATION 13-DJ-006 REV. 0

minimum bus voltage that any safety train can operate to is 105.5 VDC. Below 105.5 VDC it may be possible that some loads will have inadequate voltage to operate properly.

6.0 REFERENCES

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6.1	Regul	atory
•	6.1.1	South Texas Project Technical Specifications and Bases, Amendment 198 for - Unit 1 and Amendment 186 for Unit 2
	6.1.2	South Texas Project Updated Final Safety Analysis Report (STP UFSAR) Chapter 8, Revision 16
	6.1.3	Letter from T. H. Cloninger, STPEGS, to the NRC Document Control Desk, Revised Position of 10CFR50. 63, "Loss of All Alternating Current Power," dated March 1, 199. (ST-HL-AE-5010)
	6.1.4	Letter from T. H. Cloninger, STPEGS, to the NRC Document Control Desk, Supplemental Information to Revised Position of 10CFR50.63, "Loss of All Alternating Current Power," dated June 14, 1995 (ST-HLAE-5103)
	6.1.5	Letter from the Thomas W. Alexion, NRC, to Mr. William T. Cottle, STPEGS, Revised Station Blackout (SBO) Position, South Texas Project, Units 1 and 2 (STP), dated July 24, 1995 (TAC Nos. M90061 and M90062) (ST-AE-HL-94257)
	6.1.6	10CFR 50.59 Screen # 10-17753-5 Revision 0, "Revise Station Blackout Position to delete the need for a Coping Analysis"
6.2	Techr	lical
	6.2.1	Class 1E 125 VDC Design Criteria Document, 4E520EQ0100, Rev 6
	6.2.2	VTD-A363-002 Rev 6, "Instruction and Operating Manual 10 KVA Inverter" *
· · .	6.2.3	VTD-A363-0045, Rev 1, "Vendor Technical Manual for Ametek Solidstate Controls 25KVA Inverter / Rectifier"
	6.2.4	EC05036 Rev 8 "DC Cable Sizing"
	6.2.5	EC05037 Rev 5 "Maximum Allowable Length of AC Power Cables"
	6.2.6	EC06038 Rev 9 "Power Cable Sizing Verification"
	6.2.7	VTD-S637-0009 Rev 1, ESF Load Sequencer for South Texas Project Electric Generating Station
· · · ·	6.2.8	DCN 9602493, dated 4/29/96

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0POPO5-EO-EC00

LOSS OF ALL AC POWER

PAGE 6 OF 7

ADDENDUM 4 VITAL DC BUS MONITORING

STEP

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ACTION/EXPECTED RESPONSE

RESPONSE NOT OBTAINED

CAUTION

Do <u>NOT</u> allow battery voltages to drop to LESS THAN 105 VDC for plant equipment protection.

NOTE

Train A, B, and C bus voltages should be monitored for the duration of the event, and their respective battery output breakers opened if bus voltages lowers to LESS THAN OR EQUAL TO 105.5 VDC in order to conserve the battery should a STBY DG become available.

MONITOR Class 1E 125 VDC system Train A, B, & C bus voltage.

a. Train A <u>AND</u> B bus voltages -GREATER THAN 105.5 VDC a. PERFORM the following:

- DISPATCH operator to perform ADDENDUM 3, FAILING AIR TO MSIVS AND MSIBs for all MSIV(s) and MSIB(s).
- 2) <u>WHEN</u> ADDENDUM 3, FAILING AIR TO MSIVS AND MSIBS is complete, <u>THEN</u> GO TO Step 4.b of this Addendum.

___b. Train A, B, <u>OR</u> C bus voltages -GREATER THAN 105.5 VDC. b. DISPATCH operator to open the associated battery output breaker:

o "BTRY E1A11(E2A11) MAIN BKR" E1A11(E2A11) BKR 1B (EAB 10')

o "BTRY E1B11(E2B11) MAIN BKR" E1B11(E2B11) BKR 1B (EAB 35')

o "BTRY E1C11(E2C11) MAIN BKR" E1C11(E2C11) BKR 1B (EAB 60')

RETURN TO procedure step in effect.

CHECK Sequencer(s) ready for restoration following bus energization

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		Emerge	ency Communications					
	Quality	Non Safety-Related	Usage: Available	Effective Date	: 12/03/09			
141114-04	S, Korenek	N/A	N/A	Emer	gency Response Division			
	PREPARER	TECHNICAL	. USER	COGN	IZANT ORGANIZATION			
<u>`able</u>	e of Contents				Page			
.0	Purpose and	Scope			· · · · · · · · · · · · · · · · · · ·			
.0	Definitions2							
.0	Responsibilities2							
.0		Communications System .						
0.0	Maintenance		•••••••••••••••••••••••••••••••••••••••					
5.0								
.0	Support Doc	uments			1			
	Addendum 1	, Communications Conso	le Panel					
	Addendum 2, Notification Methods to Offsite Agencies							
	Addendum 3, Station Public Address Selections							
	Addendum 4, Related Maintenance Jacks							
	Addendum 5	5, Portable Satellite Teleph	none		2			
	Addendum 6	, Desktop Satellite Telepl	ione Troubleshooting		2			

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			Emergency Communications					
		Ð						
	1.	Purpo	ose and Scope					
		1.1	This procedure provides guidance in the use of emergency communications systems when responding to an emergency at the South Texas Project Electric Generating Station (STPEGS).					
	2.	Defin	nitions					
		2.1	FTS 2001 System: A federal telephone system used by the Nuclear Regulatory Commission (NRC) and nuclear utilities for emergency communications.					
		2.2	RINGDOWN LINE: A telephone line that does <u>NOT</u> require the operator or caller to dial number to activate the circuit.	a				
. .		2.3	UNIT OVERRIDE: A circuit select switch (CSS) found on selected communications consoles, which when selected, activates prioritization circuitry for public address — announcements. Additionally, when activated, this button directs announcements to ALE public address zones.					
	3.	Responsibilities						
		3.1	The Emergency Director, or designee, is responsible for activating the Emergency Notification System (ENS) to notify the NRC of a declared emergency, and to maintain communications with the NRC Operations Center.	·				
		3.2	The Emergency Director, or designee, is responsible for activating the State/County ringdown line to notify State/County officials of a declared emergency.					
		3.3	The Radiological Manager or Radiological Director is responsible for activating the Heal Physics Network (HPN) if requested by the NRC, to inform the Health Physics Section o the NRC of the emergency radiological environmental conditions and to coordinate health physics information and response during a declared emergency at STP.	f				
		3.4	The Manager, Information Technology or designee is responsible for the installation, testing, maintenance, and modifications of the emergency communications systems.					

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			Emergency Communications		
4. Eme	rgency Co	າາມານ	nications System		
			NOTE		· · · · · · · · · · · · · · · · · · ·
			otification Methods to Offsite Agencies e used throughout this procedure.	, for alternate teleph	none numbers and
4.1	Emerge	ncy 🛛	Felephone Circuits		
	4.1.1	En	nergency Notification System (ENS)		
		•	The ENS is a telephone circuit provid FTS 2001 telephone. The principal m NRC is the ENS. The circuit may als is activated to notify the NRC of decl communications with the NRC Opera	ethod of communic o be activated by th ared emergency and	ations with the e NRC. The ENS
		•	IF the ENS is determined to be out of service, THEN notify the NRC Opera		ubsequent return t
		•	ACTIVATE the ENS by lifting the has appropriate number.	andset on the teleph	one and dialing th
	4.1.2	St	ate and County Ringdown Line		
		•	The State/County ringdown line is pr officials of a declared emergency. Th automatic ringdown telephone circuit console OR an ORANGE telephone.	he State/County ring	gdown line is an
		•	ACTIVATE the State/County ringdo	wn line by:	
			- LIFTING the HANDSET on the	ORANGE telephon	e
			OR		
			- UTILIZING the communication of Communications Console System		ce with Step 4.8,

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(1)			0PGP05-ZV-0011	Rev. 7	Page 4 of 23
			Emergency Communications	Bottleve and the state of the second	an Sama an Anglana an Santa da Anglanda. Anglanda
		4.1.3	Health Physics Network (HPN)		
			• The Health Physics Network (HPN) is NRC and is terminated on an FTS 200 request of the NRC. The HPN telepho communications with the NRC Health power plants during a declared emerge MAY request a conference call with o by asking the NRC to connect the desi	1 telephone. It is to one is designed to p Physics Section an ency. STP health p ther nuclear power	be used only at the provide nd/or other nuclear physics personnel
			• IF the HPN telephone line is determine subsequent return to service, THEN no (IEN 89-19)		
			• ACTIVATE the HPN by lifting the ha appropriate number.	ndset on the teleph	one and dialing the
	<u> </u>	4.1.4	STP Coordinator Ringdown Line		
(· · · · · · · · · · · · · · · · · · ·		• The STP Coordinator ringdown line is Qualified Scheduling Entity (QSE) and		
			 Utilize the communications console in Communications Console System. 	accordance with S	Step 4.8,
	4.2	Telephor	ne System		
		4.2.1	The STP Telephone System consists of co telephone switching equipment and cable regular telephone services via an onsite d are provided by Verizon and Southwester	. The onsite systemarcation point.	m is connected to The offsite services
			telephone services are augmented by a Co operated microwave system. The microw data services via tie lines into the Housto office telephone system interconnects int Houston. The combined microwave and provide augmentation to the normal local STP.	enter Point Energy vave system provid n corporate offices o the local telepho corporate office te	owned and les telephone and The corporate ne system in lephone systems

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Sana (SVI) A		Emergency Communications		an a
	4.2.2	Calling in (from offsite) may be accomplis	shed in one of two	ways:
		• Direct inward dialing (DID), OR		• •
-		• Calling the site number of (361) 972-3 attendant. Direct inward dialing extensions must go through the automated attenda	sions begin with a	
	4.2.3	Calling offsite (from onsite) may be accor	nplished in one of	two ways:
		• DIAL 9-1-AREA CODE - telephone r	umber, OR	
		• DIAL 32-0 to Center Point Energy and	d have the Operato	or complete the call
	4.2.4	Onsite calling is accomplished by dialing	the desired extensi	ion number.
<u>-</u>			ovided to Offsite F	ield-Teams-as a——
	4.3 Portab	back up to radio communications.	· · ·	
				· · · · · · · · · · · · · · · · · · ·
		NOTE		
own		nt satellite telephones are provided to the Stat cial telephone equipment/services. These tele ia satellite.	-	
	4.3.1	Need clear view of the sky, outdoors, awa	y from buildings a	and tall structures.
	4.3.2		old-the-Power-Bu	tton-for-1-t o-2
	4.3.3	Rotate and pull extend antenna into vertic	al position.	
	4.3.4	To dial, press and hold the 0+ button until is an international calling code), then pro	1 V	<u> </u>
		distance call $(1 + \text{area code} + \text{phone num})$		ne uny outer rong
	4.3.5		ber).	

4.3.6 Talk with antenna above your head and vertical to the ground.

4.3.7 When you complete the call press OK again to hang up.

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	Emergency Communications	e waaroo narke shekeelee eeksise	
4.3.8	Each portable satellite telephone is labeled an outside caller to call back.	with number and	required codes :
4.3.9	To retrieve Voice Mail messages perform the	ne following:	
	• Dial the satellite telephone number.		
	• During the voice greeting, enter *.		
	• When prompted for your password, enter	er 1111.	
	• Follow the voice prompts to:		
	a. Play your messages.		
	b. Record a Message.		
<u>.</u>	cChange your-greeting		
	d. Access personal options.		
	e. Make a call.		
4.4 Deskto	p Satellite Telephone		

<u>NOTE</u>

Independent desktop satellite telephones are provided to the Station as a backup to all company owned and commercial telephone equipment/services. These telephones can be utilized for worldwide access via satellite. A desktop satellite telephone is maintained in both control rooms, both Technical Support Centers, and the Emergency Operations Facility.

4.4.1

Although similar in many respects to a normal telephone, the desktop Satellite Telephone has some differences:

• Pick up the telephone handset and listen; you should hear the normal steady state dial tone. The Satellite Terminal Box call status indicator should also shine green continuously. A continuous orange indicator signifies acceptable but marginal signal strength. The Satellite Terminal Box is located in the EOF Communications Room, and in the Unit 1 and 2 TSC Copy Room's on the communications rack.

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an an that the second states	2+2242/02-24-6-6-6-592-84 -	nen en ante la presenta de la contra de la presenta de la presenta de la presenta de la presenta de la presenta La presenta de la pres		
		a. If you hear nothing, there is potential telephone or cable. Refer to A Troubleshooting.		
		b. If you hear a single tone interr check that the signal strength may be a problem with your S Satellite Telephone Troublesh	indicator is orange or a IM card. Refer to Add	green. If not, there
		•. Dial 001 + area code + phone num number you will hear progress pip to 30 seconds for the Iridium netw stage is not unusual.	s from the Iridium net	work. It can take up
		• Eventually you will hear the other message indicating why your call answers the call status indicator w orange, indicating a call in progres	was unsuccessful. Whe	en the other party
		• To terminate the call just hang up	the handset. The call s	tatus light turns off.
	4.4.2	Each desktop satellite telephone is lab an outside caller to call back.		
	4.4.3	To retrieve Voice Mail messages perfo	orm the following:	
		• Dial the satellite telephone numbe	r.	
		• During the voice greeting, enter *.		
		 When prompted for your password 		
			i, enter 1111.	
		• Follow the voice prompts to:		
		a. Play your messages.		
		b. Record a Message.		
		c. Change your greeting.		
		d. Access personal options.		
		e. Make a call.		
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	Emergency Communications		

4.5 Radio Communications

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- 4.5.1 The Radio Communications System consists of repeaters, mobile, handheld, and base two-way FM transceivers licensed to STP Nuclear Operating Company by the Federal Communications Commission. The radio repeaters are installed in a radio communications building at the base of the radio antenna tower onsite. The repeaters are supplied normal power from the plant power and emergency power from an automatic starting engine driven generator. The generator is supplied fuel from a local fuel tank. The handheld, mobile and base stations are programmed to operate through the repeaters or direct.
- 4.5.2 Radio communications with the Matagorda County Emergency Operations Center is accomplished by the use of a radio transmitter/receiver in the Security Central and Secondary Alarm Stations, and a transmitter/receiver at the Matagorda County Sheriff's Office tuned to an STP radio frequency.
- 4.5.3 Offsite Field Team radio communications are accomplished on STP Nuclear Operating Company licensed radio channels. The repeaters provide coverage of the ten mile Emergency Planning Zone from one handheld radio to another handheld radio or to a base station.

CAUTION

Handheld radios <u>SHALL NOT</u> be used to transmit from inside the ESF Switch Gear Room, Control Room, Technical Support Center, Emergency Operations Facility, Auxiliary Shut Down Panel Rooms, Computer Rooms, nor within ten (10) feet of an open instrument cabinet, computer or computer terminals. The only exceptions to the above restrictions are emergencies where a threat exists to the plant <u>OR</u> human safety and no other means of emergency communications are available.

4.5.4 PERFORM the following to use a radio for communication:

- ALIGN the assigned radio channel on the handheld by selecting the appropriate channel number and Modes A and B for repeater or Mode C for direct communication.
- PRESS the microphone button and talk, keeping the microphone about 2 inches in front of the mouth, and
- RELEASE the microphone button to receive, <u>AND</u> ADJUST the volume by turning the knob marked VOL.

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i Si sinawi	an a		Emergency Communications	an a	States & the States States of States
			• ADJUST the squelch by turning the known heard, then back until the speaker is quie sensitivity, only on mobile radios.	b marked SQUE t. This setting is	LCH until noise is for the maximum
			• Communicate with other portable, mobil	e or base radio s	tations.
	4.6	Glenayı	e Paging System		
•		4.6.1	The Glenayre Paging System is a tone system telephones or from an offsite touch-tone tele over a 60-mile radius from the site. The system emergency power generators with automatic	phone. The syst tem transmitters	em has a range of are connected to
		4.6.2	Instructions for activating the Glenayre Pagin 0ERP01-ZV-IN03, Emergency Response Or		
	4.7	Mainter	nance Jack Communications System	<u> </u>	
** 		4.7.1	A maintenance jack amplified and sound-pow for onsite communication between certain an Related Maintenance Jacks. The system is p designed circuits. Each circuit may be activ circuit by the proper selections on the system Control Room. The system has the capabilit activated circuit is one loop that interconnec terminals into one circuit.	reas. Refer to A powered by ampl ated or combine n control panels ty to be voice ac	Idendum 4, ifiers on pre- d with another located in each tivated. The voice
		4.7.2	IF it is desired to have amplified voice comm following:	unications, THE	N PERFORM the
			• SELECT the desired zones on the select	ion panel in the	Control Room.
			- INSERT a headset plug into one of tarea.	he jack stations	narked 1 or 2 at t
			- INSERT a headset plug into the jack	marked plant fo	r voice-nowered

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	an the state of th	Emergency Communications	T MERICAN STRAT	
4.8	Commu	nications Console System		
	4.8.1	 The communications console is an integrated switching system which is subdivided into a telephone, radio (RF), public address (PA), voice direct line (VDL). Refer to Addendu Panel, for locations of the console controls. composed of several two-position switches. MONITOR - Top position (amber light TALK/LISTEN - Down position (green) 	seven groups: di alarm system, c m 1, Communic Each communi These position will glow)	rect line (ringdown) onference, and ations Console cations group is s are:
	4.8.2	When the TALK/LISTEN switch is activate Ringdown line this locks out all other comr is completed, deactivate by depressing the time to clear the green light.	d (green light) fo	or the State/County soles. When the cal
	4.8.3	These Consoles are installed in the Control I Rooms, Operations Support Centers, Techn Operations Facility, Security Force Supervi Alarm Stations, Simulator and in the Maint Refueling Outages, console(s) may be insta Stop Shop.	ical Support Ce isor's Office, Ce enance Office F	nters, Emergency ntral and Secondary acility. During
		NOTE		<u></u>

Many circuits may be monitored simultaneously. These circuits are heard through the left ear if using the headset. The volume for the monitor position is controlled by the MONITOR VOLUME control

located in the Handset/Headset Control Group.

Usually the communicator operating the console will be talking (TALK/LISTEN switch is activated) on only one circuit at a time. These conversations will be heard through the right ear if using the headset. The volume control for the TALK/LISTEN position is controlled with the RECEIVE VOLUME control also located in the Handset/Headset Control Group.

The communicator may actively communicate with all circuits simultaneously. It is important to note that all circuits with the TALK/LISTEN switch activated will hear the communicators conversation, which may not be desirable. To deactivate, depress the TALK/LISTEN switch a second time to clear the green light.

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Emergency Communications

4.8.4 Direct Line (Ringdown) Group Operation

CAUTION

Activating the circuit select switch (CSS) in the MONITOR (top position) will activate an Idle Circuit and cause the ringdown line to ring. The position switch SHALL be in the TALK/LISTEN (bottom position) before speaking.

a. WHEN it is desired to place a call, THEN perform the following:

<u>NOTE</u>

The next step will ringdown the other phone.

- Activate the appropriate circuit select switch-in-the TALK/LISTEN position.
- WHEN the phone is answered, THEN PRESS the push-to-talk button when speaking.
- WHEN communication is terminated, THEN DEACTIVATE the bottom TALK/LISTEN position switch.

<u>NOTE</u>

An audible signal will be heard through the speaker and the CSS red lamp will flash when a party is calling.

- b. WHEN a call is received, THEN perform the following:
 - ACTIVATE the bottom TALK/LISTEN position switch.
 - WHEN it is desired to talk, THEN press the push-to-talk button when speaking.
 - WHEN communication is terminated THEN deactivate the bottom TALK/LISTEN position switch.

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		Emergency Communications	ICV. /						
	4.8.5	Telephone Group Operation		yda podróg nej zakon si konstru i konstrukcija i bila basi na					
	NOTE								
	Al	normal site phone functions are available th	rough the console	•					
		a. IF it is desired to make a call, THEN P	ERFORM the foll	owing:					
		• ACTIVATE the circuit select switt TALK/LISTEN (bottom) position received on the headset or handset	AND WAIT until						
		• DIAL the number using the teleph	ione keypad.						
		• WHEN the number called answerbutton-while-speaking	s, THEN PRESS t	he push-to-talks					
· .	· ·	• WHEN communication is termina TALK/LISTEN switch.	ited, THEN DEAC	CTIVATE the					
		b. WHEN a call is received, THEN PERI	FORM the followi	ng:					
		NOTE	· · · · · · · · · · · · · · · · · · ·						
	An audible signal wi party is calling.	ll be heard through the speaker and the CSS	red light will flash	when another					
		• ACTIVATE the circuit select swi (bottom position.	tch (CSS) in the T	ALK/LISTEN					
		 WHEN it is desired to talk, THEN speaking. 	NPRESS the push	-to-talk button while					
		• WHEN communication is termina position TALK/LISTEN switch.	ated THEN DEAC	TIVATE the two					
		• IF it is desired to place a call on h MONITOR switch.	old, THEN ACTI	VATE the					

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	Emergency Communications		

4.8.6 Radio Group Operation

<u>NOTE</u>

Radio channels may be monitored by moving the circuit select switch (CSS) to the MONITOR (top) position.

- a. IF it is desired to transmit a message on a radio frequency, THEN activate the circuit select switch to the TALK/LISTEN (bottom) position.
- b. PRESS the push-to-talk button when speaking.
- c. WHEN communication is terminated THEN deactivate the bottom TALK/LISTEN position switch.

4.8.7 Plant Public Address and Alarm System

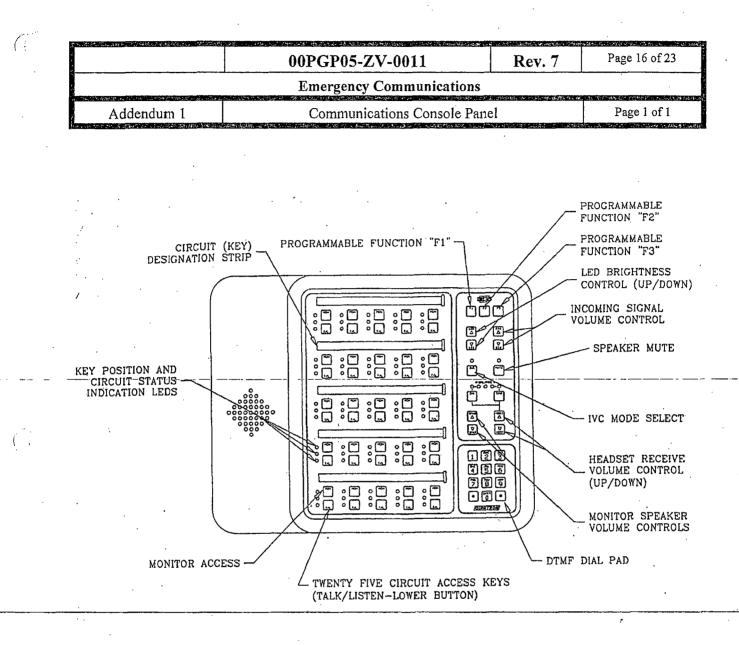
<u>NOTE</u>

Emergency alarm and public address override switch capabilities are found on the communications console panels in the following locations: all panels in each Unit's Control Room, and Technical Support Center, the Emergency Operations Facility, Central Alarm Station, Secondary Alarm Station, and the Simulator.

- a. IF it is desired to make a public address announcement, THEN perform the following:
 - SELECT the two position switch corresponding to the desired zone (listed on Addendum 3) that is to receive the announcement.
 - Activate the two position switch(es) to the TALK/LISTEN (bottom) position in the appropriate zone(s).
 - PRESS the push-to-talk button when speaking.
 - Deactivate the bottom TALK/LISTEN position switch at the conclusion of the announcement.

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to a star of the start of the start of the	Emergency Communications		- 1926 - A. MARING - MARINE MARINE		
	b. Emergency Public Address Alarms an	d Announcement			
	NOTE				
There are three put	olic address emergency alarms: Assembly, Fire	, and RCB Evacua	tion Alarm.		
	adcast as directed over the PA system. Alarm s he PUSH-TO-TALK button on the handset is a		r 8 seconds, then		
	• WHEN directed, THEN select the	e appropriate alarm	1.		
 WHEN the alarm is completed, THEN DEACTIVATE the alarm switch, activate the Unit override switch, AND make the appropriate emergency announcement over the PA system as directed. WHEN the alarm/announcement is completed, THEN deactivate all switches. 					
<u>NOTE</u>					
	NOTE				
Loops may be mor	<u>NOTE</u> nitored for informational purposes by selecting	the MONITOR ci	rcuit select switch		
Loops may be mor		······································			
Loops may be mor	nitored for informational purposes by selecting	group conference:			
Loops may be mor	nitored for informational purposes by selecting a. PERFORM the following to establish	group conference: es are on the same es on the loop have	loop. e the circuit select		
Loops may be mor	 a. PERFORM the following to establish VERIFY that all conferring partic VERIFY that all conferring partic 	group conference: es are on the same es on the loop have EN (bottom) positi	loop. e the circuit select on.		
Loops may be mor	 a. PERFORM the following to establish VERIFY that all conferring parties VERIFY that all conferring parties VERIFY that all conferring parties WHEN it is desired to talk, THE 	group conference: es are on the same es on the loop have EN (bottom) positi N press the push-to	loop. e the circuit select on. o-talk button wher		

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		Sector States	Emergency Communications
			4.8.9 Voice Direct Line (VDL)
•			a. The Voice Direct Line (VDL) is a direct line from Quintron communication console to console.
			• Lift the handset on the appropriate console.
			• Activate the appropriate circuit selector switch on the communication to the TALK/LISTEN position.
	5.	Maint	tenance
		5.1	Information Technology personnel SHALL maintain the emergency communications systems.
		5.2	Maintenance SHALL be done as required to keep the system in good operating condition and as committed to in license documents.
	6.	Refere	ences
2 \$		6.1	NUREG-0654/FEMA-REP-1, Criteria For the Development and Evaluation of Emergency Preparedness in Support of Nuclear Power Plants
		6.2	South Texas Project Electric Generating Station Emergency Plan
		6.3	0PGP07-ZA-0011, Communications Systems
		6.4	0ERP01-ZV-IN03, Emergency Response Organization Notification
`.		6.5	IEN 89-19, Health Physics Network
	7	Suppo	ort-Documents
		7.1	Addendum 1, Communications Console Panel
		7.2	Addendum 2, Notification Methods to Offsite Agencies
		7.3	Addendum 3, Station Public Address Selections
		7.4	Addendum 4, Related Maintenance Jacks
		7.5	Addendum 5, Portable Satellite Telephone
		7.6	Addendum 6, Desktop Satellite Telephone Troubleshooting



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Emergency Communications						
Addendum 2 Notification Methods to Offsite Agencies Page 1 of 1						

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ENS	STATE (DPS, PIERCE)	COUNTY (SHERIFF)	HPN
1-301-816-5100	1-979-541-4595	1-979-241-3205	1-301-816-5100
1-301-951-0550	N/A	1-979-244-1178 (ONLY when EOC is Activated)	1-301-951-0550

	NRC	State/County
ENS	X	
Ringdown Line to the DPS, Disaster District Sub 2C (State of Texas) and the Matagorda County Sheriff's Office (Matagorda County).		X
Outside Telephone Lines.	X	X
Satellite Telephone.	Х	X
Unit 1 Control Room Direct Line to Bay City.	Х	X
Microwave Line to Center Point Energy and call forwarded to the appropriate number.	X	X
Ringdown Line to the STP Coordinator (QSE) and request the call be forwarded to the appropriate number.	X	X
Security Radio communication to the Matagorda County Sheriff's Office and request the call information be passed onto the appropriate number.	Х	X

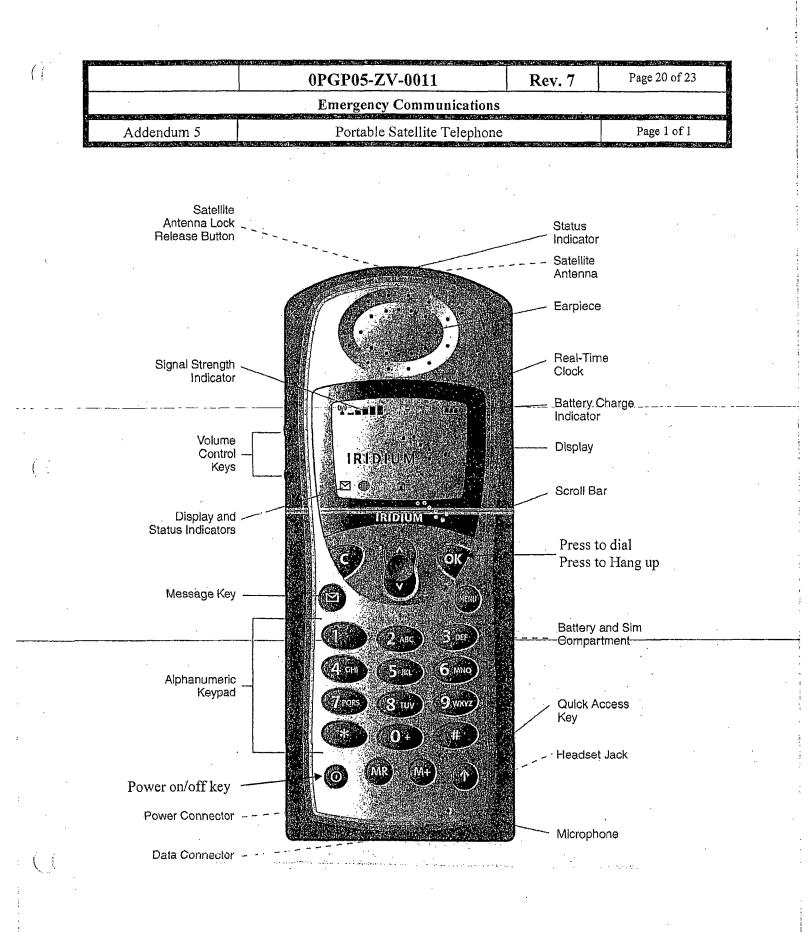
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and the second	en an	Em	ergency Comn	nunications	and a subscription of a subscription of a subscription of the subs	a the state of the second second	an a
Addendum 3	an ta Martin Constant an Anna di Charles de tres	Station P	ublic Address S	elections		Pag	elof1
Unit 1		nit 2	Units 1 & 2	Unit	50		51
ALL Zone 1		LL ne 2	ALL Zones	Override Zones	Telephon Zone 3		Telephone Zone 4
Electrical Auxiliary Building (EAB) Mechanical Auxiliary Building (MAB) Isolation Valve Cubicle (IVC) Reactor Containment Building (RCB) Fuel Handling Building (FHB) Diesel Generator Building (DC Turbine Generator Building (T	(MAB) Isolation Valve Reactor Contain (RCB) Fuel Handling B BB) Diesel Generato	iliary Building Cubicle (IVC) ment Building	1,2, & 3 Unit 1 & 2 Yard	1 - 4 All Zones simultaneously with activated prioritization circuitry	Essential Cooling Water In (ECWIS) Circulating Water Intake S (CWIS) Lighting Diesel Generator Load Center Buildings 12J 12M and the Electrical Loa Building (EL) Hypochlorination Make Up Demineralizer (M South/East Load Center Bu Fire Pump House North, East and West Gate Units 1 and 2 Main and Sta Transformer Emergency Transformer Fuel Storage Building Low Level Waste Building CWS Load Center Warehouse and Machine S Units 1 & 2	tructure Building (LD) , 12K, 12L, ad Center (IUD) uilding Houses andby	Nuclear Support Center (NSC), Nuclear Training Facility (NTF) Owner Controlled Area
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Emergency Communications							
Addendum 4	Related Maintenar	nce Jacks	Page 1 of 1				
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		UNIT 1	UNIT 2				
TRANSFER SWITCH PANEL	TRAIN A	ESF1	ESF1				
TRANSFER SWITCH PANEL	TRAIN A	ESF2	ESF2				
FRANSFER SWITCH PANEL	TRAIN B	ESF8	ESF3				
TRANSFER SWITCH PANEL	TRAIN B	ESF9	ESF9				
RANSFER SWITCH PANEL	TRAIN C	ESF10	ESF10				
FRANSFER SWITCH PANEL	TRAIN C	ESF11	ESF11				
STANDBY DIESEL GENERATOR	TRAIN A	1SDG3	2SDG3				
CONTROL PANEL							
TANDBY DIESEL GENERATOR	TRAIN B	1SDG2	2SDG2				
STANDBY DIESEL GENERATOR	TRAIN C	1SDG1	2SDG1				
CONTROL PANEL	· · ·						
CHILLER CONTROL PANEL,		TGI-17	TGI-17				
COLUMN 18V							
BORIC ACID TANK ROOM		RW-16	RW-16				
ELE. 29' MAB, ROOM 076							
CCW SURGE TANK ROOM		MA-18	MA-18				
ELE. 60' MAB			<u> </u>				
ESSENTIAL CHILLED WATER	TRAIN A	1YD5	2YD8				
NTAKE STRUCTURE	<u></u>						
ESSENTIAL CHILLED WATER	TRAIN B	1YD6	2YD9				
NTAKE STRUCTURE							
ESSENTIAL CHILLED WATER	TRAIN C	1YD7	2YD10				
NTAKE STRUCTURE	<u></u>						
AUXILIARY FEEDWATER STOR	AGE	TGI-12	TGI-12				

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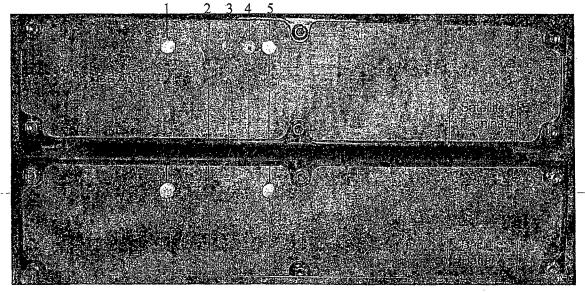
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REMOTE SATELLITE TERMINAL (RST-100)





Status LEDS are located on the front panel, they show the RST-100 status.

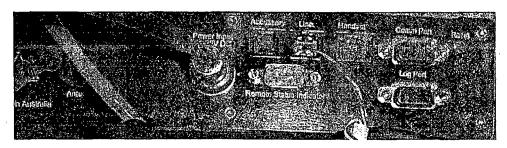
1. Power 2. Voicemail waiting 3. SMS waiting 4. Call status 5. Signal strength

Once powered up, the RST-100 attempts to register with the Iridium network. The signal level LED uses color to indicate how strong the Iridium signal is at your location.

Indicator color	Signal strength
Green	Strong
Orange	Acceptable
Red	No signal, problem with installation

In most cases the indicator will show green after a short period of approximately 15 seconds - orange indicates an acceptable but marginal signal strength. If the indicator remains red, there is a problem with the installation.

Rear Panel



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Emergency Communications					
Addendum 6 Desktop Satellite Telephone Troubleshooting Page 2 of 3					

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This table provides information to help you troubleshoot problems encountered while using the Desktop Satellite Telephone (RST-100). If the problem continues contact IT Communications at extension 7000 or if the ERO/Storm Crew is activated the Communications Systems Supervisor.

QUESTION	ANSWER
No lights on the front panel of the Satellite Terminal Box.	Check power is connected.
RST-100 fails to register with the Iridium service after 30 seconds.	Press reset button located on the rear of the Satellite Terminal Box.
-No dial tone	-Check if a data-call-is in-progress and power-is
Cannot make call, two tone signal heard.	Phone requires a PIN. See step 1 below.
You can't make calls.	Check that the antenna is properly mounted.
	Did you enter the number in international format?
	Check the signal strength. If the signal is weak, wait a few minutes for thick cloud cover to move.
·	Has a new SIM card been inserted?
You can't receive calls.	Check the antenna. Is it properly mounted?
	Check the signal strength. If the signal is weak, wait a few minutes for thick cloud cover to move.
	Check the telephone ringer setting to see if it is off.
The Voicemail indicator keeps flashing.	There is not enough memory available to store another message. Or there is a message waiting. Delete messages and free up some space.

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Addendum 6	Desktop Satellite Telephone Trouble	eshooting	Page 3 of 3

1. Personal Identification Number (PIN):

Your RST-100 may require a PIN, this will be indicated by the Signal light flashing Red and a distinctive dial tone consisting of two alternating tones.

If a distinctive, two-tone dial tone is heard, one of two access codes is required - the SIM PIN or the PIN Unlock Code (PUK).

- The SIM PIN is required if the two tones are of equal length. If so simply enter the four digit PIN (1111) and await a change of tone (up to ten seconds), then hang up. If the PIN was correct the phone will register and you may proceed with normal use as described below.
- The PUK is required if the PIN-has been incorrectly entered three times and is indicated when the high tone is longer than the low tone. Contact communications at extension 7000 or if the ERO/Storm Crew is activated the Communications Systems Supervisor
- 2. Voicemail
 - If a Voicemail message has been left for you, the RST-100 flashes the Voicemail Waiting indicator. The indicator is cleared whenever the user connects to the Voicemail retrieval number.
- 3. Facsimile Support

• The Iridium network does not support facsimile transmission.

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OPOP()4-RC-0003	Ex	cessive RCS I	Leakage	Rev	7.18	Page 5 o	of 127
EP	ACTIONS	S/EXPECTED	RESPONSE	RESP	ONSE NO)T OE	BTAINE	D
			NOTE					
	Si	tep 3.0 will dete	ermine if leakage	is actual RCS	leakage.	<u> </u>		
3.0		nds For Any O of RCS Leakag	of The Following e:	Go TO Ster	o 5.0.			
	Rad Mor	nitor RT8011 Pa	articulate – Rising	5				• .
	Reactor C	Coolant Drain Ta	nk Level – Rising					
	• Pressurize	er Relief Tank L	evel – Rising					
	RCB Nor	mal Sump Level	-Rising					
					· · · · · · · · · · · · · · · · · · ·			
4.0		Dne Of The Fo he RCS Leak H						
	• 0PSP03- Inventor	RC-0006, Read y	ctor Coolant					
		OR						
	pressuriz	MINE the RCS zer level, VCT l ng charging and		•				
	· · ·							
	•	· · ·		: .				
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0POP04-F	RC-0003	Exces	sive RCS Leakage	Rev. 1	8 Page 53 of 127	
Addendu	m 9		Basis	l I	Basis Page 5 of 77	1 1
	STEP 2	DESCRIPTION	N FOR 0POP04-RC-0003	STEP 3.0		
STEP: CH	IECK Trends	For Any Of The	Following Indications Of	RCS Leakage	:	
•		r RT8011 Partici	-	-		
•		ant Drain Tank Le	- · · ·			
•		elief Tank Level -	-			
•		Sump Level – Ri	- ·			
		•				
PURPOSE	E: To determi	ne if leakage is f	rom RCS and not CVCS.			
		č				
DACIC, T	diantian of D	TOALL DODT T	DT DCD Manuel O	1		
			PRT or RCB Normal Sump VCS which is normally tied		will confirm	
that the lea	akage is from	RCS and not CV	VCS which is normally tied	to the RCS.	will confirm	
that the lea	akage is from	RCS and not CV	VCS which is normally tied 1, RCDT, PRT or RCB No	d to the RCS.		
that the lease <u>ACTIONS</u> <u>INSTRUM</u>	akage is from <u>3</u> : Monitor tre <u>4ENTATION</u>	RCS and not CV nds from RT801 : Level indicatio	VCS which is normally tied	d to the RCS. ormal Sump. various plant c		
that the less <u>ACTIONS</u> <u>INSTRUN</u> monitors l	akage is from <u>5</u> : Monitor tre <u>AENTATION</u> ocated in con	RCS and not CV nds from RT801 : Level indicatic trol room. Radia	VCS which is normally tied 1, RCDT, PRT or RCB No ons located on CP004 and v	d to the RCS. ormal Sump. various plant c		
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 7.43 MONITOR Plant Computer grou points for the applicable pump: "RHR PUMP 1A(2A)" 	emoval System Ope p RH-12 (8412) <u>OR</u> RHFE0867 RHIA0880		wing
points for the applicable pump:"RHR PUMP 1A(2A)"	RHFE0867	TREND the follo	wing
· · · · ·			
• "RHR PUMP 1B(2B)"	RHFE0868 RHIA0881		
• "RHR PUMP 1C(2C)"	RHFE0869 RHIA0882		
CAU	TION		
centerline) <u>WHEN</u> the DG is being paralleled <u>OR</u> oper associated Trains "RHR PUMP" SHALL	NOT be started or op	erated: (CR 05-4	915)
7.44 <u>ENSURE</u> the associated train's E started in the next step is <u>NOT</u> be offsite power. (CR 05-4915)			
7.45 START the desired RHR pump:	:		· .
• "RHR PUMP 1A(2A)"			
• "RHR PUMP 1B(2B)"		·	
•	•	•	
• "RHR PUMP 1C(2C)"			

		0POP02-I	RH-0001	Rev. 63	Page 39 of 2:
		Residual Heat Ro	emoval System Op	eration	
7.43		R Plant Computer group the applicable pump:	o RH-12 (8412) <u>OR</u>	TREND the follo	owing
	• "R	HR PUMP 1A(2A)"	RHFE0867 RHIA0880		
	• "R	HR PUMP 1B(2B)"	RHFE0868 RHIA0881		
	• "R	HR PUMP 1C(2C)"	RHFE0869 RHIA0882		
•		CAU	TION	<u></u>	
• DO <u>NC</u> centerli		HR pump with vessel le	vel below 32 ft 9 in	ich. (6 inches abc	we hot leg
centerli • <u>WHEN</u>	ne) the DG is be	HR pump with vessel le eing paralleled <u>OR</u> oper CHR PUMP" SHALL <u>N</u>	ated in parallel with	n offsite power, <u>T</u>	HEN the
centerli • WHEN	ne) the DG is be ted Trains "F <u>ENSURE</u> started in	eing paralleled <u>OR</u> oper	ated in parallel with [<u>OT</u> be started or op mergency Diesel Go	n offsite power, <u>T</u> perated: (CR 05-4 enerator for the p	HEN the 915) ump to be
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	0POP04-AE-0001	First Response To Loss Of Any Or All 13.8 KV Or 4.16 KV Bus	Rev. 44	Page 29 of 58	States of the second second
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Addendum 4	Basis	Basis Pa
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PROCEDURE PURPOSE

This procedure provides guidelines for the initial response and stabilization of the plant in the event of a loss of any single or all 13.8 KV bus(es) or 4.16 KV Bus(es). This includes all 13.8 KV Auxiliary and Standby buses, and 4.16 KV buses with the exception of Buses 1K(2K), 1L(2L) and 1M. Loss of a 4.16 KV ESF bus is addressed as it indicates at least a partial loss of offsite and onsite AC power (ESF bus power can only be completely lost if both offsite and onsite power sources to the specific bus are lost).

MAJOR ACTION CATEGORIES

- Provide interface with Emergency Operating Procedures and provide the instructions to establish the minimum equipment required to safely stabilize the unit.
- Identify actions associated with commitments to perform the action within a specified time period
 after the initiating event.

DISCUSSION:

The electrical distribution system at STP has by design, a high degree of flexibility and ability to withstand casualties, especially the Class 1E alternating current systems. However throughout the nuclear industry Loss Of Offsite Power (LOOP) events have occurred as well as Station Blackout (loss of all offsite and onsite AC power) events.

When dealing with a loss of offsite AC power, both complete and partial, with the Unit in Modes 1 or 2, the loss of an Auxiliary power bus will result in the loss of a Reactor Coolant Pump requiring a reactor trip because STP is not analyzed for operation with only three Reactor Coolant Pumps. In the event that no ESF bus is available the indication is that all offsite and onsite AC power has been lost requiring transition to the Emergency Operating Procedures. Under these same conditions STP has committed to shed the Channel I Load Sequencer from its power supply within the first 30 minutes after the initiating event, and if the associated battery bank has a jumpered cell then all the loads on DP 1201 and DP 1204 will be shed except for QDPS and SG PORVs.

The initial response provided by this procedure is directed to the stabilization of critical plant parameters and then analyzes the extent of the loss of power. While this procedure does not identify the specific combination of buses that have been lost, it does identify the specific area of the power loss so that a procedure that is more specific to the method for power restoration can be referred to.

This Procedure is Applicable in all Modes

0]	POP04-AE-0001	A	To Loss Of Any (Or 4.16 KV Bus	Or Rev.	44 Page 34 of 58
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	STE	P DESCRIPTION	FOR 0POP04-AE-00	01 STEP 3.0	
	STEP: CHECK 4.16 I	KV ESF Bus Status:			
	a. ANY 4.16		nergized from offsite Bus).	power (VER	IFY the voltage
	b. VERIFY A	Applicable STBY DG	(s) running		
· ·	c. VERIFY A 4.16 KV E		(s) output breaker(s) c	closed to the a	ssociated
	<u>PURPOSE</u> : To determ actions that can be per			nd performs a	ny corrective
	<u>BASIS</u> : This step atte breaker is closed and and energize the bus.				
	If "4KV BUS O/C LC the bus cannot be ene			•	} is illuminated
	<u>ACTIONS</u> : Determin other fault protection. breaker.				
• • •	If the SDG is already breaker to energize th a fault does exist, the actuation device can b	e associated bus and a the cause of the fau	close the breaker in th lt would have to be co	e event that r	o faults exist. If
	INSTRUMENTATIC	<u>)N</u> : N/A			
	CONTROL/EQUIPM	<u>IENT</u> : N/A			
	KNOWLEDGE: If th differential lockout or and reset to energize	r an SDG overspeed l		-	-
			· ·		
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This Procedure is Applicable in all Modes

0POP04-AE-0004	Loss Of Power To One Or More 4.16 KV ESF Bus	Rev. 15	Page 77 of 95	
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PROCEDURE PURPOSE

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The purpose of this procedure is to restore power to any ESF bus which is not energized. In the case where only one ESF bus is energized by a DG, and another one cannot be energized by the associated DG or offsite power, then steps are taken to operate breakers and disconnects to use the one running DG to supply key loads on another bus.

MAJOR ACTION CATEGORIES

- Tie the operating DG to another bus via the emergency switchgear bus 1L(2L).
- Energize at least one ESF bus from the Emergency Transformer.
- Control and load essential equipment on to the available ESF buses.

DISCUSSION:

STP has committed under specific conditions related to loss of offsite and onsite power to energize at least two ESF buses from a running DG in order to energize specific loads needed to extend station battery life or provide availability of ESF equipment that is electrically powered from one of two specific ESF buses.

This Procedure is Applicable in all Modes

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ESF Power Availability

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NOTE
There are 5 possible lineups on Data Sheet 2, 3, and 4 for 13.8 KV XFMRS in the DESIGNATED Class 1E 4160 VAC Bus Power Source Table that meet Technical Specification requirements for being a power source for the 4.16 KV Buses:
• (1) UAT supplying the AUX BUS and STBY BUS
• (2) UAT supplying the AUX BUS and UNIT 1 STBY XFMR supplying the STBY BUS
• (3) UNIT 1 STBY XFMR supplying the STBY BUS and the AUX BUS
• (4) UAT supplying the AUX BUS and UNIT 2 STBY XFMR supplying the STBY BUS
• (5) UNIT 2 STBY XFMR supplying the STBY BUS and the AUX BUS
 5.2 COMPLETE Required ESF Power Train Data Sheet 2 through 4 by performing the following steps. 5.2.1 RECORD actual breaker/disconnect positions for the 13.8 KV XFMRs, AUX BUS, STBY BUSES and from the 13.8 KV STBY BUS to the 480 VAC BUSES. RECORD "CLOSED" breaker/disconnect positions by drawing a line at an angle through the breaker.
• RECORD "OPEN" breaker/disconnect positions by drawing a CIRCLE around the breaker.
EMER XFMR UNIT AUX XFMR UNIT 1 STBY XFMR UNIT 2 STBY XFMR This example illustrates: (3) UNIT 1 STBY XFMR supplying the STBY BUS and the
AUX BUS

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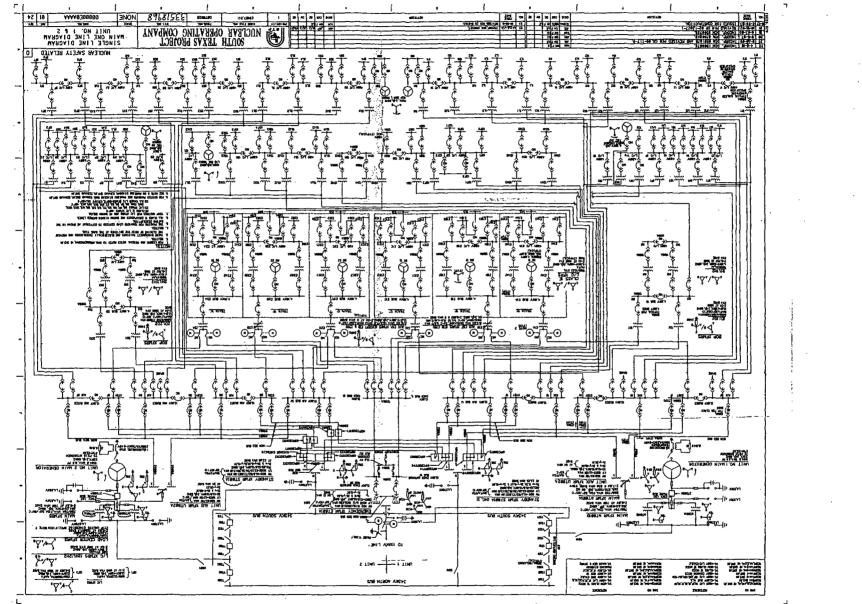
6.0 <u>Acceptance Criteria</u>

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	NOTE
• Ad off	dendum 2, Two Physically Independent Circuits, provides a drawing of rights of way and site circuits to aide in the definition of "two physically independent circuits".
	ss of one 13.8 KV Standby Bus to 4.16 KV ESF bus line constitutes loss of one required offsite arce. (Reference 8.2)
	ss of two 13.8 KV Standby busses to 4.16 KV ESF bus lines constitutes loss of two required site sources. (Reference 8.2)
• The	e preceding notes also apply when the 4.16 KV ESF bus is not energized by the 13.8 KV XFMR.
• Ste	p 6.1 applies during standby diesel inoperability.
• Ste	p 6.2 applies during offsite independent circuits inoperability.
	te and Precaution 3.28 should be referred to for additional clarification regarding allowable ication to be utilized when obtaining 345 KV switchyard voltage.
(5.1 Two physically independent circuits exist between the offsite transmission network and onsite Class 1E Distribution System as determined from Data Sheet 1, 2, 3, 4, and 9. (Technical Specifications 3.8.1.1.b, 3.8.1.1.f, and 4.8.1.1.1.a)
	• North and South Bus in service with bus voltage:
•	\circ ≥ 340 KV" for NORMAL LINEUP
	OR
	\circ ≥ 356 KV for NORMAL LINEUP with UAT or Train B ESF LTC in "MANUAL"
	OR
	 ≥ 358 KV for all ALTERNATE LINEUPs OR voltage specified in the "Minimum Voltage for Various Alternate 13.8 KV Bus Alignments" Addendum
	of 0POP02-AE-0002, Transformer Normal Breaker and Switch Lineup.
	• Two of the following Rights of Way with a 345 KV line are available:
	o NW Right of Way 1 (White Point 39)
	o NW Right of Way 2 (Elm Creek 27 <u>OR</u> WA Parish 39 <u>OR</u> Elm Creek 18)
	o Eastern Right of Way (Dow Velasco 27 <u>OR</u> Dow Velasco 18)
	• Two of the following 13.8 KV XFMRs are available:
	o Unit Aux XFMR
	o Unit 1 Stby XFMR
	o Unit 2 Stby XFMR
	• Three 13.8 KV Standby Buses energizing the 4.16 KV ESF bus lines.
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TABLE 1.2

OPERATIONAL MODES

, tekna to	- <u>MO</u> D) <u>E</u>	REACTIVITY CONDITION, K _{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
a	1.	POWER OPERATION	<u>≥</u> 0.99	> 5%	<u>></u> 350°F
	2.	STARTUP	<u>></u> 0.99	<u>≤</u> 5%	≥ 350°F
	3.	HOT STANDBY	< 0.99	0	≥ 350°F
	4.	HOT SHUTDOWN	< 0.99	0	350°F > T > 200°F avg
	5.	COLD SHUTDOWN	< 0.99	0	<u><</u> 200°F
·	6.	REFUELING**	<u><</u> 0.95	0	<u>≤</u> 140°F

*Excluding decay heat.

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**Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

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3.4 WATER LEVEL (FLOOD) DESIGN

The methods of analysis used to determine the design basis flood are discussed in Section 2.4. These methods are consistent with the requirements of Regulatory Guide (RG) 1.59.

The protection measures used to accommodate static and dynamic flood loads on Category I structures generally fall under the category of "incorporated barriers" as specified in regulatory position C.1 of RG 1.102.

3.4.1 Flood Protection

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3.4.1.1 <u>External Flood Protection Measures for Seismic Category I Structures</u>. The flooding due to a postulated Main Cooling Reservoir (MCR) embankment breach produces the maximum water level around the power block structures as well as the controlling water elevations for buoyancy calculations. This is also the controlling phenomena in determining the maximum water level at the Essential Cooling Water Intake Structure (ECWIS). Studies and analyses on the MCR embankment have demonstrated that an adequate margin of safety can be maintained for all credible failure mechanisms (Section 2.5.6). Accordingly, mechanistic effects (such as scour and erosion) associated with a postulated failure of the MCR embankment need-not-be evaluated.

The maximum water level on a vertical face at the south end of the plant structures is El. 50.8 ft mean sea level (MSL), which is El. 22.8 ft above plant grade. This maximum elevation occurs during a quasi-steady-state condition after a breach of the MCR embankment and is based on an instantaneous removal of approximately 2,000 ft of the embankment opposite the power block structures. This maximum elevation occurs on the south face of the Fuel-Handling Building (FHB) of Unit 1. The selection of postulated embankment breach widths and the assumptions made in determining the maximum flood elevations are described in Section 2.4.4.

Total inundation of the Essential Cooling Pond (ECP) occurs only under the condition of MCR embankment breach and does not affect the safe shutdown capability of the plant. The maximum water level calculated to occur at the ECWIS is El. 40.8 ft.

Safety-related structures, systems and components listed in Table 3.2.A-1 are protected against the effects of external flooding by:

1. Being designed to withstand the maximum flood level and associated effects and remain functional (such as seismic Category I structures and the Category I auxiliary feedwater storage tank) or

2. Being housed within seismic Category I structures which are designed as in item 1, above.

Flood protection of safety-related structures, systems, and components is provided for postulated flood levels and conditions described in Section 2.4.

Seismic Category I structures are designed to withstand the maximum flood levels by:

3.4-1

Having external walls and slabs of structures designed to resist the hydrostatic and hydrodynamic forces associated with surge-wave runup and steady-state water level.

2. Ensuring the overall stability of the total structure against overturning and sliding due to the hydrostatic and hydrodynamic forces associated with surge-wave runup and steady state water level, and

3. Ensuring that the total structure will not float due to buoyancy forces.

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Figure 3.4-1 shows a general section through the plant. Figure 3.4-2 shows the seismic Category I Building maximum steady-state water surface profile, and the corresponding relationship of sill elevations for entrances to seismic Category I buildings.

Table 3.4-1 shows the results of hydraulic loading and buoyancy calculations which were done for the various safety-related facilities. The water depths shown on this table were developed from the maximum water surface elevations presented in Table 2.4.4-3.

An investigation of seismic Category I structures has been made for the flood levels and associated effects as previously described. The design for gross effects upon the structure incorporates safety factors greater than 1.1. All exterior seismic Category I bullding openings are located above the maximum steady-state flood level or are equipped with watertight doors when located below this profile, except as stated below.

Exceptions to the above-stated design basis for exterior building openings in seismic Category I structures are: (1) the opening for the truck bay in the radwaste loading area of the Mechanical-Electrical Auxiliaries Building (MEAB) and (2) the opening for the rail car access in the spent fuel cask loading area of the FHB. These areas are not protected from flooding because they do not have any safety-related systems and components located near or below the maximum flood level which is required to perform any essential function. In addition, the two areas are separated from the remainder of the building by walls which do not contain openings below the maximum water surface elevation corresponding to their location. The Tendon Gallery Access Shaftcover (TGAS) is provided with a watertight cover to prevent flood waters from entering the MEAB.

The safety-related equipment in the ECWIS is protected from the effects of the design basis flood. The personnel access doors on the west wall are provided with watertight doors; all other doors and openings are above the flood level. The dividing walls and doors between the ECWIS compartments minimize the potential for the propagation of flooding from one compartment to another.

The three maintenance knockout panels in the exterior walls of the Diesel-Generator Building (DGB), which are located below the maximum water surface elevation of 45.0 ft MSL, are watertight and designed for the hydrostatic forces. Each knockout panel allows access to only one of the three separate compartments within the structure, and only one panel may be removed at one time. The dividing walls between the compartments preclude propagation of flooding from one compartment to another.

The maintenance knockout panels in the exterior wall of the room, housing the component cooling water heat exchangers in the MEAB are located below the maximum steady-state water level shown on Figure 3.4-2. These panels are watertight. Since mechanistic effects from the MCR breach need not be evaluated, there is adequate time to replace the knockout panels for the remaining flood events of concern.

All exterior seismic Category I building wall and slab surfaces below grade are waterproofed. This conservatively protects the substructure of seismic Category I buildings from groundwater, which is expected to stabilize between El. 17 ft and 26 ft (1 to 10 ft below grade) after decommissioning of the dewatering system. No waterproofing is provided on exterior wall or slab surfaces above grade to protect against the effects of surge-wave run-up because of its short duration. All construction joints in exterior walls and slabs (except for localized areas of blockouts) are provided with waterstops to the maximum flood level for that location and can withstand hydrostatic and hydrodynamic effects.

All seismic joints between Category I structures contain dual 9-in. water stops capable of withstanding potential seismic and hydrostatic effects. Cracks in concrete are minimized by imposing strict QA and QC procedures on the quality of concrete and construction techniques.

Drains are provided with check valves such that the external flooding would not result in internal flooding through the inadvertent introduction of water through these drains into seismic Category I structures.

The duct banks are sealed so as to prevent backflow into safety-related areas. The cable in the duct banks is designed/specified for submerged installations.

Leakage from groundwater into the FHB is prevented by the use of waterproofing on exterior wall and slab surfaces located below grade. Should groundwater inleakage occur, it is handled by the pumps in the FHB sump, the three-train compartment sumps, and the transfer cart area sump. For Unit 1 only, accumulated groundwater inleakage to the 64 degree tendon buttress area drains through a penetration in the RCB tendon gallery outer wall and is collected in the tendon gallery sump.

Leakage of groundwater into the MEAB is prevented by the use of waterproofing on exterior wall and slab surfaces located below grade. Should groundwater leakage occur, it will be collected in sumps. Discharge from non-radioactive sumps are routed to the reservoir via a circulating water discharge line. Potentially radioactive discharge is pumped to the Liquid Waste Processing System (LWPS).

3.4.2 Analysis Procedures

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3.4.2.1 <u>Phenomena Considered in Design Load Calculations</u>. For external flooding, the design basis events considered in design load calculations are as described in Section 3.4.1.

3.4.2.2 <u>Flood-Force Application</u>. The design flood conditions and elevations have been determined from an analysis of the phenomena discussed in Section 3.4.1.1.

In order to establish the controlling load conditions resulting from the embankment breach, both instantaneous surge wave runup as well as the longer term, quasi-steady-state conditions were analyzed. The wave runup condition conservatively assumes that the maximum total force perpendicular to the south face of the plant structures includes a dynamic component in addition to the associated hydrostatic forces. The quasi-steady state condition assumes that only the hydrostatic component contributes to the development of the total force for this case. The latter condition resulted in higher water surface elevations and greater hydraulic loads on power block structures.

The vertical buoyant loading condition is the force equal to the weight of water displaced by a structure. The discussion of lateral and vertical loadings is presented in the following subsections. Table 3.4-1 shows a summary of different lateral loadings at various locations around plant and ECP structures, caused by their respective controlling flood conditions. Procedures used to determine flood loadings are identified in Sections 3.4.2.2.1 and 3.4.2.2.2.

3.4.2.2.1 Lateral Loading:

3.4.2.2.1.1 <u>Lateral Loading on the Power Block Structures</u> – The analysis of the lateral force on the power block structures considered both the instantaneous wave runup and the quasisteady state conditions. This analysis determined that the maximum total lateral force on the power block structures occurs when the maximum water level is reached during the quasi-steady state condition. Table 3.4-1 shows the controlling lateral forces (hydrostatic) exerted on different power block structures. These lateral forces are treated as triangular loadings on a vertical surface, varying at a rate of 62.4 lb/ft2/ft of structure depth. The procedures used to determine the dynamic and hydrostatic loadings for the above analysis conditions are discussed below:

1. Dynamic Force

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The dynamic force on the south side of the power block structures is determined by application of linear momentum principles. The flow from the MCR is assumed to be normal to the south side of the power block structures. Therefore, the dynamic force exerted on the structures can be expressed by the following momentum equation (Ref. 3.4-2);

$F = p \cdot Q \cdot V_{0}$

where:

F = dynamic force normal to plant structure<math>p = density of flow Q = flow rate $V_0 = velocity of flow$

The maximum value of pQv_0 during surge formation is calculated. This is the contribution of momentum flux to the dynamic force. The contribution of the unsteadiness of momentum field is insignificant.

2. Hydrostatic Force

3.4-4

The lateral hydrostatic force is determined by the following equation (Ref. 3.4-2):

 $F_{Hyd} = \frac{1}{2} \gamma_w h^2$

where:

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 F_{Hvd} = hydrostatic force, lb/ft of width

h = water depth, ft

 $\gamma_{\rm w}$ = unit weight of water, lb/ft³

3.4.2.2.1.2 <u>Lateral Loading on the ECWIS and the South ECP Embankment</u> – The determination of the maximum lateral force on the ECWIS considered both instantaneous and quasisteady-state conditions. The maximum total force on the ECWIS is a result of the MCR embankment breach discussed in Section 2.4.4.2.2. This force is the result of a water elevation of 41.0 ft mean sea level during the quasi-steady state condition.

Since the south BCP embankment crest elevation is 34.0 ft MSL, it would be overtopped by the flood wave resulting from the MCR embankment breach. The south ECP embankment is designed to withstand the lateral force based on the maximum water elevation resulting from MCR embankment breach.

3.4.2.2.2 <u>Vertical Loading</u>: The roofs of seismic Category I structures are designed to withstand the weight of the accumulated PMP, assuming completely clogged drains (Section 2.4.2.3).

Table 3.4-1 shows the elevations of maximum water surface used for buoyancy calculations. The maximum buoyant force is calculated by assuming that the granular backfill around the structures is completely saturated so that the buoyant force will occur as soon as water arrives at the plant area.

4.3-Internal-Flood-Protection-

3.4.3.1 <u>Protection Features</u>. Safety-related systems, components and structures are protected such that the plant can achieve and maintain a safe shutdown condition and prevent unacceptable radiological releases to the environment.

In general, the plant layout arrangement is based on maximizing the physical separation of redundant or diverse safety-related components and systems from each other and from nonsafety-related items. Therefore, there is minimal effect on other systems or components which are required for safe shutdown of the plant or to mitigate the consequence of internal flooding.

Where separation is not feasible, other protection features are employed. These protection features include the following:

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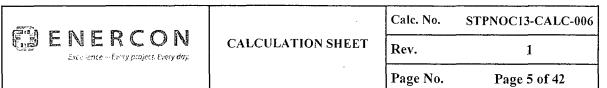
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1. Purpose and Scope

The purpose of this calculation is to evaluate dose rates as a function of water height in the reactor vessel during refueling operations in order to set Emergency Action Level (EAL) thresholds for core uncovery. The dose rates are calculated at the locations of the containment monitors RE-8055 and RE-8099 so that dose rate measurements by these devices can be used to estimate water level in the core, upon failure of other water level detection systems. This evaluation will calculate the dose rate at full core uncovery, as well as maximum water levels with a detectable dose rate response. Since the scope of this calculation concerns uncovering the reactor core, the effects of future fuel element storage in the nearby Fuel Storage Pit are not analyzed, since it's effects are negligible in comparison. The containment building, components within the building, and the reactor vessel and contents are modeled simplistically because only order of magnitude results are needed. As such, the dose rate results should be considered as reasonably representative of the magnitude of the actual dose rate only.

2. Summary of Results and Conclusion

The dose rate results for the configuration without the reactor vessel head and with the reactor vessel head are provided in Section 7.7.1 and Section 7.7.2, respectively. The dose rate with the core uncovered (i.e. water at the top of the active length) is 2.23E+04 mrem/h with the head in place and 9.30E+06 mrem/h with the head removed. Detailed results of the dose rate as a function of water height are provided in Figure 7-13 with the head removed and Figure 7-14 with the head attached.

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3. References

- 1. "Standard Composition Library," ORNL/NUREG/CSD-2/V1/R6, Volume 3, Section M8, March 2000.
- 2. Calculation NC-6510. "Core Radionuclide Inventory for Chapter 15 Accident Analysis."
- 3. RSICC Code Package CCC-750, "SCALE 6.0: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation", Jan. 2009.
- 4. "ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms", I.C. Gauld, O.W. Hermann, & R. M. Westfall. Jan. 2009.
- 5. STP001-CPC-001. Computer Program Certification MCNP5 Version 1.4 and SCALE 6.0.
- 6. ENERCON email from Paul Sudnak, dated December 9, 2013. (Appendix A).
- 7. Drawing 6C-18-N-5006, Rev. 9. "General Arrangement Reactor Containment Building Plan at El. 68' 0" Area G."
- 8. Drawing 6C-18-9-N-5007, Rev. 6. "General Arrangement Reactor Containment Building Section A-A Area G."
- Drawing 6C-18-9-N-5008, Rev. 8. "General Arrangement Reactor Containment Building Section B-B Area G."
- 10. RSICC Code Package CCC-730, "MCNP/MCNPX Monte Carlo N-Particle Transport Code System 12 Including MCNP5 1.40 and MCNPX 2.5.0 and Data Libraries," January 2006.
- 11. ANSI/ANS 6.1.1-1977, Neutron and Gamma Flux-To-Dose Conversion Factors.
- 12. ENERCON email from Paul Sudnak, dated February 3, 2014 (Appendix A).
- 13. Drawing L5-01EM101, Rev. 1. "Closure Head General Assembly."
- 14. Drawing 1142E24. "Model 4XLR Reactor 173 in. I.D. Vessel."
- 15. Drawing 2C26-9-S-1004, Rev. 4. "Steel Reactor Containment Building Cylindrical Shell Liner Sects. And Dets. Unit N° 1 & 2."
- 16. Drawing 1211E6. "4 Loop Rapid XL Reactor General Assembly."

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4. Assumptions

The following assumptions are used in the core uncovery dose rate calculation:

- 1. The core is homogenized based on the typical Vantage 5 fuel assembly dimensions, taking into account the fuel rods and space between. Any small variations in fuel parameters will have a negligible effect on containment dose rates.
- 2. Any non-fuel hardware is ignored since the primary self-shielding occurs in the fuel itself, and there may be some unknown streaming effects through the non-fuel hardware. This homogenization takes into account the water level when calculating the isotopic weight fraction and homogenized density.
- 3. The source term for this evaluation is based on the fission product inventory at the time of shutdown. Because there is no cooling time, the fuel gamma source term will predominate and the N-gamma and hardware activation can be neglected.
- 4. The compositions of the containment structure and components are based on the values in the SCALE standard composition library [1].
- 5. The RE-8055 and RE-8099 monitors are assumed to be 5 feet above the 68 foot level in order to take into account the mounting device.
- 6. The containment outer concrete thickness is modeled as 3 feet thick. Because the backscattering off the containment walls is due to the steel liner, this dimension has a negligible impact on dose rates near the reactor vessel.

5. Design Inputs

5.1 Fuel Assembly Parameters

The following fuel assembly parameters are used in the core homogenization in the MCNP model. They are based on typical fuel assembly values for Westinghouse Vantage 5 fuel.

Parameters	Value	Unit	Reference
	Westinghouse		Assumption 1
Fuel Type	Vantage +		
# Fuel Rods per Assy	264		Assumption 1
Assembly Array	17x17		Assumption 1
Enrichment	4	wt %	Assumption 1
Density (% of theoretical)	0.95		Assumption 1
Fuel Pellet OD	0.3225	[in]	Assumption 1
Fuel Rod Pitch	0.496	[in]	Assumption 1

Table 5-1 Design Input Fuel Assembly Parameters for Westinghouse Vantage 5 Fuel

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Parameters	Value	Unit	Reference	
Fuel Rod OD	0.374	[in]	Assumption 1	
Clad Thickness	0.0225	[in]	Assumption 1	
Guide Tube OD	0.482	[in]	Assumption 1	
Guide Tube Thickness	0.020	[in]	Assumption 1	
# Guide Tubes	24		Assumption 1	
Instrument Tube OD	0.482	[in]	Assumption 1	
Instrument Tube Thickness	0.020	[in]	Assumption 1	
# Instrument Tubes	1		Assumption 1	
Active Length	14	[ft]	Assumption 1	

5.2 Containment Dimensions

The following dimensions are based on drawings of the STP containment building and equipment. Some parameters are estimated using scaling when the drawings do not detail the exact dimension. These estimations are only applied to dimensions that have a negligible effect on the core uncovery dose rate analysis.

Dimension:	ft.	in	cm	reference
Reactor Pressure Vessel				
Elevation at top of active fuel	28	2	858.52	[6]
Elevation at head level platform	38	6.5	1174.75	[8]
Elevation at full water level in				
refueling cavity	66	6	2026.92	[8]
Closure head thickness	0	7.19	18.2626	[13]
Reactor pressure vessel inside				
diameter at shell	0	173	439.42	[14]
Height of reactor vessel from				
bottom of fuel to head level			742.95	Calculated
Steam Generator				
Elevation at bottom of SG	38	4	1168.4	[9]
Elevation at top of SG	105	9.875	3225.4825	[9]
Total SG height			2057.0825	Calculated
SG outer diameter			500	[7] Scaled

Table 5-2 Design Input Containment Dimensions

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Dimension:	ft.	in	cm	reference
Active Fuel				ηφηματώ το το τομήτα το το πολλάτερα το ποιοποτείδανο το το
Active fuel bottom elevation	12	1	368.3	[9]
Active fuel height	14	0	426.72	[14]
Concrete Wall				
Lower Height	38	6.5	1174.75	[9]
Upper Height	85	0	2590.8	[9]
Overall Height			1416.05	Calculated
Thickness	2	0	106	[7] Scaled
Width			874.776	[7] Scaled
Length			2499.36	[7] Scaled
Steam Generators				
Lower Modeled Height	85	0	2590.8	[9]
Upper Modeled Height	105	9.875	3225.4825	[9]
Overall Modeled Height			634.6825	Calculated
Diameter			500	[7] Scaled
Containment				
Upper modeled height	153	0	4663.44	[8]
Lower modeled height	68	0	2072.64	[8]
Net Height			2590.8	Calculated
Inner Diameter	149	11 ¹ /4	4570	[15]
Liner Thickness	0	0.375	0.9525	[15]
Dome Inner Radius	74	115/8	2285	[15]
Concrete Thickness	3	0	91.44	Assumption 6

5.3 Core Isotopic Inventory

Core isotopic activities are provided in Table 11 of [2]. The isotope specific activities are given in terms of Ci/MWt, which is converted to curies based on the total core thermal power of 4,100 MWt [2]. These calculations are performed in EXCEL spreadsheet *STP.xlsx*. A table of the input values is shown in Table 5-3, below.

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Table 5-3 Design Basis Core Shutdown Source Term¹

Isotope	Ci/MWt	Ci	Isotope	Ci/MWt	Ci
Kr83m	3.41E+03	1.40E+07	Ru106	1.34E+04	5.49E+07
Kr85m	7.07E+03	2.90E+07	Rh105	3.05E+04	1.25E+08
Kr85	2.93E+02	1.20E+06	Zr95	4.39E+04	1.80E+08
Kr87	1.34E+04	5.49E+07	Zr97	4.39E+04	1.80E+08
Kr88	1.90E+04	7.79E+07	Nb95	4.32E+04	1.77E+08
Kr89	2.32E+04	9.51E+07	La140	4.63E+04	1.90E+08
Xe131m	2.68E+02	1.10E+06	La141	4.62E+04	1.89E+08
Xe133m	1.66E+03	6.81E+06	La142	4.15E+04	1.70E+08
Xe133	5.37E+04	2.20E+08	Pr143	3.90E+04	1.60E+08
Xe135m	1.02E+04	4.18E+07	Nd147	1.73E+04	7.09E+07
Xe135	1.34E+04	5.49E+07	Am241	2.75E+00	1.13E+04
Xe137	4.63E+04	1.90E+08	Cm242	1.05E+03	4.31E+06
Xe138	4.39E+04	1.80E+08	Cm244	6.17E+01	2.53E+05
1131	2.59E+04	1.06E+08	Ce141	4.39E+04	1.80E+08
1132	3.71E+04	1.52E+08	Ce143	4.15E+04	1.70E+08
1133	5.37E+04	2.20E+08	Ce144	3.41E+04	1.40E+08
1134	5.85E+04	2.40E+08	Np239	5.12E+05	2.10E+09
1135	4.88E+04	2.00E+08	Pu238	8.71E+01	3.57E+05
Sb127	3.05E+03	1.25E+07	Pu239	1.96E+01	8.04E+04
Sb129	8.29E+03	3.40E+07	Pu240	2.48E+01	1.02E+05
Te127m	4.32E+02	1.77E+06	Pu241	4.17E+03	1.71E+07
Te127	3.05E+03	1.25E+07	Rb86	9.92E+01	4.07E+05
Te129m	1.22E+03	5.00E+06	Cs134	5.37E+03	2.20E+07
Te129	8.05E+03	3.30E+07	Cs136	1.54E+03	6.31E+06
Te131m	3.66E+03	1.50E+07	Cs137	3.17E+03	1.30E+07
Te132	3.82E+04	1.57E+08	Y90	3.56E+03	1.46E+07
Ba137m	2.93E+03	1.20E+07	Y91	3.41E+04	1.40E+08
Ba139	4.98E+04	2.04E+08	Y92	3.41E+04	1.40E+08
Ba140	4.63E+04	1.90E+08	Y93	3.90E+04	1.60E+08
Mo99	4.83E+04	1.98E+08	Sr89	2.68E+04	1.10E+08
Tc99m	4.07E+04	1.67E+08	Sr90	2.37E+03	9.72E+06
Ru103	3.90E+04	1.60E+08	Sr91	3.17E+04	1.30E+08
Ru105	2.68E+04	1.10E+08	Sr92	3.41E+04	1.40E+08

 1 Ci = Ci/MWt × 4,100 MWt

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5.4 Material Compositions

The following compositions used in the MCNP model are taken from the SCALE standard composition library [1] and are shown in Table 5-4.



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Table 5-4 SCALE Standard Compositions used in MCNP Model

Material	Isotope	Weight Fraction	Reference
Zry-4	Zr	0.9823	[1]
(6.56 g/cm^3)	Sn	0.0145	
	Cr	0.0010	
	Fe	0.0021	
	Hf	0.0001	
UO ₂	U-235	0.0353	[1]
$(10.412 \text{ g/cm}^3)^2$	U-238	0.8461	
	0	0.1186	
Air	С	0.0001	[1]
(1.21E-03 g/cm ³)	N	0.7651	1
	0	0.2348	
Water	Н	0.1111	[1]
(0.9982 g/cm^3)	0	0.8889	
SS-304	Fe	0.6838	[1]
(7.94 g/cm^3)	Cr	0.1900	
	Ni	0.0950	
	Mn	0.0200	
	Si	0.0100	
	С	0.0008	
	Р	0.0004	
Concrete	0	0.5320	[1]
(2.30 g/cm^3)	Si	0.3370	
	Са	0.0440	
	Al	0.0340	
	Na	0.0290	
	Fe	0.0140	
	Н	0.0100	
Carbon Steel	С	0.0100	[1]
(7.82 g/cm^3)	Fe	0.9900	1

 $^{^2}$ Based on 95% of theoretical density, Assumption 1.

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6. Methodology

The reactor source terms are computed with ORIGEN-S of the SCALE 6.0 code package [3, 4]. The ORIGEN-S decay sequence is used to bin design input isotope specific activities into energy dependent photon bins. These energy specific photon emission bins are used as input for the energy distribution described by the MCNP source definitions.

The ORIGEN-S sequence in the SCALE6.0 program package is verified for use in safety related calculations [5]. The program certification form is maintained in the project file.

MCNP5, release 1.40 [10], Monte Carlo transport is used to determine the dose rates. The ENDF/B-VI neutron cross section library, ENDF60, and the ENDF/B-VI Release 8 Photo-atomic Data gamma cross section library, MCPLIB04 are utilized in the transport computations. This software has been verified for use in safety related calculations [5].

The detailed engineering drawings are converted into MCNP surface and cell cards in the proper dimensions. The radiation monitors of interest are modeled as point detectors to determine the expected dose rate for those detectors. The dose rates are calculated as a function of water height for two reactor refueling conditions:

- 1. With Head the reactor is modeled with an 7.19 inch carbon steel plate as indicated in Table 5-2, which is additional attenuation between source and detector.
- 2. Without head the reactor is modeled with nothing between the active fuel zone and containment.

For low water levels, variance reduction is accomplished with a geometric importance map that is imposed on the homogenized core. Without significant amounts of water present, this is enough to calculate statistically sound dose rate results. Once the water depth reaches a height where the variance of the results reaches an unacceptable level, a superimposed weight windows mesh is utilized to improve the variance reduction of the simple geometric scheme. The weight windows are iteratively generated using the MCNP weight windows generator card with a mesh over the existing geometry. All final dose rates presented in this calculation include weight windows variance reduction.

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7. Calculations

7.1 Source Terms

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In order to convert the isotope specific activity into an energy spectrum, ORIGEN-S of the SCALE6.0 code package is used to initiate a decay and bin into 19 photon energy groups. The energy groups along with their associated activities are used in the MCNP source definition to model the anticipated radiation emission following shutdown.

The ORIGEN-S input deck, *STPEAL.inp*, is provided below in Figure 7-1. This input has a simple decay case where the inputted isotopic composition in curies is decayed. The isotope is specified in the 73\$\$ card using the special identifier described in Section F7.6.2 of the ORIGEN-S manual, and the activity in curies is specified in the 74** card. The time steps for the decay are given on the 60** card in years. Although multiple time steps are calculated, the source term with zero decay time is used in this calculation to model the core immediately after shutdown. The output of the decay is given in terms of photons/s/Energy-Group, which is automatically normalized in the MCNP input.

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Figure 7-1 ORIGEN-S Input Deck for MCNP Source Term Calculation

=origens ←Call Origen-S Sequence ←Logical Unit Assignments 0\$\$ all 71 e t -Binary Photon Library (71) PWR Source Term STP ELA Analysis ←Case Title 3\$\$ 21 1 1 a4 27 a16 4 a33 19 e t ←Library Integer Constants -- Units 83** Card Ci (4) -Gamma Energy Groups (19) 35\$\$ 0 t \leftarrow Not Used 54\$\$ a8 0 a11 2 e ← Special Calculation Options -Cutoff Value (Default) $-(\alpha,n)$ Composition Dependent 56\$\$ 0 6 a6 1 a10 0 a13 66 5 3 0 2 0 e ←Subcase Control Constants -Decay Only Subcase (0) -Number of Time Intervals (6) -Number of Nuclides (66) -Unit of Time in Years (5) 57** 0 a3 1-16 e ←Not Used ←Not Used 95\$\$ 0 t ←Subcase Title STPEAL ←Subcase Basis Ci Source Terms 60** 0 0.1 0.2 0.3 0.4 0.5 ←Time (years) 61** 5r1-8 1+6 1+4 ←Cutoff Values 65\$\$ ←Decay Period Print Triggers GRAM-ATOMS GRAMS CURIES WATTS-ALL WATTS-GAMMA 0 1 0 100 32 100 Зz 67 1 1 1 101 1 1 1 37. 6Z 3z37 3z 1 1 1 111 1 1 1 62 81\$\$ 2 0 26 1 e ←Gamma Source Constants 82\$\$ f2 ← Produces Gamma Source Spectrum 83** 1.10E+07 1.00E+07 8.00E+06 6.50E+06 5.00E+06 4.00E+06 3.00E+06 ←Gamma Energy Groups 2.50E+06 2.00E+06 1.66E+06 1.33E+06 1.00E+06 8.00E+05 6.00E+05 4.00E+05 3.00E+05 2.00E+05 1.00E+05 5.00E+04 1.00E+04 e 84** 2.00E+07 6.43E+06 3.00E+06 1.85E+06 1.40E+06 9.00E+05 4.00E+05 ←Neutron Energy Groups 1.00E+05 1.70E+04 3.00E+03 5.50E+02 1.00E+02 3.00E+01 1.00E+01 (Not Used) 3.05E+00 1.77E+00 1.30E+00 1.13E+00 1.00E+00 8.00E-01 4.00E-01 3.25E-01 2.25E-01 1.00E-01 5.00E-02 3.00E-02 1.00E-02 1.00E-05 e 73\$\$ 360831 360851 360850 360870 360880 360890 541311 541331 ←Nuclide Identifiers 541330 541351 541350 541370 541380 531310 531320 531330 531340 531350 511270 511290 521271 521270 521291 521290 521311 521320 561371 561390 561400 420990 430991 441030 441050 441060 451050 400950 450970 410950 571400 571410 571420 591430 601470 952410 962420 962440 581410 581430 581440 932390 942380 942390 942400 942410 370860 551340 551360 551370 390900 390910 390920 390930 380890 380900 380910 380920 74** 1.40E+07 2.90E+07 1.20E+06 5.49E+07 7.79E+07 9.51E+07 1.10E+06 ←Nuclide Concentrations (Ci) 6.81E+06 2.20E+08 4.18E+07 5.49E+07 1.90E+08 1.80E+08 1.06E+08 1.52E+08 2.20E+08 2.40E+08 2.00E+08 1.25E+07 3.40E+07 1.77E+06 1.25E+07 5.00E+06 3.30E+07 1.50E+07 1.57E+08 1.20E+07 2.04E+08 1.90E+08 1.98E+08 1.67E+08 1.60E+08 1.10E+08 5.49E+07 1.25E+08 1.80E+08 1.80E+08 1.77E+08 1.90E+08 1.89E+08 1.70E+08 1.60E+08 7.09E+07 1.13E+04 4.31E+06 2.53E+05 1.80E+08 1.70E+08 1.40E+08 2.10E+09 3.57E+05 8.04E+04 1.02E+05 1.71E+07 4.07E+05 2.20E+07 6.31E+06 1.30E+07 1.46E+07 1.40E+08 1.40E+08 1.60E+08 1.10E+03 9.72E+06 1.30E+08 1.40E+08 2-Actinide 3-Fission Product 3 3 t. 56\$\$ f0 t End

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The results of this calculation are summarized below in Table 7-1. These values will be used in the MCNP input source definition.

Enorgy	Energy	
Energy	Boundaries	Photons/sec
Group	(MeV)	
1	0.01-0.05	9.29E+19
2	0.05-0.1	2.93E+19
3	0.1-0.2	6.54E+19
4	0.2-0.3	4.28E+19
5	0.3-0.4	1.52E+19
6	0.4-0.6	3.58E+19
7	0.6-0.8	4.35E+19
8	0.8-1	2.66E+19
9	1-1.33	1.29E+19
10	1.33-1.66	1.65E+19
11	1.66-2	5.57E+18
12	2-2.5	5.53E+18
13	2.5-3	1.98E+18
14	3-4	7.81E+17
15	4-5	3.48E+16
16	5-6.5	3.95E+11
17	6.5-8	1.75E+08
18	8-10	3.71E+07
19	10-11	2.01E+06
totals		3.95E+20

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Table 7-1 Binned Total Core Source Term

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7.2 MCNP Model Core Homogenization

Because the source term is given for the entire core, the self-shielding from the assemblies is an important part of the dose rate response in regions above the core. Particles born in the lower section of the core are very unlikely to penetrate through the core itself, and make it to the radiation monitors. For simplicity, the core is modeled as a 3 dimensional cylinder with a uniformly distributed spatial particle distribution. The calculations for the homogenization are done in the worksheet *Compositions* of the EXCEL workbook *STP.xlsx*. A density and isotopic composition is calculated with the water level above the top of the fuel. A summary of the calculations for the core composition and density is shown below. The inputs are based on the dimensions in Table 5-1 and the compositions in Table 5-4.

Rod Volume = π (Pellet Radius)² × Active Length = (3.14)(0.16125 in)²(168 in) = 13.7 in³

Rod
$$Mass_{UO_2} = \rho \times V = \left(10.96 \frac{g}{cc}\right) (0.95)(13.72 \ in^3) \left(2.54 \frac{cm}{in}\right)^3 = 2341.5 \ g$$

Assembly $Mass_{UO_2} = Rod Mass \times \frac{Number of Fuel Rods}{Assembly} = (2341.5 g)(264) = 618.2 kg$

Clad Volume =
$$\pi \left(\frac{OD^2}{4} - \frac{ID^2}{4}\right) \times Active \ Length = (3.14) \left[\frac{(0.374 \ in)^2}{4} - \frac{(0.329 \ in)^2}{4}\right] (168 \ in)$$

= 4.17 in³

Rod Mass_{Zry-4} =
$$\rho \times V = \left(6.56 \frac{g}{cc}\right) (4.17 \text{ in}^3) \left(2.54 \frac{cm}{in}\right)^3 = 448.7 \text{ g}$$

Assembly
$$Mass_{Zry-4} = Rod Mass \times \frac{Number of Fuel Rods}{Assembly} = (448.7 g)(264) = 118.5 kg$$

Assembly $H_2OVolume = [(Assembly Width)^2 - \pi (Rod Radius)^2 \times 264] \times Active Length$ = [(8.404 in)² - (3.14)(0.187 in)²(264)](168 in) = 6993 in³

Assembly
$$Mass_{H_2O} = \rho \times V = \left(0.9982 \frac{g}{cc}\right) (6993 \ in^3) \left(2.54 \frac{cm}{in}\right)^3 = 114.4 \ kg$$

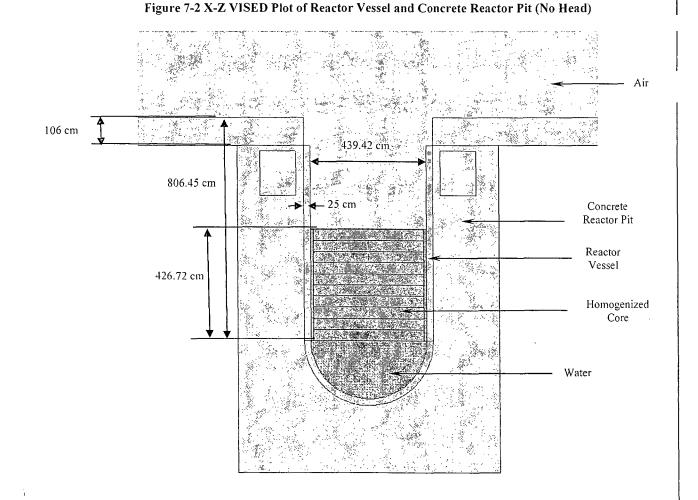
Assembly Volume = Active Length \times (Assembly Width)² = (168 in)(8.404 in)² = 11865.4 in³

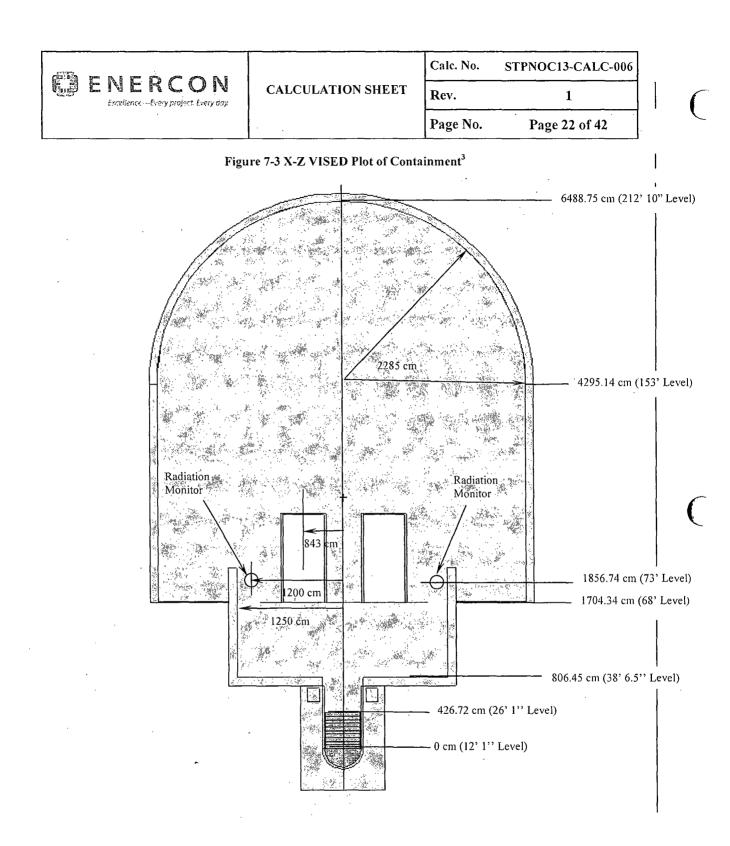
$$Density = \frac{Total \ Mass}{Volume} = \frac{1000(618.2 + 118.5 \ kg + 114.4) \ kg}{11865.4 \ in^3 \left(2.54 \frac{cm}{in}\right)^3} = 4.38 \ g/cc$$

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7.3 MCNP Model Geometry

The following MCNP model geometry is based on the containment dimensions summarized in Table 5-2. The model only focuses on the primary systems and components that provide shielding or reflection from the core to the radiation monitors. These components include the reactor vessel, concrete in reactor pit, containment walls (reflection), and steam generators (reflection). VISED plots of the model geometry are provided in Figure 7-2, Figure 7-3, and Figure 7-4. The MCNP surface cards with the model dimensions (cm) are shown in Figure 7-5, and the cell cards are shown in Figure 7-6 for the cases with no reactor head. A VISED plot of the model with the reactor head is shown in Figure 7-7. The surface and cell cards for the cases with the reactor head are shown in and Figure 7-8, respectively. Areas that are not of interest are given an importance of zero (white areas) so MCNP will not track particles in locations that will not contribute to the detector response. A summary of surfaces used in constructing this geometry is shown in Table 7-2, including a description of macrobody dimensions.

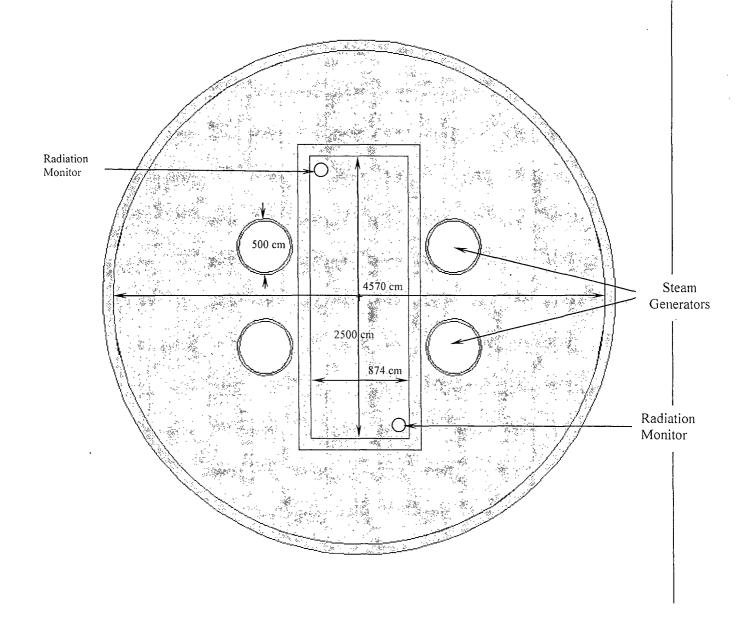




³ Steam Generators are not full height. Also, they are not on the same X-Z plane as the core shown above. They are included for visualization purposes.

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Surface	Surface						· · · ·		
Туре	Number		······		Dimensio	ns		·	Description
RCC		Xo	Yo	Z _o	V _x	Vy	V _z	R	
	1	0	0	. 0	0	0	426.72	209.71	Active Fuel Region
	2	0	0	0	0	0	700.45	219.71	Reactor Pressure Vessel Inner Surface
	3	0	0	0	0	0	700.45	244.71	Reactor Pressure Vessel Outer Surface
	31	0	0	700.45	0	0	18.26	244.71	Reactor Pressure Vessel Head
-	41	0	0	512.81	0	0	167.64	274.71	Concrete Void for Primary Loop
	42	0	0	512.81	0	0	167.64	411.71	Concrete Void for Primary Loop
	10	0	0	700.45	0	0	106	244.71	Concrete Wall Cutout
	11	444.71	843	700.45	. 0	0	2050	250	Steam Generator 1
	12	444.71	843	720.45	0	0	2010	230	Steam Generator Inner 1
	13	-444.71	843	700.45	0	0	2050	250	Steam Generator 2
	14	-444.71	843	720.45	0	0	2010	230	Steam Generator Inner 2
	15	-444.71	-843	700.45	0	0	2050	250	Steam Generator 3
	16	-444.71	-843	720.45	0	0	2010	230	Steam Generator Inner 3
	17	444.71	-843	700.45	0	0	2050	250	Steam Generator 4
	18	444.71	-843	720.45	0	0	2010	230 [.]	Steam Generator Inner 4
	21	0	0	1694.34	0	0	2600.8	2285	Containment Inner Liner Surface
	22	0	0	1694.34	0	0	2600.8	2285.95	Containment Inner Concrete Surface
	23	0	0	1694.34	0	0	2600.8	2377.39	Containment Outer Concrete Surface
RPP		-X	Х	-Y	Y	-Z	Z		
	4	-498	498	-498	498	-498	700.45		Concrete Surrounding RPV
	8	-1250	1250	-437	437	806.45	2116.45		Concrete Wall Fuel Pit Inner
	9	-1356	1356	-543	543	700.45	2116.45		Concrete Wall Fuel Pit Outer
SPH		Xo	Y ₀	Zo	R				

Table 7-2 Summary of Surfaces Used for MCNP Models

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Surface Type	ce Surface Number Dimensions		Description			
	5	0	0	0	219.71	Bottom of Reactor Pressure Vessel Inner
	6	0	0	0	244.71	Bottom of Reactor Pressure Vessel Outer
	24	0	0	4295.14	2285	Containment Dome Inner Liner Surface
	25	0	0	4295.14	2285.95	Containment Dome Inner Concrete Surface
	26	0	0	4295.14	2377.39	Containment Dome Outer Concrete Surface
PZ		Z				
	7	0				Fuel Bottom
	71	700.45				Top of RPV
_	20	Variable				Water Level
	27	4295.14				Spring Line
	28	1704.34				68' Level
	101-110	42.672	-	426.72		Geometric Importance Divisions in Active Zone

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Figure 7-5 MCNP Model Surface Cards⁴

c surfaces 1 rcc 0 0 0 0 0 426.72 209.71 \$ Active Fuel Region. 2 rcc 0 0 0 0 0 700.45 219.71 3 rcc 0 0 0 0 0 700.45 244.71 31 rcc 0 0 700.45 0 0 18.26 244.71 4 rpp -498 498 -498 498 -498 700.45 41 rcc 0 0 512.81 0 0 167.64 274.71 42 rcc 0 0 512.81 0 0 167.64 411.71 5 sph 0 0 0 219.71 6 sph 0 0 0 244.71 7 pz 0 71 pz 700.45 8 rpp -1250 1250 -437 437 806.45 2116.45 9 rpp -1356 1356 -543 543 700.45 2116.45 10 rcc 0 0 700.45 0 0 106 244.71 11 rcc 444.71 843 700.45 0 0 2050 250 12 rcc 444.71 843 720.45 0 0 2010 230 13 rcc -444.71 843 700.45 0 0 2050 250 14 rcc -444.71 843 720.45 0 0 2010 230 15 rcc -444.71 -843 700.45 0 0 2050 250 16 rcc -444.71 -843 720.45 0 0 2010 230 17 rcc 444.71 -843 700.45 0 0 2050 250 18 rcc 444.71 -843 720.45 0 0 2010 230 20 pz 365.76 21 rcc 0 0 1694.34 0 0 2600.8 2285 22 rcc 0 0 1694.34 0 0 2600.8 2285.95 23 rcc 0 0 1694.34 0 0 2600.8 2377.39 24 sph 0 0 4295.14 2285 25 sph 0 0 4295.14 2285.95 26 sph 0 0 4295.14 2377.39 27 pz 4295.14 28 pz 1704.34 101 pz 42.672 102 pz 85.344 103 pz 128.016 104 pz 170.688 105 pz 213.36 106 pz 256.032 107 pz 298.704 108 pz 341.376 109 pz 384.048 110 pz 426.72

\$ Reactor Pressure Vessel Inner Surface \$ Reactor Pressure Vessel Outer Surface \$ Reactor Vessel Head \$ Concrete Surrounding RPV \$ Concrete Void for Primary Loop \$ Concrete Void for Primary Loop \$ Bottom of Reactor Pressure Vessel \$ Bottom of Reactor Pressure Vessel \$ Bottom of Active Zone \$ Top of RPV \$ Concrete Walls Fuel Pit Inner \$ Concrete Wall Fuel Pit Outer \$ Concrete Wall Cutout \$ Steam Generator 1 \$ Inner Steam Generator 1 \$ Steam Generator 2 \$ Inner Steam Generator 2 \$ Steam Generator 3 \$ Inner Steam Generator 3 \$ Steam Generator 4 \$ Inner Steam Generator 4 \$ Water Elevation Surface \$ Containment Inner Liner Surface \$ Containment Inner Concrete Surface \$ Containment Outer Concrete Surface \$ Containment Dome Inner Liner Surface \$ Containment Dome Inner Concrete Surface \$ Containment Dome Outer Concrete Surface \$ Spring Line \$ 68' Level \$ Geometric Importance Division Fuel Zone
The surface cards for the MCNP model without the reactor vessel head does not have surface 31. The other surfaces are identical.

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Figure 7-6 MCNP Model Cell Cards (No Head)

101 1 -4.57 -1 -101 imp:p≃1 \$ Active Fuel Region 102 1 -4.57 -1 101 -102 imp:p=2 \$ Active Fuel Region 103 1 -4.57 -1 102 -103 imp:p=3 104 1 -4.57 -1 103 -104 imp:p=4 105 1 -4.57 -1 104 -105 imp:p=8 106 1 -4.57 -1 105 -106 imp:p=16 107 1 -4.57 -1 106 -107 imp:p=32 108 1 -4.57 -1 107 -108 imp:p=64 109 1 -4.57 -1 108 -109 imp:p=128 110 1 -4.57 -1 109 -110 imp:p=256 2 2 -0.9982 1 -3 #4 -20 imp:p=256 4 4 -7.94 2 -3 7 -71 imp:p=256 5 4 -7.94 5 -6 -7 #7 imp:p=256 6 2 -0.9982 -5 -7 imp:p=256 61 2 -0.9982 -20 71 (-10:-8) imp:p=256 71 3 -1.21E-03 -42 41 imp:p=256 7 5 -2.3 6 3 -4 #71 imp:p=256 8 5 -2.3 8 -9 10 imp:p=256 9 4 -7.94 -11 12 28 imp:p=256 10 0 -12 28 imp:p=0 11 4 -7.94 -13 14 28 imp:p=256 12 0 -14 28 imp:p=0 13 4 -7.94 -15 16 28 imp:p=256 14 0 -16 28 imp:p=0 15 4 -7.94 -17 18 28 imp:p=256 16 0 -18 28 imp:p=0 20 4 -7.94 21 -22 imp:p=256 imp:p=256 21 5 -2.3 22 -23 22 4 -7.94 24 -25 27 imp:p=256 23 5 -2.3 25 -26 27 imp:p=256 24 5 -2.3 -21 -28 9 #21 #22 11 13 15 17 imp:p=256 30 3 -1.21E-03 (-24:-21:-8:-10:-2) imp:p=256 11 13 15 17 20 #8 #24 #2 1 999 0 1 #2 #4 #5 #6 #7 #71 #8 #9 #10 #11 #12 #13 #14 #15 #16 #20 #21 #22 #23 #24 #30 #61 imp:p=0

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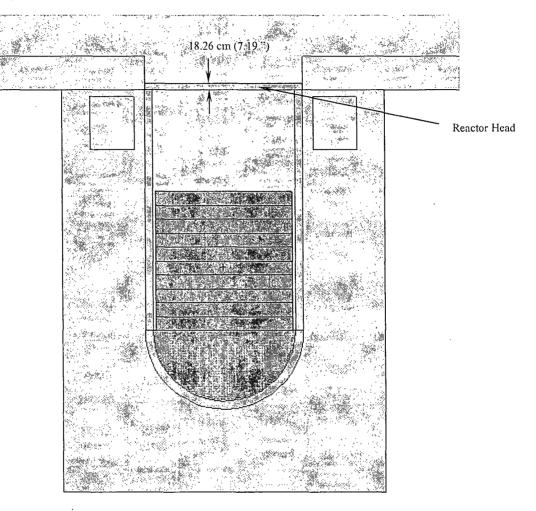
c cells

\$ Active Fuel Region \$ Active Fuel Region \$ Active Fuel Region \$ Active Fuel Region \$ Active Fuel Region \$ Active Fuel Region \$ Active Fuel Region \$ Active Fuel Region \$ Water Region \$ RPV Shell \$ Bottom RPV Shell \$ Water Above Fuel \$ Water Above Vessel Head \$ Void for Primary Loop \$ Concrete Surrounding RPV \$ Concrete above RPV \$ Steam Generator 1 \$ Inner Steam Generator 1 \$ Steam Generator 2 \$ Inner Steam Generator 2 \$ Steam Generator 3 \$ Inner Steam Generator 3 \$ Steam Generator 4 \$ Inner Steam Generator 4 \$ Containment Liner \$ Containment Wall \$ Containment Dome Liner \$ Containment Dome Concrete \$ 68 foot level \$ Air in Containment

\$ Problem Boundary

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Figure 7-7 X-Z VISED Plot of Reactor Vessel and Concrete Reactor Pit (With Head)



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Figure 7-8 MCNP Cell Cards (With Head)

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c cells	
$101 \ 1 \ -4 \ .57 \ -1 \ -101$	imp:p=1
$102 \ 1 \ -4.57 \ -1 \ 101 \ -102$	imp:p=2
103 1 -4.57 -1 102 -103	imp:p=3
$104 \ 1 \ -4.57 \ -1 \ 103 \ -104$	imp:p=4
105 1 -4.57 -1 104 -105	imp:p=8
106 1 -4.57 -1 105 -106	imp:p=16
107 1 -4.57 -1 106 -107	imp:p=32
108 1 -4.57 -1 107 -108	imp:p=64
109 1 -4.57 -1 108 -109	imp:p=128
110 1 -4.57 -1 109 -110	imp:p=256
2 2 -0.9982 1 -3 #4 -20 31	imp:p=256
4 4 -7.94 2 -3 7 -71	imp:p=256
5 4 -7.94 5 -6 -7 #7	imp:p=256
6 2 -0.9982 -5 -7	imp:p=256
62 6 -7.8212 -31	imp:p=256
61 2 -0.9982 -20 71 (-10:-8) 31	imp:p=256
71 3 -1.21E-03 -42 41	imp:p=256
7 5 -2.3 6 3 -4 #71	imp:p=256
8 5 -2.3 8 -9 10	imp:p=256
9 4 -7.94 -11 12 28	imp:p=256
10 0 -12 28	imp:p=0
11 4 -7.94 -13 14 28	imp:p=256
12 0 -14 28	imp:p=0
13 4 -7.94 -15 16 28	imp:p=256
14 0 -16 28	imp:p=0
15 4 -7.94 -17 18 28	imp:p=256
16 0 -18 28	<pre>imp:p=0</pre>
20 4 -7.94 21 -22	imp:p=256
21 5 -2.3 22 -23	imp:p=256
22 4 -7.94 24 -25 27	imp:p=256
23 5 -2.3 25 -26 27	imp:p=256
24 5 -2.3 -21 -28 9 #21 #22 11 13	
15 17	imp:p=256
30 3 -1.21E-03 (-24:-21:-8:-10:-2)	
11 13 15 17 20 31 #8 #24 #2 1	imp:p=256
999 0 1 #2 #4 #5 #6 #7 #71 #8 #9 #10	
#11 #12 #13 #14 #15 #16 #20 #21	
#22 #23 #24 #30 #61 31	imp:p=0

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	Active Fuel Region
	Active Fuel Region
\$	Active Fuel Region
	Active Fuel Region
	Active Fuel Region
	Active Fuel Region
\$	Active Fuel Region
\$	
\$	Active Fuel Region
\$	Active Fuel Region
\$	Water Region
\$	RPV Shell
Ş	Bottom RPV Shell
· \$	Water Above Fuel
\$	Reactor Vessel Head
\$	Water Above Vessel Head
\$	Void for Primary Loop
\$	Concrete Surrounding RPV
\$	Concrete above RPV
\$	Steam Generator 1
Ş	Inner Steam Generator 1
\$	Steam Generator 2
\$	Inner Steam Generator 2
Ş	Steam Generator 3
\$	Inner Steam Generator 3
\$ \$ \$	Steam Generator 4
\$	Inner Steam Generator 4
\$	Containment Liner
\$	Containment Wall
\$	Containment Dome Liner
\$	Containment Dome Concrete
\$	68 foot level
\$	Air inside Containment

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\$ External to Problem

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7.4 MCNP Source Definition

The core source term is assumed to be uniformly distributed throughout the volume, and has an energy spectra based on the core inventory [2]. Only the gamma source term is taken into account for this evaluation. Because the source term is generated immediately after shutdown, the fuel gamma source term will predominate. Therefore the N-gamma and hardware activation source terms can be neglected (Assumption 3). The source is defined on the MCNP *sdef* card using distributions to define the particle location and energy. The radius of the core is defined with the *rad* parameter, which automatically creates a uniform distribution based on a cylindrical geometry. The *ext* and *axs* parameters define the direction and distance of the cylinder axis. These parameters combined define the core where the particles can be born. The *erg* parameter defines the energy spectrum of source particles and is based on the results of the ORIGEN-S calculation discussed previously. This distribution is a histogram of energies represented by activities. These are automatically normalized by MCNP to create a probability distribution. The total activity is preserved in the tally multiplier. The MCNP source definition cards are shown below in Figure 7-9. The *sb* card is a source biasing card, which in this case biases the particle generation to the upper end of the core. This is a variance reduction technique to improve the statistical certainty in the results.

Figure 7-9 MCNP Source Definition Cards

sdef rad=d1 ext=d2 axs=0 0 1 erg=d8	←Source Definition Card -Radius = d1 -Extent = d2 -Axis = +Z -Energy = d8
sil 209.71	←Core Radius Distribution
si2 h 0 42.672 85.344 128.016 170.688 213.36 256.032 298.704	\leftarrow Core Axial Distribution
341.376 384.048 426.72	
sp2 0 1 1 1 1 1 1 1 1 1 1 1	\leftarrow Actual Uniform Distribution
sb2 0 0.001 0.001 0.01 0.01 0.01 0.1 0.1 0.	\leftarrow Biased to Top Distribution
c Fuel Gamma Spectra	
<pre>si8 h 1.000e-002 5.000e-002 1.000e-001 2.000e-001 3.000e-001 4.000 6.000e-001 8.000e-001 1.000e+000 1.330e+000 1.660e+000 2.000e 2.500e+000 3.000e+000 4.000e+000 5.000e+000 6.500e+000 8.000e 1.000e+001 1.100e+001</pre>	+000
sp8 0.00E+00 9.288E+19 2.926E+19 6.537E+19 4.277E+19 1.521E+19	3.578E+19 ←Source Emission on Energy Basis
4.352E+19 2.66E+19 1.289E+19 1.649E+19 5.572E+18 5.527E+18	1.984E+18
7 .812E+17 3.48E+16 3.947E+11 1.75E+08 37100000 2009000	

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7.5 MCNP Tally Specification

The tallies used in this evaluation are point detectors placed at approximate locations of radiation monitors RE-8055 and RE-8099. Point detectors are chosen because they use quasi-deterministic dose calculations that will provide better results than surface or cell based tallies that require the particles to enter those regions. The inputs to this card are the coordinates of the dose points followed by an exclusion zone (reduce variance), as well as a multiplier card, which represents the total core activity in photons/sec. The tally cards are shown in Figure 7-10.

Figure	7-10	MCNP	Tally	Cards
--------	------	------	-------	-------

f5c RE-8055 and RE-8099	←Tally Comment Card
f5:p -1200 -400 1909.24 20	\leftarrow Tally 5 (point detector)
1200 400 1909.24 20	x y z exclusion
	-1200 -400 1909.24 20
	1200 400 1909.24 20
fm5 3.947E+20	← Tally Multiplier
	(Total Activity)

In addition, the flux is multiplied by ANSI/ANS flux-dose conversion factors [11]. This is specified in MCNP using the *de/df* cards. These are shown in Figure 7-11.

Figure 7-11 ANSI/ANS-6.1.1-1977 Gamma Flux to Dose Conversion Factors

C _____ c ANSI/ANS-6.1.1-1977 c Gamma Flux to Dose Conversion Factors c (mrem/hr)/(photons/cm2-s) с ----de0 .01 .03 .05 .07 .10 .15 .20 .25 .30 .35 .40 ←Energy Bins for Flux .45 .50 .55 .60 .65 .70 .80 1. 1.4 1.8 2.2 2.6 2.8 3.25 3.75 4.25 4.75 5. 5.25 5.75 6.25 to Dose Conversion 6.75 7.5 9. 11. 3.79E-04 df0 3.96E-03 5.82E-04 2.90E-04 2.58E-04 2.83E-04 ←Energy Dependent 5.01E - 046.31E-04 7.59E-04 8.78E-04 9.85E-04 1.08E-03 Flux Multipliers 1.68E-03 1.17E-03 1.27E-03 1.36E-03 1.44E-03 1.52E-03 1.98E-03 2.51E-03 2.99E-03 3.42E-03 3.82E-03 4.01E-03 4.41E-03 4.83E-03 5.23E-03 5.60E-03 5.80E-03 6.01E-03 6.74E-03 7.11E-03 7.66E-03 8.77E-03 1.03E-02 6.37E-03

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7.6 MCNP Material Cards

The MCNP material cards are provided in Figure 7-12. These are based on the compositions described in Table 5-4.

Figure 7-12 MCNP Material Cards⁵

m1	92235	-0.0245		· .
	92238	-0.5891		
	8016	-0.2521		
	40000	-0.1118		
	50000	-0.0017		:
	24000	-0.0001		
	26000	-0.0002		
	1001	-0.0211		
	6012	-0.0001		
	1001 2 8016 1			Water
m3	6012 -0.00		\$	Air
	7014 -0.76			
	8016 -0.23			
m4	6000 -0.00		\$	SS 304
	14000 -0.0			
	15031 -0.0			
	24000 -0.1			
	25055 -0.0			
	26000 -0.6			
-	28000 -0.0			
m5	26000 -0.0		Ş	Reg-Concrete
	1001 -0.01			
	13027 -0.0			
	.20000 -0.0			
	8016 -0.53	-		
	14000 -0.3			
~	11023 -0.0			
mб	6012 -0.01			\$ Carbon Steel
	26056 -0.9	9		

⁵ Material 1 composition will change based on the water level relative to the core. This only applies to water heights below 14 feet.

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7.7 Results

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File Naming Scheme:

The MCNP input files are named with the following convention:

P-height-condition where:

P = Project (STP)

Height = water height from bottom of core (ft)

Condition = h - with headn - no head

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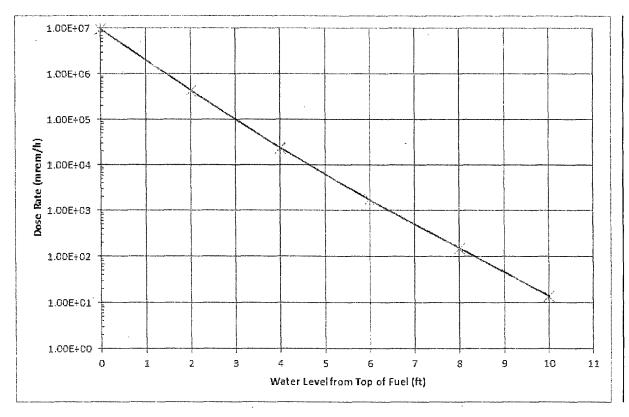
7.7.1 Results without Head

The dose rate as a function of water level is provided in Table 7-3 and plotted in Figure 7-13, below. Because the MCNP model geometry is symmetric in the x and y planes, the two point detector locations should provide the same dose rate. To increase the statistical certainty in the final result, the two individual dose rate responses and uncertainties are combined using inverse variance averaging. All of the water levels described in the following sections refer to the level at the top of the fuel (i.e. 0 foot water level is at the top of the fuel assemblies and \sim 13 feet is flange level).

Water Level (ft)	Dose Rate 1	fsd	Dose Rate 2	fsd	Dose Rate Avg	Avg fsd
0	9.27E+06	0.0081	9.34E+06	0.0109	9.30E+06	0.0065
2	4.26E+05	0.0078	4.31E+05	0.0093	4.28E+05	0.0060
4	2.31E+04	0.0236	2.32E+04	0.0247	2.32E+04	0.0171
6	1.73E+03	3.10E-02	1.69E+03	2.44E-02	1.70E+03	0.0192
8	1.51E+02	0.0302	1.51E+02	0.0287	1.51E+02	0.0208
10	1.40E+01	0.036	1.36E+01	0.0323	1.38E+01	0.0240

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Figure 7-13 Dose Rate versus Water Height Plot for no Head Configuration



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7.7.2 Results with Head

The dose rate results for the cases with the head in place are the same, except the minimum detectable dose rate is lower due to the lower ambient dose rate in the containment. The dose rates are listed in Table ,7-4 and plotted in Figure 7-14.

Table.7-4 Dose Rate Response as a Function of Water Level for Head on Configuration (mrem/h)

Water Level (ft)	Dose Rate 1	fsd	Dose Rate 2	fsd	Dose Rate Avg	Avg fsd
0	2.16E+04	0.094	2.56E+04	0.185	2.24E+04	0.0838
2	1.87E+03	0.083	1.83E+03	0.074	1.85E+03	0.0554
4	1.11E+02	0.061	1.08E+02	0.069	1.10E+02	0.0455
6	8.89E+00	0.085	· 7.48E+00	0.048	7.82E+00	0.0418
8	8.95E-01	0.125	8.12E-01	0.093	8.42E-01	0.0742

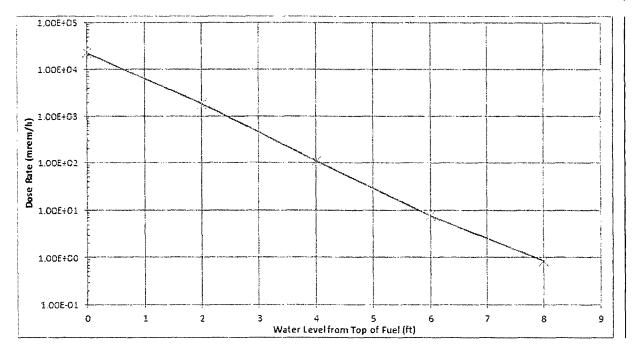
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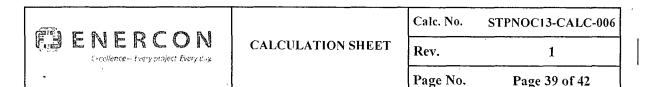
Figure 7-14 Dose Rate versus Water Height Plot for with Head Configuration



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Appendix A – ENERCON Reference EMAILS



Dresv	Blackwel	1

From: Sent:	Paul Sudnak Monday, December 09, 2013 9:55 AM
То:	Chad Cramer, Joanne Morris
Cci	Marvin Morris; Jeff Gromatzky, Michael Falkner, Jay Maisler, Caleb Trainor
Subject:	RE: STP Refueling Cavity Level Calc

Sure, let me find the elevation drawing for the cavity. The water level during refueling is the same water level as the spentituel pool during fuel transfer. The height of the active fuel is 28'-2 inches. The vessel flange level is 39'-3", and mid-loop is 32'-3".

RCB radiation monitors (RE-8055 and RE-8099) read from < 1 mR/hr to 2.5 mR/hr during refueling. If the upper-internal package or head are being removed, levels can increase to over ~ 100mR/hr for the upper internals. Levels on the refueling deck (65'0'') at mid-loop will only increase to ~ 10 mR/hr with the water level that low. When the head is being de-tensioned by worker on the head level platform (~ 39' EL), dose rates at that location can read ~ 50 to over 100 mR/hr. The general area dose rates from core radiation is usually less than 100 mR/hr, unless there are lots of fuel leaks or high RCS corrosion and activation products. Dose rates at the monitors at figning level are usually less than 5 mR/hr.

Paul J. Sudnak Enercon Services, Inc. 12906 Tampa Daks Boulevard Suite 131 Temple Tarrace, Florida 33637

Office: (813) 962-1800 X603 Fax: (813) 962-1881 Cell: (813) 389-0960



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From: Chad Cramer Sent: Friday, December 06, 2013 2:08 PM To: Joanne Morris Cc: Marvin Morris; Jeff Gromatzky; Paul Sudnak; Michael Falkner Subject: STP Refueling Cavity Level Calc

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Joanne,

Michael Falkner has completed the STP SFP cale and sent it to me for review. I spoke with he and Jeff Gromatzky and they indicated that he should have ability over the next week or so to do the refueling cavity level cale.

Paul,



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Drew Blackwell

From: Sent: To: Cc Subject: Sudnak, Paul <pjsudnak@STPEGS.COM> Monday, February 03, 2014 3:44 PM Caleb Trainor, Drew Blackwell Domal, Michael; Jay Maisler RE: Fuel Assembly Dimension

Thanks Caleb,

Assume the only attenuation is from the materials between the detectors and the core. Disregard reflection, I don't think the SGs are between the core and detector, nor is the containment wall. Concrete should be high density. Atmosphere should be saturated steam at greater than 9.5 psi (containment spray initiation) and less than 56 psi (containment design pressure) mixed with air at the original containment volume at STP. The detectors are ion chambers. Do not include neutrons. The reactor vessel head is around ~ 8° thick and carbon steel. I will get you the actual drawing of the RPV head thickness, but I think it is from the UFSAR,

.....

Again, a peer check from Mike or Jay? ... Paul

From: Caleb Trainor [mailto:ctrainor@enercon.com] Sent: Monday, February 03, 2014 2:21 PM To: Sudhak, Paul; Drew Blackwell Cc: Domal, Michael; Jay Maisler Subject: RE: Foel Assembly Dimension

Drew is working on CS1/CG1 where the concern is direct shine from the core due to lowered water levels and no fuel damage assumed. I think you may be thinking of the fission product barrier calcs that I'm working on.

-Caleb

From: Sudnak, Paul <<u>Disudnak@STPEGS.COM</u>> Sent: Monday, February 3, 2014 3:05 PM To: Caleb Trainor, Drew Blackwell Cc: Domai, Michael, Jay Maisler Subject: RE: Fuel Assembly Dimension

I think Caleb is right here. Once the concentration is known, the detectors are going to respond to the gases primarily above the 68' Elevation, all the rest will be significantly attenuated by the concrete floors, inner and outer Bio-shield wall, steam generators, and the pressurizer. To model all of those structures would require an extensive geometry and a considerable amount of data. Our intent here is to identify the concentration of gases above the 68' El and determine the monitor response. Disregard the Steam Generators, the inner and outer bio-shield walls, and the Pressurizer. With an assumed homogenous mix based on 20% fuel damage, the dose rates should be significant. Factoring in additional structures and elevations will not significantly change the outcome: A General Emergency will be declared.

Earlier today, I sent the location of the containment high range monitors (73' Elevation). Assume they see the volume of the reactor containment building above the 68' El.

Mike/Jay, can you give me a peer check here? Paul

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Appendix B – Electronic File Listing

Volume in drive F is My Passport Volume Serial Number is 1AEA-6007 Directory of F:\STPNOC013-CALC-006\Rev 1

i.

03/14/2014	04:12 PM	<dir></dir>	
03/14/2014	04:12 PM	<dir></dir>	••
03/21/2014	09:33 PM	0	dir.dat
02/06/2014	02:03 PM	100,953	EMAIL from Paul Sundak, Dec. 9 2013.pdf
02/07/2014	10:26 AM	8,795	Inverse Variance Weighting.xlsx
03/14/2014	08:44 AM	332,025	liner plate info.pdf
03/21/2014	09:32 PM	<dir></dir>	mcnp
03/14/2014	04:12 PM	<dir></dir>	origen
02/07/2014	12:14 PM	111,247	RE Fuel Assembly Dimension.pdf
03/14/2014	09:10 AM	462,166	RPV with core.pdf
03/14/2014	08:48 AM	537,808	RPV.pdf
03/14/2014	04:06 PM	43,842	STP.xlsx
03/14/2014	04:02 PM	1,036,800	STPNOC013-CALC-006 R1.doc
	9 File(s) 2,633,630	6 bytes

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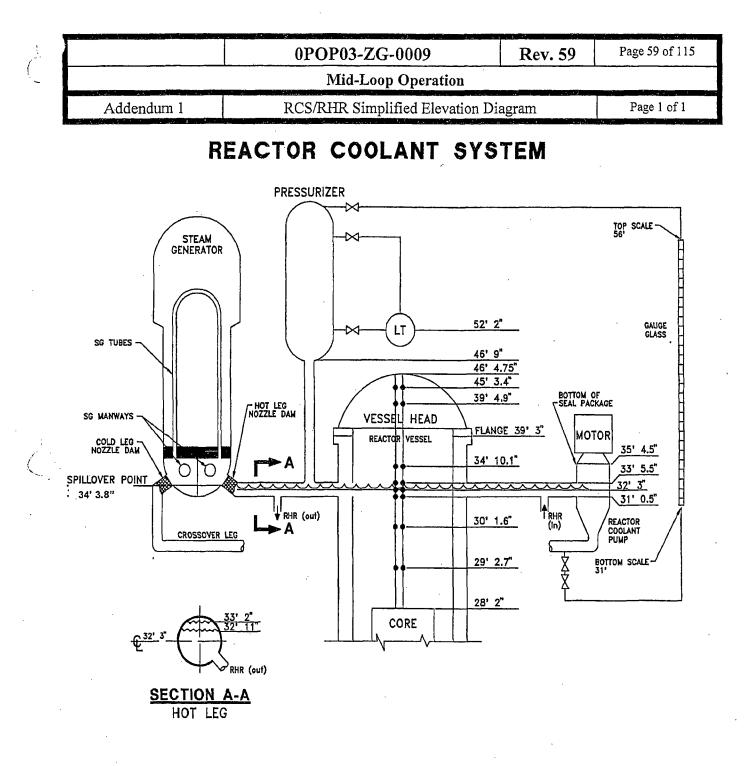
03/21/2014 03/21/2014 03/21/2014 02/06/2014 03/21/2014 03/21/2014 03/21/2014	09:32 PM 09:32 PM 09:32 PM 11:31 AM 09:34 AM 09:45 AM 09:45 AM	<dir> <dir> <dir> <dir></dir></dir></dir></dir>	18,720 4,053 9,744	head no head STP.bat STP.sx STP_default.sx sx.log
03/21/2014 03/21/2014	09:45 AM		2,007	_sx.var
	5 File(s)	34,66	l bytes

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03/21/2014	09:32	PM	<dir></dir>		
03/21/2014	09:32	PM	<dir></dir>		
03/12/2014	12:51	PM		8,990	STP14h5
03/12/2014	08:50	PM		1,104	STP14h5m
03/12/2014	08:50	РM		924,317	STP14h5o
03/21/2014	09:45	AM		8,587	STP14h8
03/21/2014	04:41	PM		1,260	STP14h8m
03/21/2014	09:17	PM		1,312	STP14h8m2
03/21/2014	04:41	PM		545,780	STP14h8o
03/21/2014	09:17	PM		557,996	STP14h8o2
03/14/2014	08:27	AM		8,990	STP16h7
03/14/2014	03:42	PM		1,260	STP16h7m
03/14/2014	03:42	РM		942,029	STP16h7o
03/21/2014	09:45	AM		8,587	STP16h8
03/21/2014	04:43	PM		1,312	STP16h8m
03/21/2014	09:10	PM		1,364	STP16h8m2
03/21/2014	04:43	PM		557,572	STP16h8o
03/21/2014	09:10	PM		543,468	STP16h8o2
03/13/2014	04:35	ΡM		8,990	STP18h6
03/13/2014	08:40	PM		1,156	STP18h6m
03/13/2014	08:40	PM		552,616	STP18h6o
03/21/2014	09:45	AM		8,587	STP18h8
03/21/2014	04:43	PM		1,260	STP18h8m
03/21/2014	09:17	РМ		1,312	STP18h8m2
03/21/2014	04:43	РМ		551,487	STP18h8o
03/21/2014	09:17	РM		565,735	STP18h8o2
03/12/2014	01:17	РM		8,989	STP20h5
03/12/2014	08:51	PM		1,104	STP20h5m
03/12/2014	08:51	РМ		966,684	STP20h5o

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03/21/2014	09:45	ΔM		8,586 STP20h8				1
03/21/2014	01:52			1,260 STP20h8m		· .		1
03/21/2014	07:40	РМ	:	1,364 STP20h8m2				
03/21/2014	01:52			0,735 STP20h8o				
03/21/2014	07:40			8,209 STP20h8o2				
03/12/2014 03/12/2014	01:17 08:51			8,990 STP22h5 1,104 STP22h5m				ļ
03/12/2014	08:51			6,997 STP22h50				
03/21/2014	09:45			8,587 STP22h8				1
03/21/2014	01:52			1,260 STP22h8m				1
03/21/2014	07:07			1,364 STP22h8m2				1
03/21/2014 03/21/2014	01:52 07:07			7,911 STP22h8o 5,528 STP22h8o2				
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03/14/2014	09:14			1,364 STP14n7m				
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03/21/2014	09:45			8,468 STP14n8				
03/21/2014 03/21/2014	03:46 03:46			1,364 STP14n8m 7,366 STP14n8o				
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03/14/2014	11:49			1,364 STP16n7m				
03/14/2014	11:49	AM	70	2,157 STP16n7o				ł
03/21/2014	09:45			8,468 STP16n8				
03/21/2014 03/21/2014	03:46 03:46			1,416 STP16n8m				
03/13/2014	03:40			7,587 STP16n80 8,863 STP18n6				
03/14/2014	01:10			1,520 STP18n6m				
03/14/2014	01:10	AM	54	1,013 STP18n60				
03/21/2014	09:45			8,468 STP18n8				
03/21/2014 03/21/2014	02:02 02:02			1,572 STP18n8m 4,616 STP18n8o				1
03/13/2014	02:02			8,268 STP20n7				
03/14/2014	12:32			1,364 STP20n7m				Í
03/14/2014	12:32	PM	90	2,858 STP20n7o				
03/21/2014	09:45			8,467 STP20n8				
03/21/2014	03:46			1,312 STP20n8m				
03/21/2014 03/13/2014	03:46 04:56			0,778 STP20n80 8,273 STP22n7				
03/14/2014	08:58			1,364 STP22n7m				
03/14/2014	08:58			8,128 STP22n7o				
03/21/2014	09:45	AM		8,468 STP22n8				
03/21/2014	03:46			1,312 STP22n8m			-	
03/21/2014 03/13/2014	03:46 04:56			54,349 STP22n80 8,272 STP24n7				1
03/14/2014	12:45			1,364 STP24n7m				ļ
03/14/2014	12:45	PM	87	0,869 STP24n7o				
03/21/2014	09:45			8,468 STP24n8				
03/21/2014	03:46			1,312 STP24n8m				1
03/21/2014	03:46 36	PM File(s)		60,196 STP24n8o 599,621 bytes				
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	Mid-Loop Operation		
Addendum 2	RVWL Sensor Elevation	ons	Page 1 of 1

<u>NOTE</u>

- Top of Core is elevation 28 ft 2 inches.
- SG spillover is elevation 34 ft 3.8 inches.

SENSOR UNCOVERED	UPPER HEAD INDICATED LEVEL (%)	PLENUM INDICATED LEVEL (%)	SENSOR	LEVEL DESCRIPTION
All Covered	100	100	46' 4.75"	Upper Head Full
· 1	64	100	45' 3.4"	Upper Head Partially Drained
2	0	100	39' 4.9"	Plenum Full
3	0	85	34' 10.1" -	Plenum <u>NOT</u> Full (Enter Reduced Inventory)
4	0	66	33' 5.5"	Top of Hot Leg Nozzle
5	0	48	32' 3."	Hot Leg Centerline
6	0	33	31' 0.5"	Bottom of Hot Leg Nozzle
7	0	20	30' 1.6"	Midway between Hot Leg Nozzle and Upper Core Plate
8	0	0	29' 2.7"	Upper Core Plate

TABLE 12.3.4-1

AREA RADIATION MONITORS

Reactor Containment Building

Tag Number and Location ⁽¹⁾	Range (mR/hr) ⁽³⁾	High Alarm Setpoint (mr/hr) ⁽²⁾
N1RA-RE-8052 Incore Instrumentation Room (-1ft-6 in.)	10 ⁻¹ -10 ⁴	1,000
N1RA-RE-8053 Support across from elevator (-11 ft-3 in.)	$10^{-1} - 10^4$	100
N1RA-RE-8054 West Stair Landing (19 ft-0 in.)	10 ⁻¹ -10 ⁴	100
N1RA-RE-8055 North SG wall across from the head laydown area (68 ft-0 in.)	10 ⁻¹ -10 ⁴	100
N1RA-RE-8056 Support across from elevator (52 ft-0 in.)	$10^{-1} - 10^{4}$	100
N1RA-RE-8099 South SG wall across from the in-containment fuel pool (68 ft-0 in.)	10 ⁻¹ -10 ⁴	100

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

3. Range is based on microprocessor conversion factor and a detector signal which has a high degree of confidence. Conversion factor will vary dependent on the detector calibration. Exact ranges are found in plant instrument scaling manuals.

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Fuel Handling Building

Tag Number and Location ⁽¹⁾	Range (mR/hr)	High Alarm Setpoint (mr/hr) ⁽²⁾
N1RA-RE-8081 ~11 ft S of cols. 30.2 and S_5 (68 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5
N1RA-RE-8084 ~24 ft S of cols. 28 and T ₅ (-21 ft-0 in.)	$10^{-1}-10^4$	2.5
N1RA-RE-8085 ~24 ft S of col. 28 and ~6 ft E of col. S_5 (-21 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5
N1RA-RE-8086 ~24 ft S of col. 28 and ~11 ft E of col. R ₁ (-21 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5
N1RA-RE-8087 col. 30.2 and 12 ft W of col. R ₁ (4 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5
N1RA-RE-8088 3 ft S of col. 30.9 and col. R_1 (30 ft-0 in.)	$10^{-1}-10^4$	2.5
N1RA-RE-8089 col. 28 and col. N (68 ft-0 in.)	$10^{-1} - 10^{4}$	2.5
N1RA-RE-8090 18 ft N of col. 30.2 and col. T_5 (68 ft-0 in.)	$10^{-1} - 10^4$	2.5
N1RA-RE-8091 col. 34 and col. N (68 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

12.3-25

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Fuel Handling Building (Continued)

Tag Number and Location ⁽¹⁾	Range (mR/hr)	High Alarm Setpoint (mr/hr) ⁽²⁾
N1RA-RE-8097	10 ⁻² -10 ⁷	1,000
33 ft S of cols. 28 and 10 ft W of col. N		
(68 ft-0 in.)		

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Mechanical Electrical Auxiliaries Building

Tag Number and Location ⁽¹⁾	Range (mR/hr) ⁽³⁾	High Alarm Setpoint (mr/hr) ⁽²⁾
N1RA-RE-8057 col. 22 and ~10 ft E of col. J (10 ft-0 in.)	10 ⁻² -10 ³	0.5
N1RA-RE-8058 col. 26 and col. J (10 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5
NIRA-RE-8059 col. 27 and col G (10 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5
N1RA-RE-8060 ~10 ft S of col. 30 and col. E (10 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5
N1RA-RE-8061 ~10 ft S of col. 24 and ~11 ft W of col. E (10 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5
N1RA-RE-8062 ~6 ft S of col. 31 and col. C (10 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5
N1RA-RE-8063 ~9 ft S of col. 28 and col. B (10 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5
N1RA-RE-8064 ~12 ft S of col. 24 and col. F (29 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5
N1RA-RE-8065 ~5 ft N of col. 32 and col. C (29 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

3. Range is based on microprocessor conversion factor and a detector signal which has a high degree of confidence. Conversion factor will vary dependent on the detector calibration. Exact ranges are found in plant instrument scaling manuals

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Mechanical Electrical Auxiliaries Building (Continued)

Tag Number and Location (1)	Range (mR/hr) ⁽³⁾	High Alarm Setpoint (mr/hr) ⁽²⁾
N1RA-RE-8066 ~4 ft N of col. 22 and 14 ft E of col. C (35 ft-0 in.)	10 ⁻² -10 ³	0.5
N1RA-RE-8067 col. 22 and 10 ft E of col. J (35 ft-0 in.)	10 ⁻² -10 ³	0.5
N1RA-RE-8068 ~10 ft N of col. 25 and col. H (41 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5
N2RA-RE-8068 ~10 ft S of col. 24 and col. G (41 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5
N1RA-RE-8069 ~12 ft S of col. 24 and ~14 ft E of col. C (41 ft-0 in.)	10 ⁻² -10 ³	0.5
N1RA-RE-8070 col. 29 and col. C (41 ft-0 in.)	10 ⁻² -10 ³	2.5
N1RA-RE-8071 ~18 ft S of col. 28 and 3 ft W of col. B (41 ft-0 in.)	10 ⁻² -10 ³	2.5
N1RA-RE-8072 ~11 ft N of col. 29 and 5 ft W of col. D (41 ft-0 in.)	10 ⁻¹ -10 ⁴	-100
N1RA-RE-8073 col. 29 and col. E (41 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5
N1RA-RE-8074 ~5 ft S of col. 31 and ~7 ft W of col. C (41 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

12.3-28

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Mechanical Electrical Auxiliaries Building (Continued)

Tag Number and Location (1)	Range (mR/hr)	High Alarm Setpoint (mr/hr) ⁽²⁾
N1RA-RE-8075 col. 28 and ~3 ft W of col. G (41 ft-0 in.)	10 ⁻¹ -10 ⁴	15.0
N1RA-RE-8076 col. 22 and ~10 ft E of col. J (60 ft-0 in.)	10 ⁻² -10 ³	0.5
N1RA-RE-8077 col. 27 and col. J (60 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5
N1RA-RE-8078 col. 27 and col. F (60 ft-0 in.)	10 ⁻¹ -10 ⁴	15.0
N1RA-RE-8079 col. 25 and ~2 ft W of col. F (60 ft-0 in.)	10 ⁻¹ -10 ⁴	15.0
N1RA-RE-8080 col. 26 and col. H (41 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5
N1RA-RE-8082 col. 28 and ~8 ft E of col. H (69 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5
N1RA-RE-8083 ~10 ft S of col. 29 and 15 ft W of col. E (41 ft-0 in.)	10 ⁻¹ -10 ⁴	15.0
N1RA-RE-8098 ~6 ft N of col. 25 and col. H (60 ft-0 in.)	10 ² -10 ⁷	1000

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

12:3-29

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Miscellaneous Buildings

Tag Number and Location ⁽¹⁾	Range (mR/hr)	High Alarm Setpoint (mr/hr) ⁽²⁾
N1RA-RE-8092 col. 9 and col. P TGB (29 ft-0 in.)	$10^{-2} - 10^{3}$	0.5
N1RA-RE-8093 col. 7 and col. M TGB (29 ft-0 in.)	10 ⁻² -10 ³	0.5
N1RA-RE-8094 ~3 ft N of col. 23 and ~14 ft W of col. B TSC-MEAB	10 ⁻² -10 ⁷	1000
(72 ft-0 in.)		CN-2963

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

12.3-30

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Post-Accident Monitors

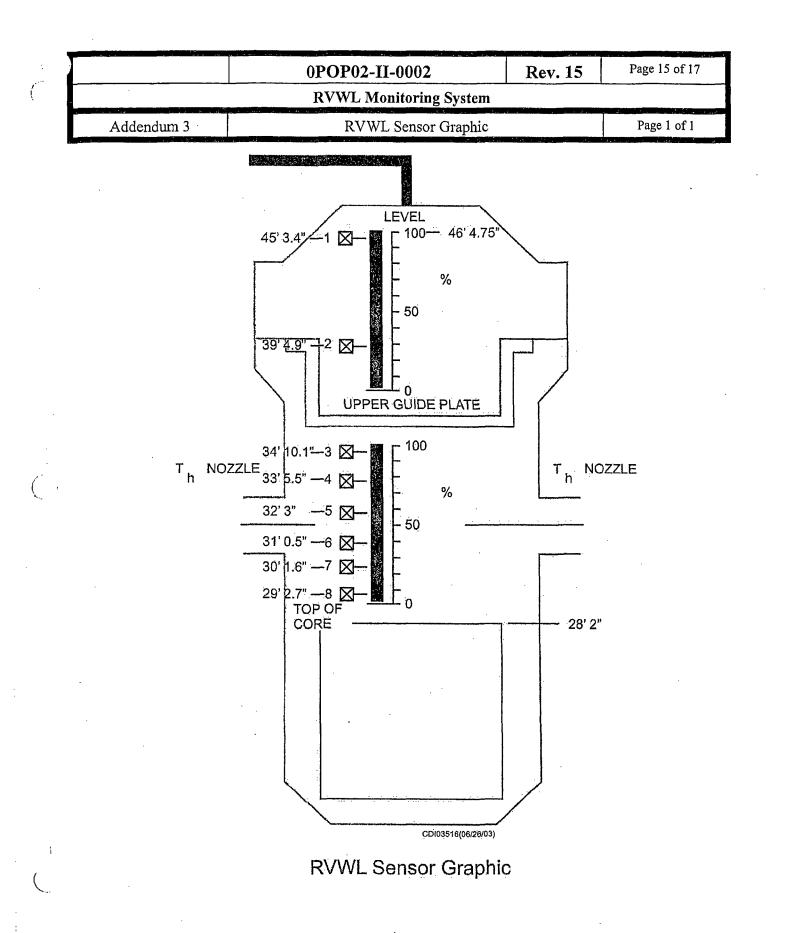
Tag Number and Location ⁽¹⁾	Range (R/hr)	High Alarm Setpoint (R/hr) ⁽²⁾
A1RA-RE-8050 RCB (68 ft-0 in.)	10 ⁰ -10 ⁸	2000
C1RA-RE-8051 RCB (68 ft-0 in.)	10^{0} -10 ⁸	2000

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

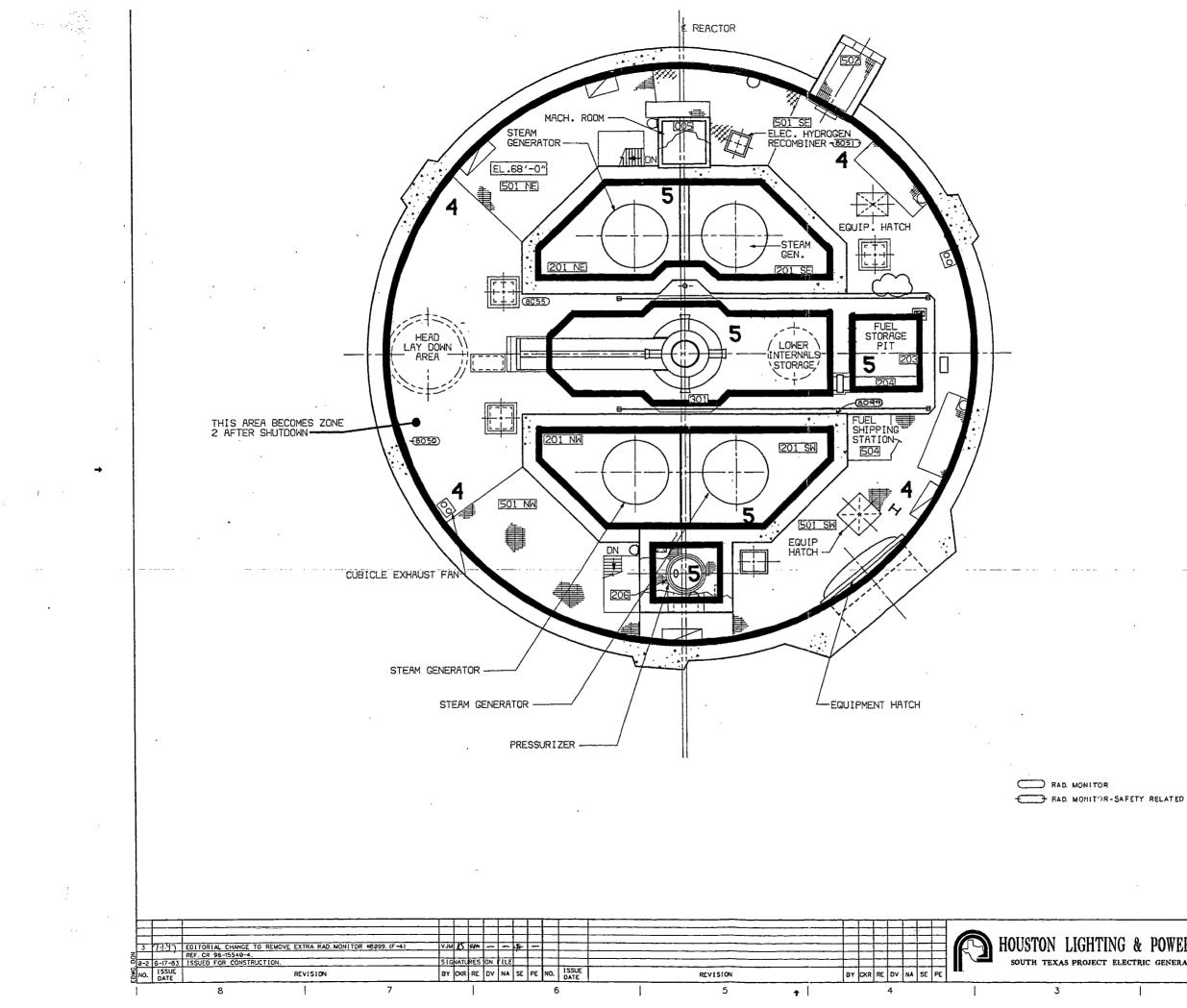
2. The alarm setpoints listed are typical and may be varied as necessary.

12.3-31

			0POP02-II-0	002	Rev. 15	Page 13 of 17
			RVWL Monit	oring Systen	1	
Ad	dendum 1		RVWL Ser	Page 1 of 1		
			NO	 TE	<u> </u>	
• ,	Гор of Co	re is elevation 28 f				
		ver is elevation 34				
					D the sensor is "we n Data Sheet 1 or 2	
					CD the sensor is "dr able on Data Sheet	
J	RVWL PI		ED LEVEL (%)	would be 85	0R No. 4 WET is ci %, SENSOR Locat	
WET	OR No. 7/DRY e one)	UPPER HEAD INDICATED LEVEL (%)	PLENUM INDICATED LEVEL (%)	SENSOR	LEVEL DE	SCRIPTION
All	Wet	100	100	46' 4.75"	Upper I	lead Full
	DR No. 1 VDRY	64	100	45' 3.4"	Upper Head P	artially Drained
	OR No. 2 /DRY	0	100	39' 4.9"	Plent	m Full
-	OR No. 3 /DRY	0	85	34' 10.1"	Plenum	NOT Full
	DR No. 4 7/DRY	0	66	33' 5.5"	Top of Ho	t Leg Nozzle
SENSC	OR No. 5 7/DRY	• 0	48	32' 3"	Hot Leg	Centerline
WEI		0	33	31' 0.5"	Bottom of H	ot Leg Nozzle
SENSO	DR No. 6 MDRY					
SENSO WET		0	20	30' 1.6"		Hot Leg Nozzle and Core Plate



0POP04-RC-0003	Excessive RCS Leaks	age Rev.	18 Page 53 of 12
Addendum 9	Basis	en na de la companya br>Nota ya companya de la	Basis Page 5 of 7
	DESCRIPTION FOR 0POP04-R		
•	For Any Of The Following Indicati	ions Of RCS Leakag	ge:
	r RT8011 Particulate – Rising		
Reactor Cool	ant Drain Tank Level – Rising	· ·	
	elief Tank Level – Rising		
RCB Normal	Sump Level – Rising		
PURPOSE: To determin	ne if leakage is from RCS and not (CVCS.	
	T8011, RCDT, PRT or RCB Norm		
that the leakage is from	RCS and not CVCS which is norm	ally tied to the RCS	•
ACTIONS: Monitor tre	nds from RT8011, RCDT, PRT or	RCB Normal Sump.	
		_	
INSTRUMENTATION	nds from RT8011, RCDT, PRT or : Level indications located on CP0 trol room. Radiation Monitor Comp	04 and various plant	
INSTRUMENTATION	: Level indications located on CP0 trol room. Radiation Monitor Comp	04 and various plant	
INSTRUMENTATION monitors located in com	: Level indications located on CP0 trol room. Radiation Monitor Comp	04 and various plant	
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	2		140	PERMISSIBLE	1 1	
	3	< 15	YES	CONTROLLED ACCESSIBLE ON A PERIODIC BASIS]	
	4	≼ 100	YES	CONTROLLED LIMITED ACCESS		
	5	> 100	YES	NORMALLY INACCESSIBLE		
						¥
			NDIATION ZON CONTAINMENT			
TING & POWER COMPANY	r		N AT EL. 68'		1	
ECT ELECTRIC GENERATING STATION	SCALE	T	DWG. NO.		REV.	
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2 \$	MSECM NO. CI 29A81105	PRICRITY C	@ w@%;	NO. A1 043 0928 A1 043	. CIT . Ø31	

PLANT	RADIATION	SHIELDI	NG ZONES
ZONE NUMBER	MAX. DOSE RATE	POSTING	ANT ICIPATED ACCESS
1	< 0.5	ND	UNRESTRICTED
2	< 2.5	NO	CONTROLLED.40HR/H
3	< 15	YES	CONTROLLED ACCESSIBLE ON A PERIODIC BASIS
4	≼ 100	YES	CONTROLLED
5	> 100	YES	NORMALLY INACCESSIBLE

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E) E	NERCON Excellence – Every project. Every day	CALCULATION COVER SHEET	REV. 1		
			PAGE NO. 1 of 42		
Title:	Dose Rate Evaluation of Reactor Vessel Water Levels Client: STP during Refueling for EAL Thresholds Project: STPNOC Cover Sheet Items		Client: STP		<u>-</u>
			Project: STPNO	C013	
Item			Yes	No	
1	Does this calculation contain any open assumptions that require confirmation? (If YES , Identify the assumptions)				
2	design verified calcu	n serve as an "Alternate Calculation"? (If lation.) Iculation No	•		
3	Does this calculation Supersede an existing Calculation? (If YES, identify the superseded calculation.) Superseded Calculation No				
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		ET REV. 1	
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PAGE NO.	PAGE REVISION STA	TUS PAGE NO. <u>REVISION</u>	
All 5-16,18,19,21- 26,28,30,31,34-42	0 1		
	APPENDIX REVISION S	TATUS	
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Excellence — Evety project. Every day:	DESIGN VERIFICATION PLAN AND SUMMARY SHEET	REV.	1
		PAGE NO.	3 of 42
Calculation Design Verification I	Plan;	——————————————————————————————————————	
The calculation will be reviewed for criteria.	r correctness of Inputs, design criteria, analy	zed methods, and a	cceptan
The stated objectives and conclusi	ions will be confirmed to be reasonable and v	valid.	
Assumptions will be reviewed and principles and practices.	confirmed to be appropriate and verified to b	e valld based on so	und eng
-		·	
(Print Nan	ne and Sign for Approval – mark "N/A" If I	not required)	
Approver: Marvin Morris	Marvin Morris Contraction	Date:	
Calculation Design Verification	Cer: Wel\$1511.000		
	ted as Safety Related as noted in the cover	sheet	
	to be correct and performed using appropriat		umptior
The conclusions developed in the	calculation are reasonable, valid, and consis	tent with the purpos	e and s
The assumptions are appropriate	and valid.		
Based On The Above Summary	, The Calculation Is Determined To Be Acc	ceptable.	
	(Print Name and Sign)		
	AMA D	1	
Design Verifier: Curt Lindner	INVIAC	Date: 3/2	1114
Others:	10	Date:	¥

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CHECKLIST PAGE NO. 4 of 42 Item CHECKLIST ITEMS Yes N 1 Design Inputs - Were the design inputs correctly selected, referenced (latest revision), consistent with the design basis, and incorporated in the calculation? X 2 Assumptions - Were the assumptions reasonable and adequately described, justified and/or verified, and documented? X 3 Quality Assurance - Were the appropriate QA classification and requirements assigned to the calculation? X 4 codes, Standards, and Regulatory Requirements - Were the applicable codes, standards, and regulatory requirements - Were the applicable codes, standards, and regulatory requirements - Were the applicable codes, standards, and regulatory requirements, including issue and addenda, properly identified and their requirements satisfied? X 5 Construction and Operating Experience - Have applicable construction and operating experience been considered? X 7 Methods - Was the calculation methodology appropriate and properly applied to satisfy the calculation objective? X 8 it correspond directly with the objectives, and are the results reasonable compared to the inputs? X 9 Radiation Exposure - Has the calculation properly considered radiation exposure to the public and plant personnel? X	
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Acceptance Criteria - Are the acceptance criteria incorporated in the	<u>x</u>
10 calculation sufficient to allow verification that the design requirements have X been satisfactorily accomplished?	
11Computer Software - Is a computer program or software used, and if so, are the requirements of CSP 3.02 met?X	
COMMENTS: In accordance with CSP 3.02, MCNP5 and SCALE6.0 have been verified for use on computers.	ENERCO
(Privit Name and Sign)	;

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Purpose and Scope

1.

The purpose of this calculation is to evaluate dose rates as a function of water height in the reactor vessel during refueling operations in order to set Emergency Action Level (EAL) thresholds for core uncovery. The dose rates are calculated at the locations of the containment monitors RE-8055 and RE-8099 so that dose rate measurements by these devices can be used to estimate water level in the core, upon failure of other water level detection systems. This evaluation will calculate the dose rate at full core uncovery, as well as maximum water levels with a detectable dose rate response. Since the scope of this calculation concerns uncovering the reactor core, the effects of future fuel element storage in the nearby Fuel Storage Pit are not analyzed, since it's effects are negligible in comparison. The containment building, components within the building, and the reactor vessel and contents are modeled simplistically because only order of magnitude results are needed. As such, the dose rate results should be considered as reasonably representative of the magnitude of the actual dose rate only.

2. Summary of Results and Conclusion

The dose rate results for the configuration without the reactor vessel head and with the reactor vessel head are provided in Section 7.7.1 and Section 7.7.2, respectively. The dose rate with the core uncovered (i.e. water at the top of the active length) is 2.23E+04 mrem/h with the head in place and 9.30E+06 mrem/h with the head removed. Detailed results of the dose rate as a function of water height are provided in Figure 7-13 with the head removed and Figure 7-14 with the head attached.

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3. References

- "Standard Composition Library," ORNL/NUREG/CSD-2/V1/R6, Volume 3, Section M8, March 2000.
- 2. Calculation NC-6510. "Core Radionuclide Inventory for Chapter 15 Accident Analysis."
- 3. RSICC Code Package CCC-750, "SCALE 6.0: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation", Jan. 2009.
- 4. "ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms", I.C. Gauld, O.W. Hermann, & R. M. Westfall. Jan. 2009.
- 5. STP001-CPC-001. Computer Program Certification MCNP5 Version 1.4 and SCALE 6.0.
- --- 6. ENERCON email from Paul Sudnak, dated December 9, 2013. (Appendix A).
 - 7. Drawing 6C-18-N-5006, Rev. 9. "General Arrangement Reactor Containment Building Plan at El. 68' 0" Area G."
 - 8. Drawing 6C-18-9-N-5007, Rev. 6. "General Arrangement Reactor Containment Building Section A-A Area G."
 - Drawing 6C-18-9-N-5008, Rev. 8. "General Arrangement Reactor Containment Building Section B-B Area G."
 - 10. RSICC Code Package CCC-730, "MCNP/MCNPX Monte Carlo N-Particle Transport Code System 12 Including MCNP5 1.40 and MCNPX 2.5.0 and Data Libraries," January 2006.
 - 11. ANSI/ANS 6.1.1-1977, Neutron and Gamma Flux-To-Dose Conversion Factors.
 - 12. ENERCON email from Paul Sudnak, dated February 3, 2014 (Appendix A).
 - 13. Drawing L3-01EM101, Rev. 1. "Closure Head General Assembly."
 - 14. Drawing 1142E24. "Model 4XLR Reactor 173 in. I.D. Vessel."
 - Drawing 2C26-9-S-1004, Rev. 4. "Steel Reactor Containment Building Cylindrical Shell Liner Sects. And Dets. Unit N° 1 & 2."
 - 16. Drawing 1211E6. "4 Loop Rapid XL Reactor General Assembly."

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4. Assumptions

The following assumptions are used in the core uncovery dose rate calculation:

- 1. The core is homogenized based on the typical Vantage 5 fuel assembly dimensions, taking into account the fuel rods and space between. Any small variations in fuel
 - parameters will have a negligible effect on containment dose rates.
- Any non-fuel hardware is ignored since the primary self-shielding occurs in the fuel itself, and there may be some unknown streaming effects through the non-fuel hardware. This homogenization takes into account the water level when calculating the isotopic weight fraction and homogenized density.
- 3. The source term for this evaluation is based on the fission product inventory at the time of shutdown. Because there is no cooling time, the fuel gamma source term will
- predominate and the N-gamma and hardware activation can be neglected.
- The compositions of the containment structure and components are based on the values in the SCALE standard composition library [1].
- . The RE-8055 and RE-8099 monitors are assumed to be 5 feet above the 68 foot level in order to take into account the mounting device.
- 6. The containment outer concrete thickness is modeled as 3 feet thick. Because the backscattering off the containment walls is due to the steel liner, this dimension has a negligible impact on dose rates near the reactor vessel.

5. Design Inputs

5.1 Fuel Assembly Parameters

The following fuel assembly parameters are used in the core homogenization in the MCNP model. They are based on typical fuel assembly values for Westinghouse Vantage 5 fuel.

Parameters	Value	Unit	Reference
Fuel Type	Westinghouse Vantage +		Assumption 1
# Fuel Rods per Assy	264		Assumption 1
Assembly Array	17x17		Assumption 1
Enrichment	4	wt %	Assumption 1
Density (% of theoretical)	0.95.		Assumption 1
Fuel Pellet OD	0.3225	[in]	Assumption 1
Fuel Rod Pitch	0.496	[in]	Assumption 1

Table 5-1 Design Input Fuel Assembly Parameters for Westinghouse Vantage 5 Fuel

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Parameters	Value	Unit	Reference

Value	Unit	Reference
0.374	[in]	Assumption 1
0.0225	[in]	Assumption 1
0.482	[in]	Assumption 1
0.020	[in]	Assumption 1
24		Assumption 1
0.482	[in]	Assumption 1
0.020	[in]	Assumption 1
1		Assumption 1
14	[ft]	Assumption 1
	0.374 0.0225 0.482 0.020 24 0.482 0.020 1	0.374 [in] 0.0225 [in] 0.482 [in] 0.020 [in] 24 [in] 0.482 [in] 0.020 [in] 1 [in]

5.2 Containment Dimensions

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The following dimensions are based on drawings of the STP containment building and equipment. Some parameters are estimated using scaling when the drawings do not detail the exact dimension. These estimations are only applied to dimensions that have a negligible effect on the core uncovery dose rate analysis.

Table 5-2 Design	Input Containment Dimensio	ns

_Dimension:	ft	in	cm	reference
Reactor Pressure Vessel				
Elevation at top of active fuel	28	2	858.52	[6]
Elevation at head level platform	38	6.5	1174.75	[8]
Elevation at full water level in				
refueling cavity	66	6	2026.92	[8]
Closure head thickness	0	7.19	18.2626	[13]
Reactor pressure vessel inside				
diameter at shell	0	173	439.42	[14]
Height of reactor vessel from	<u></u>			
bottom of fuel to head level			742.95	Calculated
Steam Generator				
Elevation at bottom of SG	38	4	1168.4	[9]
Elevation at top of SG	105	9.875	3225.4825	[9]
Total SG height			2057.0825	Calculated
SG outer diameter		1	500	[7] Scaled

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Dimension:	ft.	in	CI	n	reference	
Active Fuel						
Active fuel bottom elevation	12	1	36	8.3	[9]	
Active fuel height	14	0	426	.72	[14]	
Concrete Wall						
Lower Height	. 38	6.5	117	4.75	[9]	
Upper Helght	85	0	259	0.8	[9]]
Overall Height			141	6.05	Calculated	
Thickness	2	0	1()6	[7] Scaled	
Width		1	874	776	[7] Scaled	
Length			249	9.36	[7] Scaled	
						-
Steam Generators	til en til som		200			
Lower Modeled Height	85	0	259	0.8	[9]	1
Upper Modeled Height	105	9.875	3225	.4825	[9]	
Overall Modeled Height			634.	6825	Calculated	
Diameter			. 50	00	[7] Sçaled	1 .
						1.
Containment						1
Upper modeled height	153	° 0	466	3.44	[8]	1
Lower modeled height	68		, 207	2.64	[8]	1
Net Height	<u>. </u>		- 259	90.8	Calculated	1
Inner Diameter	149	11 ¹ /4	45	70	[15]	1.
Liner Thickness	0	0.375	0.9	525	[15]	1
Dome Inner Radius	. 74	11 ⁵ /8	22	.85	[15]	1
Concrete Thickness	3	0	91	.44	Assumption 6	1

5.3 Core Isotopic Inventory

Core isotopic activities are provided in Table 11 of [2]. The isotope specific activities are given in terms of Ci/MWt, which is converted to curies based on the total core thermal power of 4,100 MWt [2]. These calculations are performed in EXCEL spreadsheet *STP.xlsx*. A table of the input values is shown in Table 5-3, below.

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Table 5-3 Design Basis Core Shutdown Source Term¹

lsotope	Ci/MWt	Ci	Isotope	Ci/MWt	Cl
Kr83m	3.41E+03	1.40E+07	Ru106	1.34E+04	5.49E+07
Kr85m	7.07E+03	2.90E+07	Rh105	3.05E+04	1.25E+08
Kr85	2.93E+02	1.20E+06	Zr95	4.39E+04	1.80E+08
Kr87	1.34E+04	5.49E+07	Zr97	4.39E+04	1.80E+08
Kr88	1.90E+04	7.79E+07	Nb95	4.32E+04	1.77E+08
Kr89	2.32E+04	9.51E+07	La140	4.63E+04	1.90E+08
Xe131m	2.68E+02	1.10E+06	La141	4.62E+04	1.89E+08
Xe133m	1.66E+03	6.81E+06	La142	4.15E+04	1.70E+08
Xe133	5.37E+04	2.20E+08	Pr143	3.90E+04	1.60E+08
Xe135m	1.02E+04	4.18E+07	Nd147	1.73E+04	7.09E+07
Xe135	1.34E+04	5.49E+07	Am241	2.75E+00	1.13E+04
Xe137	4.63E+04	1.90E+08	Cm242	1.05E+03	4.31E+06
Xe138	4.39E+04	1.80E+08	Cm244	6.17E+01	2.53E+05
1131	2.59E+04	1.06E+08	Ce141	4.39E+04	1.80E+08
1132	3.71E+04	1.52E+08	Ce143	4.15E+04	1.70E+08
1133	5.37E+04	2.20E+08	Ce144	3.41E+04	1.40E+08
1134	5.85E+04	2.40E+08	Np239	5.12E+05	2.10E+09
1135	4.88E+04	2.00E+08	Pu238 [.]	8.71E+01	3.57E+05
Sb127	3.05E+03	-1-25E+07-	=Pu239	1-96E+01=	=8:04E+04=
Sb129	8.29E+03	3.40E+07	Pu240	2.48E+01	1.02E+05
Te127m	4.32E+02	1.77E+06	Pu241	4.17E+03	1.71E+07
Te127	3.05E+03	1.25E+07	Rb86	9.92E+01	4.07E+05
Te129m	1.22E+03	5.00E+06	Cs134	5,37E+03	2.20E+07
Te129	8.05E+03	3.30E+07	Cs136	1.54E+03	6.31E+06
Te131m	3.66E+03	1.50E+07	Cs137	3.17E+03	1.30E+07
Te132	3.82E+04	1.57E+08	Y90	3.56E+03	1.46E+07
Ba137m	2.93E+03	1.20E+07	Y91	3.41E+04	1.40E+08
Ba139	4.98E+04	2.04E+08	Y92	3.41E+04	1.40E+08
Ba140	4.63E+04	1.90E+08	Y93	3.90E+04	1.60E+08
Mo99	4.83E+04	1.98E+08	Sr89	2.68E+04	1.10E+08
Tc99m	4.07E+04	1.67E+08	Sr90	2.37E+03	9.72E+06
Ru103	3.90E+04	1.60E+08	Sr91	3.17E+04	1.30E+08
Ru105	2.68E+04	1.10E+08	Sr92	3.41E+04	1.40E+08

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 1 Ci = Ci/MWt × 4,100 MWt

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5.4 Material Compositions

The following compositions used in the MCNP model are taken from the SCALE standard composition library [1] and are shown in Table 5-4.

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Table 5-4 SCALE Standard Compositions used in MCNP Model

Material	Isotope	Weight Fraction	Reference
Zry-4	Zr	0.9823	[1]
(6.56 g/cm^3)	Sn	0.0145	
	Cr	0.0010	
	Fe	0.0021	
	Hf	0.0001	
UO ₂	U-235	0.0353	[1]
$(10.412 \text{ g/cm}^3)^2$	U-238	0.8461	
· · · · · · · · · · · · · · · · · · ·	0	0.1186	
Air	C	0.0001	[1]
$(1.21E-03 \text{ g/cm}^3)$	N	0.7651	
	0	0.2348	
Water	H	0.1111	[1]
(0.9982 g/cm ³)	0	0.8889	
SS-304	Fe	0.6838	[1]
(7.94 g/cm^3)	Cr	0.1900	·····
	Ni	0.0950	<u> </u>
LET LA COMPANY OF THE REAL PRODUCTS AND A COMPANY AND A	Mn	0.0200	
	Si	0.0100	
	C	0.0008	
	Р	0.0004	
Concrete	0	0.5320	[1]
(2.30 g/cm^3)	Si	0.3370	
	Ca	0.0440	
	Al	0.0340	
	Na	0.0290	
	Fe	0.0140	
	Н	0.0100	
Carbon Steel	С	0.0100	[1]
(7.82 g/cm^3)	Fe	0.9900	

 2 Based on 95% of theoretical density, Assumption 1.

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6. Methodology

The reactor source terms are computed with ORIGEN-S of the SCALE 6.0 code package [3, 4]. The ORIGEN-S decay sequence is used to bin design input isotope specific activities into energy dependent photon bins. These energy specific photon emission bins are used as input for the energy distribution described by the MCNP source definitions.

The ORIGEN-S sequence in the SCALE6.0 program package is verified for use in safety related calculations [5]. The program certification form is maintained in the project file.

MCNP5, release 1.40 [10], Monte Carlo transport is used to determine the dose rates. The ENDF/B-VI neutron cross section library, ENDF60, and the ENDF/B-VI Release 8 Photo-atomic Data gamma cross section library, MCPLIB04 are utilized in the transport computations. This software has been verified for use in safety related calculations [5].

The detailed engineering drawings are converted into MCNP surface and cell cards in the proper dimensions. The radiation monitors of interest are modeled as point detectors to determine the expected dose rate for those detectors. The dose rates are calculated as a function of water height for two reactor refueling conditions:

- 1. With Head the reactor is modeled with an 7.19 inch carbon steel plate as indicated in Table 5-2, which is additional attenuation between source and detector.
- 2. Without head the reactor is modeled with nothing between the active fuel zone and containment.

For low water levels, variance reduction is accomplished with a geometric importance map that is imposed on the homogenized core. Without significant amounts of water present, this is enough to calculate statistically sound dose rate results. Once the water depth reaches a height where the variance of the results reaches an unacceptable level, a superimposed weight windows mesh is utilized to improve the variance reduction of the simple geometric scheme. The weight windows are iteratively generated using the MCNP weight windows generator card with a mesh over the existing geometry. All final dose rates presented in this calculation include weight windows variance reduction.

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7. Calculations

7.1 Source Terms

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In order to convert the isotope specific activity into an energy spectrum, ORIGEN-S of the SCALE6.0 code package is used to initiate a decay and bin into 19 photon energy groups. The energy groups along with their associated activities are used in the MCNP source definition to model the anticipated radiation emission following shutdown.

The ORIGEN-S input deck, *STPEAL.inp*, is provided below in Figure 7-1. This input has a simple decay case where the inputted isotopic composition in curies is decayed. The isotope is specified in the 73\$\$ card using the special identifier described in Section_F7.6.2 of the ORIGEN-S manual, and the activity in curies is specified in the 74** card. The time steps for the decay are given on the 60** card in years. Although multiple time steps are calculated, the source term with zero decay time is used in this calculation to model the core immediately after shutdown. The output of the decay is given in terms of photons/s/Energy-Group, which is automatically normalized in the MCNP input.

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Figure 7-1 ORIGEN-S Input Deck for MCNP Source Term Calculation

←Call Origen-S Sequence =origens 0\$\$ a11 71 e t ←Logical Ünit Assignments -Binary Photon Library (71) ←Case Title PWR Source Term STP ELA Analysis ←Library Integer Constants -Units 83** Card C1 (4) -Units 83** Card C1 (4) -Gamma Energy Groups (19) -Solution Calculation Catalogue -Cutoff Value (Default) -(a,n) Composition Dependent 56\$\$ 0 6 a6 1 a10 0 a13 66 5 3 0 2 0 e Subcase Control Constants -Decay Only Subcase (0) -Nimber of Time Informa -Number of Time Intervals -Number of Nuclides (66) -Unit of Time in Years (5) STPPAL Ci Source Terms 60** 0 0.1 0.2 0.3 0.4 0.5 -Number of Time Intervals (6) Ci Source Terms 60** 0 0.1 0.2 0.3 0.4 0.5 61** 5r1-8 1+6 1+4 65\$\$ 61** 5r1-8 1+6 1+4 65\$\$ GRAM-ATOMS GRAMS CURIES WATTS-ALL WATTS-GAMMA
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 67 32 6Z · 111 3Z 1 1 1 1 1 i 3z 6Z 81\$\$ 2 0 26 1 e ←Gamma Source Constants 82\$\$ f2 ←Produces Gamma Source Spectrum 83** 1.10E+07 1.00E+07 8.00E+06 6.50E+06 5.00E+06 4.00E+06 3.00E+06 Camma Emergy Groups 2.50E+06 2.00E+06 1.66E+06 1.33E+06 1.00E+06 8.00E+05 6.00E+05 4.00E+05 3.00E+05 2.00E+05 1.00E+05 5.00E+04 1.00E+04 e 84** 2.00E+07 6.43E+06 3.00E+06 1.85E+06 1.40E+06 9.00E+05 4.00E+05

Neutron Energy Groups ==1=00E+05=1=70E+04=3=00E+03=5=550E+02=1=00E+02=3=00E+01=1=00E+01====(Not=Used)===================== 3.05E+00 1.77E+00 1.30E+00 1.13E+00 1.00E+00 8.00E-01 4.00E-01 3.25E-01 2.25E-01 1.00E-01 5.00E-02 3.00E-02 1.00E-02 1.00E-05 e 541330 541351 541350 541370 541380 531310 531320 531330 531340 531350 511270 511290 521271 521270 521291 521290 521311 521320 561371 561390 561400 420990 430991 441030 441050 441060 451050 400950 450970 410950 571400 571410 571420 591430 601470 952410 962420 962440 581410 581430 581440 932390 942380 942390 942400 942410 370860 551340 551360 551370 390900 390910 390920 390930 380890 380900 380910 380920 74** 1.40E+07 2.90E+07 1.20E+06 5.49E+07 7.79E+07 9.51E+07 1.10E+06 ←Nuclide Concentrations (Ci) 6.81E+06 2.20E+08 4.18E+07 5.49E+07 1.90E+08 1.80E+08 1.06E+08 1.52E+08 2.20E+08 2.40E+08 2.00E+08 1.25E+07 3.40E+07 1.77E+06 1.25E+07 5.00E+06 3.30E+07 1.50E+07 1.57E+08 1.20E+07 2.04E+08 1.90E+08 1.98E+08 1.67E+08 1.60E+08 1.10E+08 5.49E+07 1.25E+08 1.80E+08 1.80E+08 1.77E+08 1.90E+08 1.89E+08 1.70E+08 1.60E+08 7.09E+07 1.13E+04 4.31E+06 2.53E+05 1.80E+08 1.70E+08 1.40E+08 2.10E+09 3.57E+05 8.04E+04 1.02E+05 1.71E+07 4.07E+05 2.20E+07 6.31E+06 1.30E+07 1.46E+07 1.40E+08 1.40E+08 1.60E+08 1.10E+08 9.72E+06 1.30E+08 1.40E+08 2-Actinide 3-Fission Product 3 3 56\$\$ f0 t End

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The results of this calculation are summarized below in Table 7-1. These values will be used in the MCNP input source definition.

Energy	Energy	Dhatana/a
Group	Boundaries	Photons/sec
	(MeV)	0.005.10
1	0.01-0.05	9.29E+19
2	0.05-0.1	2.93E+19
3	0.1-0.2	6.54E+19
4	0.2-0.3	4.28E+19
5	0.3-0.4	1.52E+19
6	0.4-0.6	3.58E+19
7	0.6-0.8	4.35E+19
8	8 0.8-1	2.66E+19
9	1-1.33	1.29E+19
10	1.33-1.66	1.65E+19
11	1.66-2	5.57E+18
12	2-2.5	5.53E+18
13	2.5-3	1.98E+18
14	<u> </u>	7.81E+17
15	4-5	3.48E+16
16	5-6.5	3.95E+11
17	6.5-8	1.75E+08
18	8-10	3.71E+07
19	10-11	2.01E+06
totals		3.95E+20

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Table 7-1 Binned Total Core Source Term

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7.2 MCNP Model Core Homogenization

Because the source term is given for the entire core, the self-shielding from the assemblies is an important part of the dose rate response in regions above the core. Particles born in the lower section of the core are very unlikely to penetrate through the core itself, and make it to the radiation monitors. For simplicity, the core is modeled as a 3 dimensional cylinder with a uniformly distributed spatial particle distribution. The calculations for the homogenization are done in the worksheet *Compositions* of the EXCEL workbook *STP.xlsx*. A density and isotopic composition is calculated with the water level above the top of the fuel. A summary of the calculations for the core composition and density is shown below. The inputs are based on the dimensions in Table 5-1 and the compositions in Table 5-4.

Rod Volume = π (Pellet Radius)² × Active Length = (3.14)(0.16125 in)²(168 in) = 13.7 in³

Rod Mass_{U0₂} =
$$\rho \times V = \left(10.96 \frac{g}{cc}\right) (0.95) (13.72 \text{ in}^3) \left(2.54 \frac{cm}{in}\right)^3 = 2341.5 \text{ g}$$

Assembly
$$Mass_{UO_2} = Rod Mass \times \frac{Number of Fuel Rods}{Assembly} = (2341.5 g)(264) = 618.2 kg$$

Clad Volume =
$$\pi \left(\frac{OD^2}{4} - \frac{ID^2}{4}\right) \times Active \ Length = (3.14) \left[\frac{(0.374 \ in)^2}{4} - \frac{(0.329 \ in)^2}{4}\right] (168 \ in)$$

= 4.17 in³

Rod Mass_{Zry-4} =
$$\rho \times V = \left(6.56 \frac{g}{cc}\right) (4.17 in^3) \left(2.54 \frac{cm}{in}\right)^3 = 448.7 g$$

Assembly $Mass_{Zry-4} = Rod Mass \times \frac{Number of Fuel Rods}{Assembly} = (448.7 g)(264) = 118.5 kg$

Assembly H_2O Volume = $[(Assembly Width)^2 - \pi (Rod Radius)^2 \times 264] \times Active Length$ = $[(8.404 in)^2 - (3.14)(0.187 in)^2(264)](168 in) = 6993 in^3$

Assembly
$$Mass_{H_2O} = \rho \times V = \left(0.9982 \frac{g}{cc}\right) (6993 in^3) \left(2.54 \frac{cm}{in}\right)^3 = 114.4 kg$$

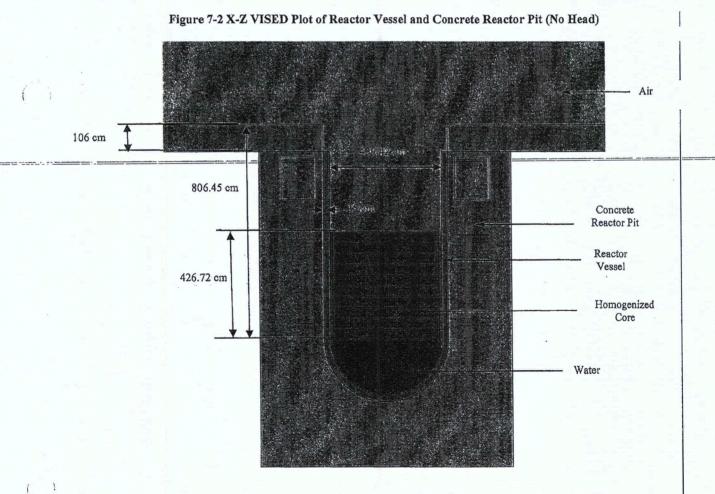
Assembly Volume = Active Length \times (Assembly Width)² = (168 in)(8.404 in)² = 11865.4 in³

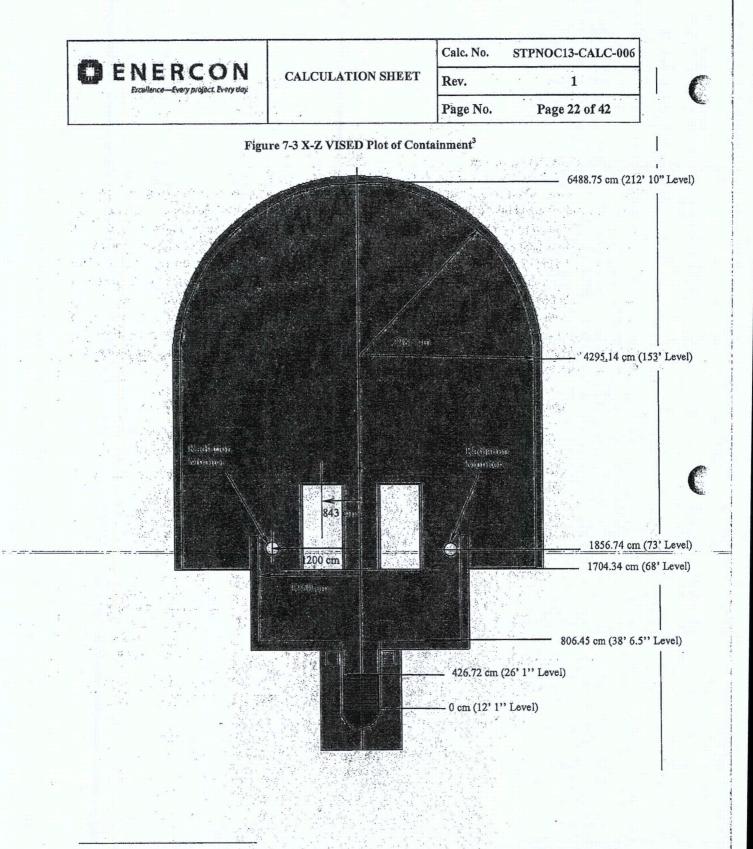
$$Density = \frac{Total Mass}{Volume} = \frac{1000(618.2 + 118.5 kg + 114.4) kg}{11865.4 in^3 \left(2.54 \frac{cm}{in}\right)^3} = 4.38 g/cc$$

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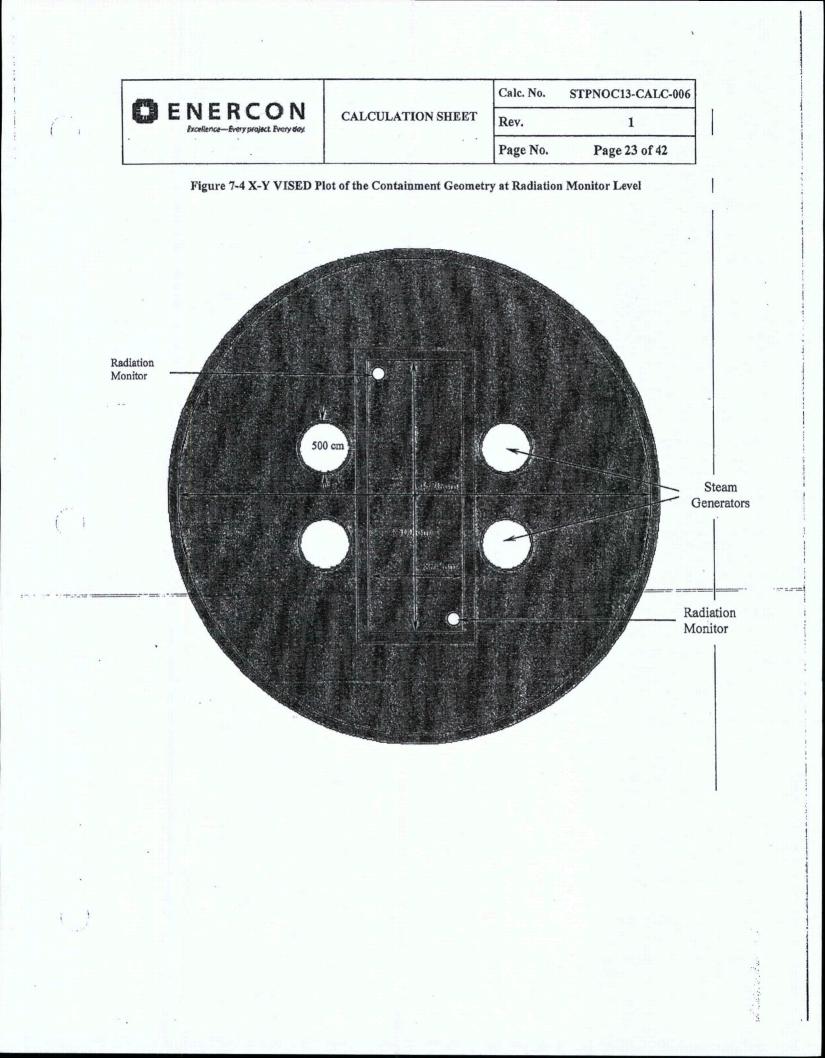
7.3 MCNP Model Geometry

The following MCNP model geometry is based on the containment dimensions summarized in Table 5-2. The model only focuses on the primary systems and components that provide shielding or reflection from the core to the radiation monitors. These components include the reactor vessel, concrete in reactor pit, containment walls (reflection), and steam generators (reflection). VISED plots of the model geometry are provided in Figure 7-2, Figure 7-3, and Figure 7-4. The MCNP surface cards with the model dimensions (cm) are shown in Figure 7-5, and the cell cards are shown in Figure 7-6 for the cases with no reactor head. A VISED plot of the model with the reactor head is shown in Figure 7-7. The surface and cell cards for the cases with the reactor head are shown in and Figure 7-8, respectively. Areas that are not of interest are given an importance of zero (white areas) so MCNP will not track particles in locations that will not contribute to the detector response. A summary of surfaces used in constructing this geometry is shown in Table 7-2, including a description of macrobody dimensions.





³ Steam Generators are not full height. Also, they are not on the same X-Z plane as the core shown above. They are included for visualization purposes.



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Table 7-2 Summary of Surfaces Used for MCNP Models Surface Surface Number Description Type Dimensions V_x Vy RCC Yo Z_o V_z R Xo 0 0 426.72 Active Fuel Region 0 0 0 209.71 1 2 0 .0 0.. 0 700.45 219.71 **Reactor Pressure Vessel Inner Surface** 0 3 .0^{.1}. 0 700.45 244.71 Reactor Pressure Vessel Outer Surface 0 0 0 31 0 0 700.45 0 0 18.26 244.71 **Reactor Pressure Vessel Head** Concrete Void for Primary Loop 512.81 0 167.64 274.71 41 0 0 0 42 0 0 512.81 0 0 167.64 411.71 Concrete Void for Primary Loop 10 0 0 700.45 0 0 106 244.71 Concrete Wall Cutout 444.71 843 700.45 0 0 2050 250 Steam Generator 1 11 230 444.71 843 720.45 0 2010 Steam Generator Inner 1 12 0 700.45 2050 250 13 -444.71 0 0 Steam Generator 2 843 14 -444.71 843 720.45 0 0 2010 230 Steam Generator Inner 2 2050 700.45 250 Steam Generator 3 15 -444.71 -843 0 0 720.45 Ó... 2010 230 16 -444.71 -843 0 Steam Generator Inner 3 444.71 700.45 0 0 2050 250 Steam Generator 4 17 -843 18 444.71 -843 720.45 0 0 2010 230 Steam Generator Inner 4 2285 2600.8 21 0 0 1694.34 0 0 **Containment Inner Liner Surface** 0 2600.8 2285.95 **Containment Inner Concrete Surface** 22 0 0 1694.34 0 2600.8 2377.39 23 0 0 1694.34 0 0 Containment Outer Concrete Surface -X X -Y Y -Z Z RPP 1.1 -498 4 -498 498 -498 498 700.45 **Concrete Surrounding RPV** 806.45 2116.45 **Concrete Wall Fuel Pit Inner** 8 -1250 1250 -437 437 9 -1356 1356 -543 543 700.45 2116.45 **Concrete Wall Fuel Pit Outer**

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Surface Type	Surface Number		Dimensions		Dimensions	Description
	5	0	0	0	219.71	Bottom of Reactor Pressure Vessel Inner
	6	0	0	0	244.71	Bottom of Reactor Pressure Vessel Outer
	24	0	0	4295.14	2285	Containment Dome Inner Liner Surface
	25	0	0	4295.14	2285.95	Containment Dome Inner Concrete Surface
	26	0	0	4295.14	2377.39	Containment Dome Outer Concrete Surface
₽Z		Z				
	7	. 0				Fuel Bottom
	71	700.45				Top of RPV
	20	Variable				Water Level '
	27	4295.14				Spring Line
	28	1704.34				68' Level
	101-110	42.672	-	426.72		Geometric Importance Divisions in Active Zone

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Figure 7-5 MCNP Model Surface Cards⁴

c surfaces 1 rcc 0 0 0 0 0 426.72 209.71 \$ Active Fuel Region 2 rcc 0 0 0 0 0 700.45 219.71 \$ Reactor Pressure Vessel Inner Surface \$ Reactor Pressure Vessel Outer Surface 3 rcc 0 0 0 0 0 700.45 244.71 31 rcc 0 0 700.45 0 0 18.26 244.71 \$ Reactor Vessel Head 4 rpp -498 498 -498 498 -498 700.45 \$ Concrete Surrounding RPV \$ Concrete Void for Primary Loop 41 rcc 0 0 512.81 0 0 167.64 274.71 42 rcc 0 0 512.81 0 0 167.64 411.71 \$ Concrete Void for Primary Loop 5 sph 0 0 0 219.71 \$ Bottom of Reactor Pressure Vessel 6 sph 0 0 0 244.71 \$ Bottom of Reactor Pressure Vessel 7 pz 0 \$ Bottom of Active Zone \$ Top of RPV 71 pz 700.45 8 rpp -1250 1250 -437 437 806.45 2116.45 \$ Concrete Walls Fuel Pit Inner 9 rpp -1356 1356 -543 543 700.45 2116.45 \$ Concrete Wall Fuel Pit Outer 10 rcc 0 0 700.45 0 0 106 244.71 \$ Concrete Wall Cutout 11 rcc 444.71 843 700.45 0 0 2050 250 \$ Steam Generator 1 12 rcc 444.71 843 720.45 0 0 2010 230 \$ Inner Steam Generator 1 13 rcc -444.71 843 700.45 0 0 2050 250 \$ Steam Generator 2 14 rcc -444.71 843 720.45 0 0 2010 230 \$ Inner Steam Generator 2 15 rcc -444.71 -843 700.45 0 0 2050 250 \$ Steam Generator 3 16 rcc -444.71 -843 720.45 0 0 2010 230 \$ Inner Steam Generator 3 17 rcc 444.71 -843 700.45 0 0 2050 250 \$ Steam Generator 4 18 rcc 444.71 -843 720.45 0 0 2010 230 \$ Inner Steam Generator 4 \$ Water Elevation Surface 20 pz 365.76 21 rcc 0 0 1694.34 0 0 2600.8 2285 \$ Containment Inner Liner Surface 22 rcc 0 0 1694.34 0 0 2600.8 2285.95 \$ Containment Inner Concrete Surface 23 rcc 0 0 1694.34 0 0 2600.8 2377.39 \$ Containment Outer Concrete Surface \$ Containment Dome Inner Liner Surface 24 sph 0 0 4295.14 2285 \$ Containment Dome Inner Concrete Surface 25 sph 0 0 4295.14 2285.95 26 sph 0 0 4295.14 2377.39 \$ Containment Dome Outer Concrete Surface \$ Spring Line 27 pz-4295-14---\$ 68' Level 28 pz 1704.34 101 pz 42.672 \$ Geometric Importance Division Fuel Zone \$ Geometric Importance Division Fuel Zone 102 pz 85.344 103 pz 128.016 \$ Geometric Importance Division Fuel Zone 104 pz 170.688 \$ Geometric Importance Division Fuel Zone \$ Geometric Importance Division Fuel Zone 105 pz 213.36 106 pz 256.032 \$ Geometric Importance Division Fuel Zone \$ Geometric Importance Division Fuel Zone 107 pz 298.704 108 pz 341.376 \$ Geometric Importance Division Fuel Zone 109 pz 384.048 \$ Geometric Importance Division Fuel Zone \$ Geometric Importance Division Fuel Zone 110 pz 426.72

⁴ The surface cards for the MCNP model without the reactor vessel head does not have surface 31. The other surfaces are identical.

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Figure 7-6 MCNP Model Cell Cards (No Head)

c cells			
101 1 -4.57 -1 -101	<pre>imp:p=1</pre>	\$	Active Fuel Region
102 1 -4.57 -1 101 -102	<pre>imp:p=2</pre>	\$	Active Fuel Region
103 1 -4.57 -1 102 -103	imp:p=3	\$	Active Fuel Region
104 1 -4.57 -1 103 -104	imp:p=4	Ş	Active Fuel Region
105 1 -4.57 -1 104 -105	imp:p=8	\$	Active Fuel Region
106 1 -4.57 -1 105 -106	imp:p=16	\$	Active Fuel Region
107 1 -4.57 -1 106 -107	imp:p=32	Ş	Active Fuel Region
108 1 -4.57 -1 107 -108	imp:p=64	\$	Active Fuel Region
109 1 -4.57 -1 108 -109	imp:p=128	\$	Active Fuel Region
110 1 -4.57 -1 109 -110	imp:p=256	\$	Active Fuel Region
2 2 -0.9982 1 -3 #4 -20	imp:p=256	\$	Water Region
4 4 -7.94 2 -3 7 -71	imp:p=256	\$	RPV Shell
5 4 -7.94 5 -6 -7 #7	imp:p=256	\$	Bottom RPV Shell
6 2 -0.99825 -7	imp:p=256	\$	Water Above Fuel
61 2 -0.9982 -20 71 (-10:-8)	imp:p=256	\$	Water, Above Vessel Head
71 3 -1.21E-03 -42 41	imp:p=256	\$	Vold for Primary Loop
7 5 -2.3 6 3 -4 #71	imp:p=256	\$	Concrete Surrounding RPV
8 5 -2.3 8 -9 10	imp:p=256	\$	Concrete above RPV
9 4 -7.94 -11 12 28	imp:p=256	\$	Steam Generator 1
10 0 -12 28	<pre>imp:p=0</pre>	\$	Inner Steam Generator 1
11 4 -7.94 -13 14 28	imp:p=256	Ş	Steam Generator 2
12 0 -14 28	imp:p=0	`\$	Inner Steam Generator 2
13 4 -7.94 -15 16 28	imp:p=256	\$	Steam Generator 3
14 0 -16 28	imp:p=0	\$	Inner Steam Generator 3
15 4 -7.94 -17 18 28	imp:p=256	\$	Steam Generator 4
16 0 18 28	imp:p=0	\$	Inner Steam Generator 4
20 4 -7.94 21 -22	imp:p=256	Ş	Containment Liner
21 5 -2.3 22 -23	imp:p=256	\$	Containment Wall
 =22=47-94-24=-25-27	imp+p=256	- \$	-Containment Dome Liner
23 5 -2.3 25 -26 27	imp:p=256	\$	Containment Dome Concrete
24 5 -2.3 -21 -28 9 #21 #22 11 13			
15 17	imp:p=256	\$	68 foot level
30 3 -1.21E-03 (-24:-21:-8:-10:-2)			
11 13 15 17 20 #8 #24 #2 1	imp:p=256	\$	Air in Containment
999 0 1 #2 #4 #5 #6 #7 #71 #8 #9 #10			
#11 #12 #13 #14 #15 #16 #20 #21			
#22 #23 #24 #30 #61	imp:p=0	\$	Problem Boundary

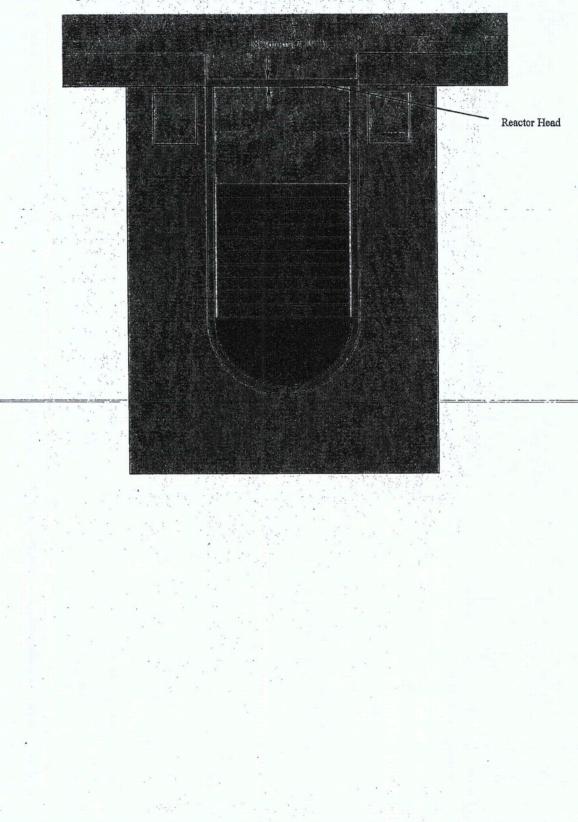
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Figure 7-7 X-Z VISED Plot of Reactor Vessel and Concrete Reactor Pit (With Head)



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Fig	ure 7-8 MCNP Cell Cards (Wi	th Head)	
c cells			
101 1 -4.57 -1 -101	<pre>imp:p=1</pre>	6.3-4	from The D m (
102 1 -4.57 -1 101 -102	imp: p=2		ive Fuel Region
103 1 - 4.57 - 1 102 - 103	imp: p=3		ive Fuel Region
104 1 -4.57 -1 103 -104	imp: p=4		ive Fuel Region
105 1 -4.57 -1 104 -105	imp: p=8	9 ACC	ive Fuel Region
106 1 -4.57 -1 105 -106	imp:p=16	Ş ACL	ive Fuel Region
107 1 -4.57 -1 106 -107	imp:p=32		ive Fuel Region
108 1 -4.57 -1 107 -108	imp:p=64		ive Fuel Region
109 1 -4.57 -1 108 -109	imp:p=128		ive Fuel Region
110 1 - 4.57 - 1 109 - 110	imp:p=256		ive Fuel Region
2 2 -0.9982 1 -3 #4 -20 31			ive Fuel Region
4 4 -7.94 2 -3 7 -71	imp:p=256		er Region
5 4 ~7.94 5 ~6 ~7 #7	imp:p=256		Shell
6 2 -0.9982 -5 -7	imp:p=256		tom RPV Shell
62 6 -7.8212 -31	imp:p=256		er Above Fuel
61 2 -0.9982 -20 71 (-10:-8) 31	imp:p=256		ctor Vessel Head
71 3 -1.21E-03 -42 41	imp:p=256 imp:p=256		er Above Vessel Head
7 5 ~2.3 6 3 -4 #71			d for Primary Loop
8 5 -2.3 8 -9 10	imp:p=256		crete Surrounding RPV
9 4 -7.94 -11 12 28	imp:p=256		crete above RPV
10 0 -12 28	imp:p=256		am Generator 1
11 4 -7.94 -13 14 28	imp:p=0		er Steam Generator 1
12 0 -14 28	imp:p=256		am Generator 2
13 4 -7.94 -15 16 28	imp:p=0		er Steam Generator 2
14 0 -16 28	imp:p=256		am Generator 3
15 4 -7.94 -17 18 28	imp:p=0		er Steam Generator 3
16 0 -18 28	imp:p=256		am Generator 4
20 4 -7.94 21 -22	imp:p=0		er Steam Generator 4
21 5 -2.3 22 -23	imp:p=256		tainment Liner
21 5 -2.3 22 -23 22 4 -7.94 24 -25 27	imp:p=256		tainment Wall
	imp:p=256		tainment Dome Liner
23 5 -2.3 25 -26 27	imp:p=256	\$ Con	tainment Dome Concret
24 5 -2.3 -21 -28 9 #21 #22 11 13		· .	
15 17	imp:p=256	\$ 68	foot level
-30-3-=1-21E=03=(=24;=21;=8:-10;=2			······································
	imp:p=256	\$ Air	inside Containment
999 0 1 #2 #4 #5 #6 #7 #71 #8 #9	FT U		

99 0 1 #2 #4 #5 #6 #7 #71 #8 #9 #10 #11 #12 #13 #14 #15 #16 #20 #21 #22 #23 #24 #30 #61 31 <u>i</u>

i 1

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imp:p=0

\$ External to Problem

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7.4 MCNP Source Definition

The core source term is assumed to be uniformly distributed throughout the volume, and has an energy spectra based on the core inventory [2]. Only the gamma source term is taken into account for this evaluation. Because the source term is generated immediately after shutdown, the fuel gamma source term will predominate. Therefore the N-gamma and hardware activation source terms can be neglected (Assumption 3). The source is defined on the MCNP *sdef* card using distributions to define the particle location and energy. The radius of the core is defined with the *rad* parameter, which automatically creates a uniform distribution based on a cylindrical geometry. The *ext* and *axs* parameters define the direction and distance of the cylinder axis. These parameters combined define the core where the particles can be born. The *erg* parameter defines the energy spectrum of source particles and is based on the results of the ORIGEN-S calculation discussed previously. This distribution is a histogram of energies represented by activities. These are automatically normalized by MCNP to create a probability distribution. The total activity is preserved in the tally multiplier. The MCNP source definition cards are shown below in Figure 7-9. The *sb* card is a source biasing card, which in this case biases the particle generation to the upper end of the core. This is a variance reduction technique to improve the statistical certainty in the results.

Figure 7-9 MCNP Source Definition Cards

sdef rad=d1 ext=d2 axs=0 0 1 erg=d8 si1 209.71 si2 h 0 42.672 85.344 128.016 170.688 213.36 256.032 298.704	<pre>←Source Definition Card</pre>
341.376 364.048 426.72 sp2 0 1 1 1 1 1 1 1 1 1 sb2 0 0.001 0.001 0.01 0.01 0.01 0.1 0.1 1 1	←Actual Uniform Distribution ←Biased to Top Distribution
c Fuel Gamma Spectra si8 h 1.000e-002 5.000e-002 1.000e-001 2.000e-001 3.000e-001 4.	
6.000e-001 8.000e-001 1.000e+000 1.330e+000 1.660e+000 2.00 2.500e+000 3.000e+000 4.000e+000 5.000e+000 6.500e+000 8.00 1.000e+001 1.100e+001	
2.500e+000 3.000e+000 4.000e+000 5.000e+000 6.500e+000 8.00	00e+000 19 3.578E+19 ←Source Emission on Energy Basis

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7.5 MCNP Tally Specification

The tallies used in this evaluation are point detectors placed at approximate locations of radiation monitors RE-8055 and RE-8099. Point detectors are chosen because they use quasi-deterministic dose calculations that will provide better results than surface or cell based tallies that require the particles to enter those regions. The inputs to this card are the coordinates of the dose points followed by an exclusion zone (reduce variance), as well as a multiplier card, which represents the total core activity in photons/sec. The tally cards are shown in Figure 7-10.

Figure 7-10 MCNP Tally Cards

f5c RE-8055 and RE-8099	←Tally Comment Card
f5:p -1200 -400 1909.24 20	←Tally 5 (point detector)
1200 400 1909.24 20	x y z exclusion
	-1200 -400 1909.24 20
	1200 400 1909.24 20
fm5 3.947E+20	← Tally Multiplier
	(Total Activity)

In addition, the flux is multiplied by ANSI/ANS flux-dose conversion factors [11]. This is specified in MCNP using the de/df cards. These are shown in Figure 7-11.

Figure 7-11 ANSI/ANS-6.1.1-1977 Gamma Flux to Dose Conversion Factors

c Ga			ersion Facto					·
de0	01030)5, 0.7 <u></u> 10 <u></u> ,	-15 20 25-) <u></u>			
	.45 .50 .5	5.60.65	.70 .80 1. 1	1.4 1.8 2.2			to Dose Conversion	
	2.6 2.8 3.	25 3.75 4.2	25 4.75 5. 5	5.25 5.75 6.	.25			
	6.75 7.5 9). 11.						
df0	3.96B-03	5.82E-04	2,90E-04	2.58E-04	2.83E-04	3.79E-04	←Energy Dependent	
	5.01E-04	6.31E-04	7.59E-04	8.78E-04	9.85E-04	1.08E-03	Flux Multipliers	
	1.17E-03	1.27E-03	1.36E-03	1.44E-03	1.52E-03	1.68E-03		•
	1.98E-03	2.51E-03	2,99E-03	3.42E-03	3.82E-03	4.01E-03		
	4.41E-03	4.83E-03	5.23E-03	5.60E-03	5.80E-03	6.01E-03		
	6.37E-03	6.74E-03	7.11E-03	7.66E-03	8.77E-03	1.03E-02		

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7.6 MCNP Material Cards

The MCNP material cards are provided in Figure 7-12. These are based on the compositions described in Table 5-4.

• •		Figure 7-12 MCNP Material Cards ⁵
ml	92238 -	0.0245 0.5891 0.2521
	40000 - 50000 -	0.1118 0.0017
	26000 -	0.0001 0.0002 0.0211
	6012 - 1001 2 8016 1	0.0001 sitis series statis and the series of
m3	6012 -0.0001 7014 -0.7650 8016 -0.2347	
m4	6000 -0.0008 14000 -0.01	\$ \$\$ 304
	15031 -0.000 24000 -0.19 25055 -0.02	
_	26000 -0.683 28000 -0.095	serve and the server of the server and the server of the server server as the server of the
m5.	26000 -0.014 1001 -0.01 13027 -0.034	그는 그는 것 같은 것 같
	.20000 -0.044 8016 -0.532	
	14000 -0.337 <u>110230-029</u> 6012 -0.01	
1	26056 -0.99	
		en en en en fan en
	·	(a) A set of the set is the set of br>the set of the s
	• •	

⁵ Material 1 composition will change based on the water level relative to the core. This only applies to water heights below 14 feet.

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7.7 Results

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File Naming Scheme:

The MCNP input files are named with the following convention:

P-height-condition where:

P = Project (STP)

Height = water height from bottom of core (ft)

Condition = h - with headn - no head

ALC: N	Chinic		11-11-1-1-1-1-1-1-1-1-1-1-1-1-1-1-1-1-			Calc. No.	STPNOC13-CALC-006	
0		NE Excellence			CALCULATION SHEET	Rev.	1	
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7.7.1 Results without Head

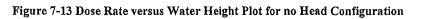
The dose rate as a function of water level is provided in Table 7-3 and plotted in Figure 7-13, below. Because the MCNP model geometry is symmetric in the x and y planes, the two point detector locations should provide the same dose rate. To increase the statistical certainty in the final result, the two individual dose rate responses and uncertainties are combined using inverse variance averaging. All of the water levels described in the following sections refer to the level at the top of the fuel (i.e. 0 foot water level is at the top of the fuel assemblies and ~13 feet is flange level).

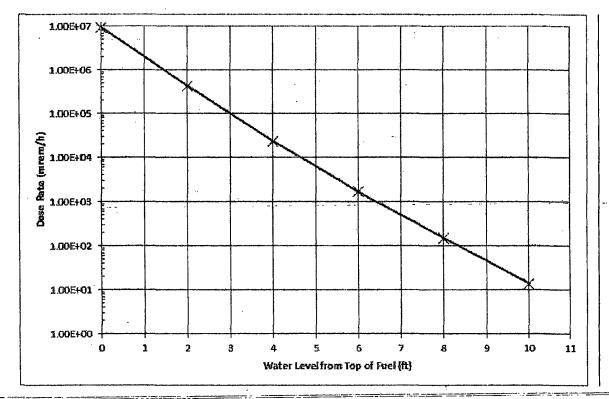
Table 7-3 Dose Rate Response as a Function of Water Level for no Head Configuration (mrem/h)

1	<u>A. A. A</u>			<u> </u>		
Water Level (ft)	Dose Rate 1	fsd	Dose Rate 2	fsd	Dose Rate Avg	Avg fsd
0.	9.27E+06	0.0081	9.34E+06	0.0109	9.30E+06	0.0065
2	4.26E+05	0.0078	4.31E+05	0.0093	4.28E+05	0.0060
4	2.31E+04	0.0236	2.32E+04	0.0247	2.32E+04	0.0171
6	1.73E+03	3.10E-02	1.69E+03	2,44E-02	1.70E+03	0.0192
8	1.51E+02	0.0302	1.51E+02	0.0287	1.51E+02	0.0208
10	1.40E+01	0.036	1.36E+01	0.0323	1.38E+01	0.0240

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7.7.2 Results with Head of a sufficiency and a set was a sufficiency of the set of the s

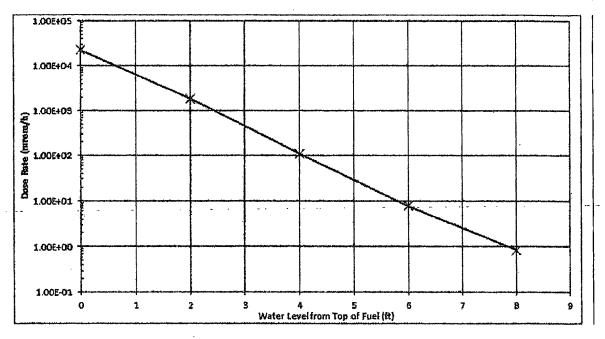
The dose rate results for the cases with the head in place are the same, except the minimum detectable dose rate is lower due to the lower ambient dose rate in the containment. The dose rates are listed in Table 17-4 and plotted in Figure 7-14.

Table.7-4 Dose Rate Response as a Function of Water Level for Head on Configuration (mrem/h)

	i da seconda de la composición de la c	나는 이 가지 않는 것이 같이 하는 것이 않아? 않아 않아? 않아 않아? 않아? 않아 않아? 않아 않아? 않아?	<u> </u>	<u></u>		
Water Level (ft)	Dose Rate 1	fsd	Dose Rate 2	fsd	Dose Rate Avg	Avg fsd
0	2.16E+04	0.094	2.56E+04	0.185	2.24E+04	0.0838
2	1.87E+03	0.083	1.83E+03	0.074	1.85E+03	0.0554
4	1.11E+02	0.061	1.08E+02	0.069	1.10E+02	0.0455
6	8.89E+00	0.085	7.48E+00		7.82E+00	0.0418
8	8.95E-01	0.125	8.12E-01	0.093	8.42E-01	0.0742

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	Page No.	• Page 37 of 42

Figure 7-14 Dose Rate versus Water Height Plot for with Head Configuration



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Appendix A – ENERCON Reference EMAILS

CALCULATION SHEET Rev. 1 Page No. Page 39 of 42			NERCON		STPNOC13-CALC-006
Page No Page 30 of 42			CALCULATION SHEET	Rev. 1	1
	•	·		Page No.	Page 39 of 42

Fronte	Paul Sudmak
Sentr	Monday, December 09, 2013 9:55 AM
Tot	Chad Cramer, Joanne Morris
Ce	Marvin Morris; Jeff Gromatzky; Michael Falkner; Jay Maisler, Caleb Trainor
Subjecti	RE: STP Refueling Cavity Level Calc

Sure, let ma find the elevation drawing for the cavity. The water level during refueling is the same water fevel as the spent fuel pool during fuel transfer. The beight of the active fuel is 28-2 loches. The vessel flange level is 39-3", and mid-loop is 32'-3".

RCB radiation monitors (RE-8055 and RE-8099) read from <1 mR/hr to 2.5 mR/hr during reflicting. If the upper-internal package or head are being removed, levels can increase to over "100mR/hr for the upper internals. Levels on the reflieling deck (68'0") at mid-loop will only increase to " 10 mR/hr with the water level that low, When the bead is being de-tensioned by worker on the bead level platform (*39 EL), dose rates at that location can read * 50 to over 100 mR/Inr. The general area dose rates from core radiation is usually less than 100 mR/hr, unless there are lots of fuel leaks or high RCS corrosion and activation products. Dose rates at the monitors at flange level are usually less than 5 mR/fir.

Paul J. Sudnak Enercon Services, Inc. 12906 Tampa Daks Boulevard SLATE 131 Temple Terrace, Fiorida 33637

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Drew Blackwell

Office: (813) 962-1800 X603 Fax: (813) 952-1881. Cell: (813) 389-0960.



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The shall and any of the standards and contain BERGON providency tolerables, which is positived contracts, in subject to comparing to BERGON, The cause is increased units of an inclusion or ends to which is a contracted. If you are not the included section of the ends, denous devide it which teaching it. You are heady notified but any deventision, devided and the original indicated and increase to the end it schedy possible and any devided of any deventision, devided by any of station takes in making to be and any device to the end it schedy possible and any devided any deventision, devided and possible, or station takes in making to the angle of any internet to the end it schedy possible and any devided and any construct the end of any of the original and any copy of white any instant. There you

From: Chad Gramer Sent: Friday, December 06, 2013 2:08 PM Tot Joanne Morris Co: Marvin Plottis: Jeff Gromatzky: Paul Suchak: Michael Falkner Subject: STP Refueling Cavity Level Calc

Joamie,

Michael Falmer has completed the STP cale and sent it to me for review. I spoke with he and left Gromabky and they indicated that he should have ability over the next week or so to do the refueling cavity level cale.

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Paul

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		CALCULATION SHEET	Rev.	1	ſ.
			Page No.	Page 40 of 42	
Drew Blackwell				and the second second	
Fram: Sent: Tot Cc Subject	Mon Cale Dom	ak, Pau) <pjøudnak@stpegs.cdm> lay, February 03, 2014 3:44 PM Trainor, Direv Blackvell al, Michael, Jay Malsier al Ascembly Dimension</pjøudnak@stpegs.cdm>			

Thanks Caleb,

Assume the only attenuation is from the materials between the detectors and the core. Distegard reflection, (don't think the SSIs are between the core and detector, nor is the containment wall. Concrete should be high density. Atmosphere should be esturated Steam at greater than 9.5 psi (containment spray initiation) and less than 56 psi (containment dealer pressure) mixed with sir at the original containment volume at STP. The detectors are for chambers. Do not include neutrops. The reactor vessel hand is arguind ⁶⁰8° thick and cactoon steel. I will get you the actual drawing of the RPY head thickness, but I think in inform the UESAR.

Again, a peer check from Mike or Jay? Paul

From: Caleb Trainor [mailtouctrainor@erercon.com] Senit: Monday, February 03, 2014 2:21 PM Tot Sudnak, Paul: Drew Blackwell Coi Donal: Michael: Jay Maisler Subject: RE: Fiel Assembly Dimension

Drew is working on CS1/CG1 where the concern is direct shine from the core due to lowered water levels and no fuel damage assumed. I think your may be thinking of the fission product barrier calcs that I'm working on.

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The second

-Caleb

From: Sudrak, Paul o<u>peudiat/#97PESS:CDM</u>> Senti Munday, February 3, 2014 3:US PM To: Caleb Trainor; Drew Blackwell Cc: Domal, Michael; Say Makler Subject: RE: Fuel Assembly Dimension

I think Caleb is right here. Once the concentration is known, the detectors are going to respond to the gases primarily above the 68' Elevation, all the rest will be significantly attenuated by the concrete floors, inner and outer Bio shield wall, steam generators, and the pressurizor. To model all of those structures would require an extensive geometry and a considerable amount of data. Our intent here is to identify the concentration of gases above the 68' El and determine the monitor response. Disregard the Steam Generators, the inter and outer Bio shield walls, and the Pressurizer. With an assumed homogeneous mix based on '20% fuel damage, the dose rates should be significant. Factoring in additional structures and elevations will not significantly change the outcome. A General Emergency will be declared.

Earlier today, I sent the location of the containment high range monitors (73' Elevation). Assume they see the volume of the reactor containment building above the 68' EL

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Mike/lay, can you give me a peer check have? Paul

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Appendix B – Electronic File Listing

Volume in drive F is My Passport Volume Serial Number is 1AEA-6007 Directory of F:\STPNOC013-CALC-006\Rev 1 03/14/2014 04:12 PM <DIR> 03/14/2014 04;12 PM <DIR> 03/21/2014 09:33 PM 0 dir.dat 100,953 EMAIL from Paul Sundak, Dec. 9 2013.pdf 02/06/2014 02:03 PM 02/07/2014 8,795 Inverse Variance Weighting.xlsx 10:26 AM 332,025 liner plate info.pdf 03/14/2014 08:44 AM <DIR> 03/21/2014 09:32 PM mcnp 03/14/2014 04:12 PM <DIR> origen 02/07/2014 12:14 PM 111,247 RE Fuel Assembly Dimension.pdf 03/14/2014 09:10 AM 462,166 RPV with core.pdf 537,808 RPV.pdf 03/14/2014 08:48 AM 03/14/2014 04:06 PM 43,842 STP.xlsx 03/14/2014 04:02 PM 1,036,800 STPNOC013-CALC-006 R1.doc 9 File(s) 2,633,636 bytes

Directory of F:\STPNOC013-CALC-006\Rev 1\mcnp

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03/21/2014	09:34 AM		18,720 STP.sx
03/21/2014	09:45 AM		4,053 STP_default.sx
03/21/2014	09:45 AM		9,744 sx.log
03/21/2014	09:45 AM		2,007 sx.var
	5 File(s)	34,661 bytes

Directory of F:\STPNOC013-CALC-006\Rev 1\mcnp\head

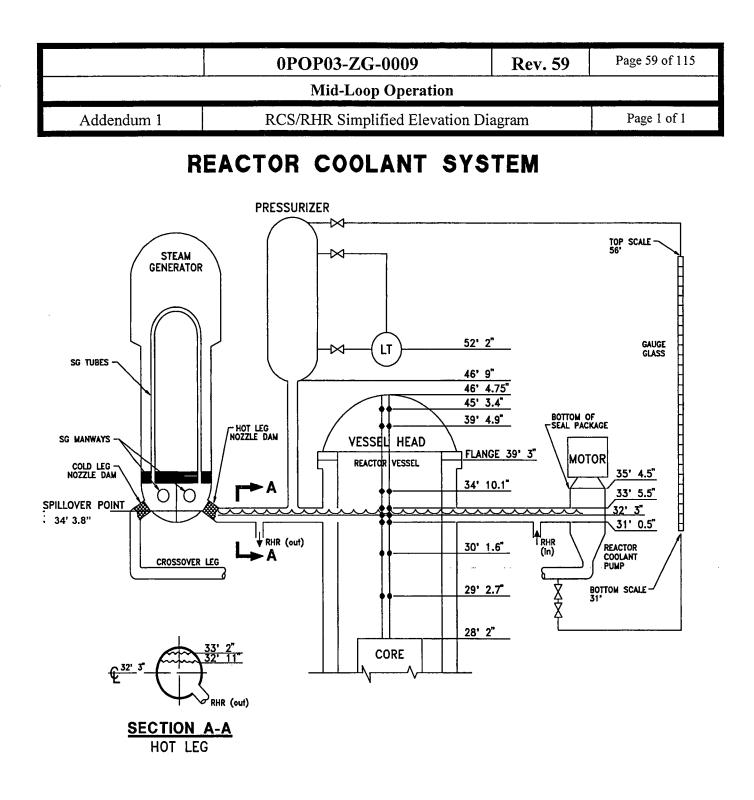
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03/12/2014	08:50	PM		1,104	STP14h5m	
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03/21/2014		AM		8,587	STP14h8	
03/21/2014	04:41	PM		1,260	STP14h8m	
03/21/2014	09:17	PM		1,312	STP14h8m2	
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03/14/2014	08:27	λM		8,990	STP16h7	
03/14/2014	03:42	PM		1,260	STP16h7m	
03/14/2014	03:42	PM		942,029	STP16h7o	
03/21/2014	09:45	AM		8,587	STP16h8	
03/21/2014	04:43	PM		1,312	STP16h8m	
03/21/2014	09:10	PM		1,364	STP16h8m2	
03/21/2014	04:43	PM		557,572	STP16h8o	
03/21/2014	09:10	PM		543,468	STP16h8o2	
03/13/2014	04:35	PM		8,990	STP18h6	
03/13/2014	08:40	PM		1,156	STP18h6m	
03/13/2014	08:40	PM		552,616	STP18h6o	
03/21/2014	09:45	AM		8,587	STP18h8	
03/21/2014	04:43	PM		1,260	STP18h8m	
03/21/2014	09:17	PM		1,312	STP18h8m2	
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03/21/2014	09:17	PM		565,735	STP18h8o2	
03/12/2014	01:17	PM		8,989	STP20h5	
03/12/2014	08:51	PM		1,104	STP20h5m	
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STP D-0794 Rev 2

	0POP03-ZG-0009	Rev. 59	Page 60 of 115
	Mid-Loop Operation		
Addendum 2	RVWL Sensor Elevations		Page 1 of 1

<u>NOTE</u>

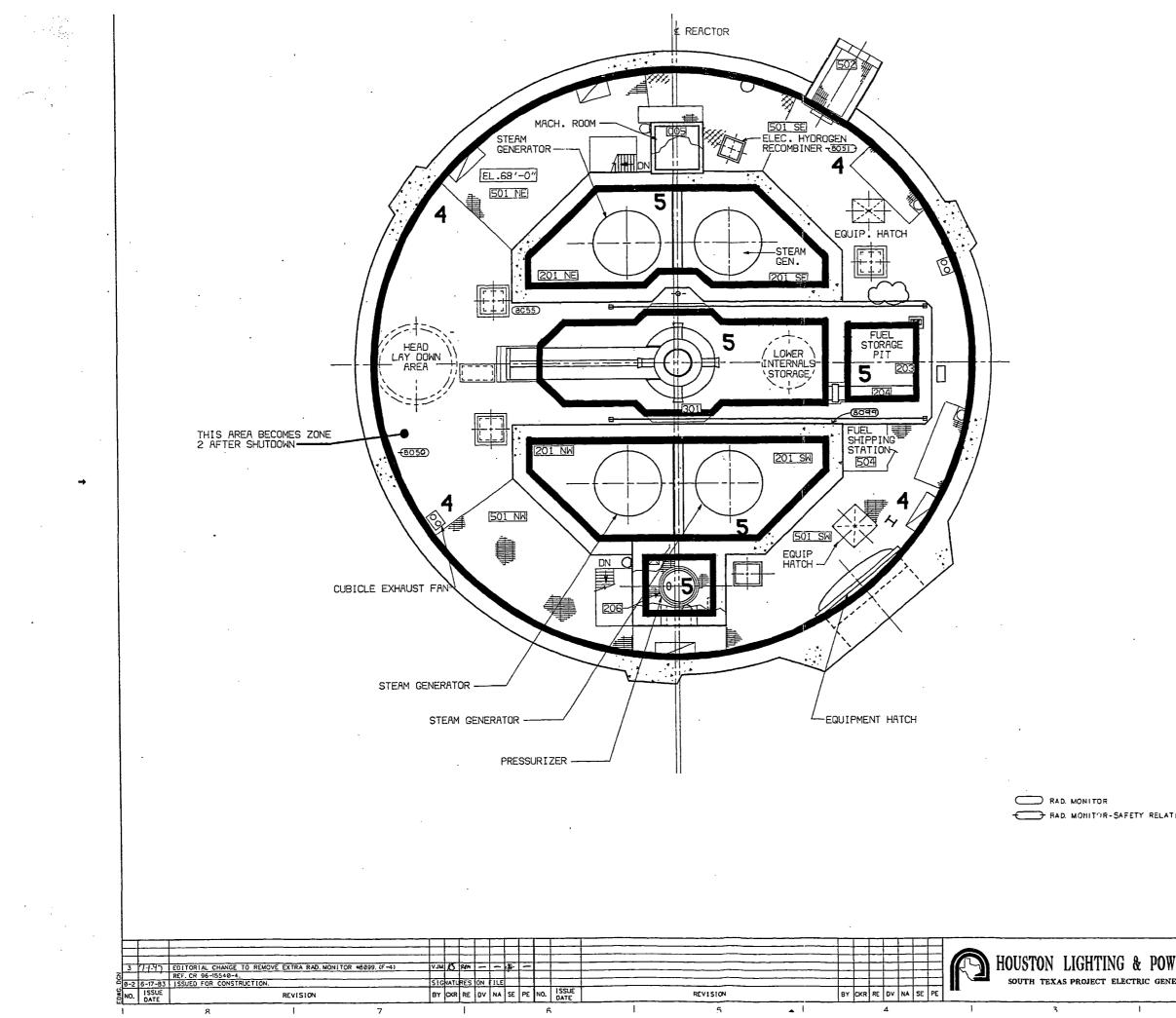
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• Top of Core is elevation 28 ft 2 inches.

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• SG spillover is elevation 34 ft 3.8 inches.

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SENSOR UNCOVERED	UPPER HEAD INDICATED LEVEL (%)	PLENUM INDICATED LEVEL (%)	SENSOR	LEVEL DESCRIPTION
All Covered	100	100	46' 4.75"	Upper Head Full
1	64	100	45' 3.4"	Upper Head Partially Drained
2	0	100	39' 4.9"	Plenum Full
3	0	85	34' 10.1"	Plenum <u>NOT</u> Full (Enter Reduced Inventory)
4	0	66	33' 5.5"	Top of Hot Leg Nozzle
5	0	48	32' 3"	Hot Leg Centerline
6	0	33	31' 0.5"	Bottom of Hot Leg Nozzle
7	0	20	30' 1.6"	Midway between Hot Leg Nozzle and Upper Core Plate
8	0	0	29' 2.7"	Upper Core Plate



HTING & POWER COMPANY	RADIATION ZONES REACTOR CONTAINMENT BUILDING PLAN AT EL. 68' -0"				
IECT ELECTRIC GENERATING STATION	SCALE	DWG. NO.	REV.	1	
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ED	ZONE NUMBER	MAX. DOS	HR.)	POSTING	ANT ICIPATED ACCESS
	1	≼ 0	.5	ND	UNRESTRICTED ACCESS
	2	≼ _2	.5	NO	CONTROLLED. 40HR/WH PERNISSIGLE
	3	\$	15	YES	CONTROLLED ACCESSIBLE ON A PERIODIC BASIS
	4	≼ 1	00	YES	CONTROLLED LIMITED ACCESS
	5	> 1	00	YES	NORMALLY INACCESSIBLE

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TABLE 12.3.4-1

AREA RADIATION MONITORS

Reactor Containment Building

Tag Number and Location ⁽¹⁾	Range (mR/hr) ⁽³⁾	High Alarm Setpoint (mr/hr) ⁽²⁾
N1RA-RE-8052 Incore Instrumentation Room (-1ft-6 in.)	$10^{-1} - 10^4$	1,000
N1RA-RE-8053 Support across from elevator (-11 ft-3 in.)	$10^{-1} - 10^4$	100
N1RA-RE-8054 West Stair Landing (19 ft-0 in.)	10 ⁻¹ -10 ⁴	100
N1RA-RE-8055 North SG wall across from the head laydown area (68 ft-0 in.)	10 ⁻¹ -10 ⁴	100
N1RA-RE-8056 Support across from elevator (52 ft-0 in.)	$10^{-1}-10^{4}$	100
N1RA-RE-8099 South SG wall across from the in-containment fuel pool (68 ft-0 in.)	10 ⁻¹ -10 ⁴	100

^{1.} Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

^{2.} The alarm setpoints listed are typical and may be varied as necessary.

^{3.} Range is based on microprocessor conversion factor and a detector signal which has a high degree of confidence. Conversion factor will vary dependent on the detector calibration. Exact ranges are found in plant instrument scaling manuals.

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Fuel Handling Building

Tag Number and Location ⁽¹⁾	Range (mR/hr)	High Alarm Setpoint (mr/hr) ⁽²⁾
N1RA-RE-8081 ~11 ft S of cols. 30.2 and S ₅ (68 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5
N1RA-RE-8084 ~24 ft S of cols. 28 and T ₅ (-21 ft-0 in.)	$10^{-1}-10^{4}$	2.5
N1RA-RE-8085 ~24 ft S of col. 28 and ~6 ft E of col. S₅ (-21 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5
N1RA-RE-8086 ~24 ft S of col. 28 and ~11 ft E of col. R ₁ (-21 ft-0 in.)	$10^{-1} - 10^{4}$	2.5
N1RA-RE-8087 col. 30.2 and 12 ft W of col. R ₁ (4 ft-0 in.)	$10^{-1} - 10^{4}$	2.5
N1RA-RE-8088 3 ft S of col. 30.9 and col. R ₁ (30 ft-0 in.)	$10^{-1} - 10^{4}$	2.5
N1RA-RE-8089 col. 28 and col. N (68 ft-0 in.)	$10^{-1} - 10^4$	2.5
N1RA-RE-8090 18 ft N of col. 30.2 and col. T ₅ (68 ft-0 in.)	$10^{-1} - 10^{4}$	2.5
N1RA-RE-8091 col. 34 and col. N (68 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Fuel Handling Building (Continued)

Tag Number and Location ⁽¹⁾	Range (mR/hr)	High Alarm Setpoint (mr/hr) ⁽²⁾

N1RA-RE-8097 33 ft S of cols. 28 and 10 ft W of col. N (68 ft-0 in.) $10^{-2} - 10^{7}$

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^{1.} Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

^{2.} The alarm setpoints listed are typical and may be varied as necessary

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Mechanical Electrical Auxiliaries Building

Tag Number and Location ⁽¹⁾	Range (mR/hr) ⁽³⁾	High Alarm Setpoint (mr/hr) ⁽²⁾
N1RA-RE-8057 col. 22 and ~10 ft E of col. J (10 ft-0 in.)	10 ⁻² -10 ³	0.5
N1RA-RE-8058 col. 26 and col. J (10 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5
NIRA-RE-8059 col. 27 and col G (10 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5
N1RA-RE-8060 ~10 ft S of col. 30 and col. E (10 ft-0 in.)	$10^{-1} - 10^{4}$	2.5
N1RA-RE-8061 ~10 ft S of col. 24 and ~11 ft W of col. E (10 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5
N1RA-RE-8062 ~6 ft S of col. 31 and col. C (10 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5
N1RA-RE-8063 ~9 ft S of col. 28 and col. B (10 ft-0 in.)	10 ⁻¹ -10 ⁴	. 2.5
N1RA-RE-8064 ~12 ft S of col. 24 and col. F (29 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5
N1RA-RE-8065 ~5 ft N of col. 32 and col. C (29 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

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3. Range is based on microprocessor conversion factor and a detector signal which has a high degree of confidence. Conversion factor will vary dependent on the detector calibration. Exact ranges are found in plant instrument scaling manuals

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Mechanical Electrical Auxiliaries Building (Continued)

Tag Number and Location ⁽¹⁾	Range (mR/hr) ⁽³⁾	High Alarm Setpoint (mr/hr) ⁽²⁾
N1RA-RE-8066 ~4 ft N of col. 22 and 14 ft E of col. C (35 ft-0 in.)	10 ⁻² -10 ³	0.5
N1RA-RE-8067 col. 22 and 10 ft E of col. J (35 ft-0 in.)	10 ⁻² -10 ³	0.5
N1RA-RE-8068 ~10 ft N of col. 25 and col. H (41 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5
N2RA-RE-8068 ~10 ft S of col. 24 and col. G (41 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5
N1RA-RE-8069 ~12 ft S of col. 24 and ~14 ft E of col. C (41 ft-0 in.)	10 ⁻² -10 ³	0.5
N1RA-RE-8070 col. 29 and col. C (41 ft-0 in.)	10 ⁻² -10 ³	2.5
N1RA-RE-8071 ~18 ft S of col. 28 and 3 ft W of col. B (41 ft-0 in.)	10 ⁻² -10 ³	2.5
N1RA-RE-8072 ~11 ft N of col. 29 and 5 ft W of col. D (41 ft-0 in.)	10 ⁻¹ -10 ⁴	100
N1RA-RE-8073 col. 29 and col. E (41 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5
N1RA-RE-8074 ~5 ft S of col. 31 and ~7 ft W of col. C (41 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

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TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Mechanical Electrical Auxiliaries Building (Continued)

Tag Number and Location ⁽¹⁾	Range (mR/hr)	High Alarm Setpoint (mr/hr) ⁽²⁾
N1RA-RE-8075 col. 28 and ~3 ft W of col. G (41 ft-0 in.)	10 ⁻¹ -10 ⁴	15.0
N1RA-RE-8076 - col. 22 and ~10 ft E of col. J (60 ft-0 in.)	10 ⁻² -10 ³	0.5
N1RA-RE-8077 col. 27 and col. J (60 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5
N1RA-RE-8078 col. 27 and col. F (60 ft-0 in.)	10^{-1} -10 ⁴	15.0
N1RA-RE-8079 col. 25 and ~2 ft W of col. F (60 ft-0 in.)	$10^{-1} - 10^{4}$	15.0
N1RA-RE-8080 col. 26 and col. H (41 ft-0 in.)	$10^{-1} - 10^4$	2.5
N1RA-RE-8082 col. 28 and ~8 ft E of col. H (69 ft-0 in.)	10 ⁻¹ -10 ⁴	2.5
N1RA-RE-8083 ~10 ft S of col. 29 and 15 ft W of col. E (41 ft-0 in.)	10 ⁻¹ -10 ⁴	15.0
N1RA-RE-8098 ~6 ft N of col. 25 and col. H (60 ft-0 in.)	10 ² -10 ⁷	1000

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

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TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Miscellaneous Buildings

Tag Number and Location ⁽¹⁾	Range (mR/hr)	High Alarm Setpoint (mr/hr) ⁽²⁾
N1RA-RE-8092 col. 9 and col. P TGB (29 ft-0 in.)	10 ⁻² -10 ³	0.5
N1RA-RE-8093 col. 7 and col. M TGB (29 ft-0 in.)	10 ⁻² -10 ³	0.5
N1RA-RE-8094 ~3 ft N of col. 23 and ~14 ft W of col. B TSC-MEAB (72 ft-0 in.)	10 ⁻² -10 ⁷	1000

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CN-2963

^{1.} Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

^{2.} The alarm setpoints listed are typical and may be varied as necessary.

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Post-Accident Monitors

Tag Number and Location (1)	Range (R/hr)	High Alarm Setpoint (R/hr) ⁽²⁾
A1RA-RE-8050 RCB (68 ft-0 in.)	10 ⁰ -10 ⁸	2000
C1RA-RE-8051 RCB (68 ft-0 in.)	10 ⁰ -10 ⁸	2000

(

^{1.} Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

^{2.} The alarm setpoints listed are typical and may be varied as necessary.

0P0P05-E0	E010 LOSS OF REACTOR OR	REV. 21	_
		PAGE 13	OF 27
۲È	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED	
	CAUTIC	<u>N</u>	
	ment H2 concentration should be conti xplosive H2 concentrations.	nuously monitored following a LOCA, to	þ
12 M	ONITOR Containment H2 Concentration:		
a	. Containment H2 - GREATER THAN OR EQUAL TO ZERO (QDPS QUAL PAMS)	a. PLACE containment H2 monitorin system in service per ADDENDUN ESTABLISHING CONTAINMENT H2 MONITORING.	
b	. H2 concentration - GREATER THAN OR EQUAL TO 0.5%	 b. PERFORM the following: 1) <u>WHEN</u> H2 concentration is GREATER THAN 0.5%, <u>THEN</u> PERFORM Step 12.c and 12.d 	
		2) GO TO Step 13.	
c	. H2 concentration - LESS THAN 4% BY VOLUME	 c. PERFORM the following: 1) CONSULT TSC staff for additional recovery actions 2) GO TO Step 13. 	5.

_____d. PLACE hydrogen recombiners in service per OPOPO2-CG-0001, ELECTRIC HYDROGEN RECOMBINERS

NAC BUT IN AND AND AND A STATE OF A REAL AND A STATE OF A REAL AND A STATE OF A STATE OF A STATE OF A STATE OF		TANK STRATES AND ADDRESS OF STRATES	
0POP04-RC-0003	Excessive RCS Leakage	Rev. 18	Page 53 of 127

Addendum 9

Basis

Basis Page 5 of 77

STEP DESCRIPTION FOR 0POP04-RC-0003 STEP 3.0

STEP: CHECK Trends For Any Of The Following Indications Of RCS Leakage:

- Rad Monitor RT8011 Particulate Rising
- Reactor Coolant Drain Tank Level Rising
- Pressurizer Relief Tank Level Rising
- RCB Normal Sump Level Rising

<u>PURPOSE</u>: To determine if leakage is from RCS and not CVCS.

<u>BASIS:</u> Indication of RT8011, RCDT, PRT or RCB Normal Sump levels rising will confirm that the leakage is from RCS and not CVCS which is normally tied to the RCS.

ACTIONS: Monitor trends from RT8011, RCDT, PRT or RCB Normal Sump.

<u>INSTRUMENTATION</u>: Level indications located on CP004 and various plant computer monitors located in control room. Radiation Monitor Computer RM-11.

CONTROL/EQUIPMENT: N/A

KNOWLEDGE: N/A

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NRC FORM 651 (10-2004) 10 CFR 72

CERTIFICATE OF COMPLIANCE FOR SPENT FUEL STORAGE CASKS

U.S. NUCLEAR REGULATORY COMMISSION

1

a U.S. Nuclear	Regulatory Co	mmission is issui	ng this Certifica	te of Compliance p	oursuant to Title 10 of the C	ode of Federal
Regulations, Part	72, "Licensing	Requirements for	Independent S	Storage of Spent N	uclear Fuel and High-Level	Radioactive Waste" (10
CFR Part 72). Th	is certificate is	issued in accord	ance with 10 C	FR 72.238, certifyi	ng that the storage design a	and contents described
					and on the basis of the Fina	
		certificate is con	ditional upon fu	Ifilling the requirem	ients of 10 CFR Part 72, as	applicable, and the
conditions specifie						
Certificate No.	Effective	Expiration Date	Docket No.	Amendment No.	Amendment Effective Date	Package Identification No.
1000	Date	luna 10	70 4000			
1032	June 13,	June 12,	72-1032	0		USA/72-1032
	2011	2051		<u> </u>	· · · · · · · · · · · · · · · · · · ·	
Issued To: (Name/A	ddress)					
Holtec Interna	tional					
Holtec Center						
555 Lincoln D	rive West				•	
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Safety Analysis Rep	ort Title		The local way w		9 <u>p</u>	
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Holtec Inter	national	0.00			T A	
		Report for the		· .	4 And	
HI-STORM	FW MPC 3	Storaĝe Syste	em		0	
					and the second	
		Les S				
		11 - C2 - C2				
This certificate is	conditioned	upon fulfilling th	se requiremen	Its of the CER Pa	tt 72 as applicable the	attached
andix A (Ter	hnical Sneci	fications) and A	hoandly B /Ar	noved Content	1.72, as applicable, the a and Design Features), a	and the conditions
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			C C C C C C C C C C C C C C C C C C C	T With a second second		
APPROVED SP	ENT FUEL S	UQRAGE CAS			AMIL -	
		2018	Mrs VII	HIN Nee		
Model No.: HI-S	STORM FW N	APC Storage Sy	stêm 24			
DES	SCRIPTION:	Alter 19	mag /			
		A states a		AANING SER		
The	HI-STORM	FW MPC Storac	e System cor	isists of the follo	wing components: (1) in	terchangeable multi-
purt	ose canister	s (MPCs) whicl	h contain the f	fuel: (2) a storage	e overpåck (HI-STORM F	W), which contains the
MP(C during stor	age: and (3) a fr	ansfer cask (I	HI-TRAC VWI w	hich contains the MPC d	uring loading unloading
and	transfer_ope	ations_The_MP	2C stores un t	o.37_pressurized	water reactor fuel asser	nblies-or-up-to-89-boiling-
wot	r reactor fue	l assemblies.				
wate				XX'		
·			o Quatamia -	omified on damage	ihad in the Final Office A	nalizaia Denert (ECAD)
ine		TVV IVIPU Storag	je System is c	entined as described	ibed in the Final Safety A	
				ion's (NRC) Safe	ty Evaluation Report (SE	rk) accompanying the
Cer	incate of Cor	npliance (CoC)				1
						· .
					welded, cylindrical canis	
		seplate, a lid, a	closure ring, a	and the canister	shell. All MPC compone	nts that may come into
					are made entirely of staii	
		nt fuel pool wate				11622 Steel OL Dassivateu I
. con	act with sper					
coni alur	act with sper ninum/alumin	um alloys. The	canister shel	l, baseplate, lid,	vent and drain port cover	plates, and closure ring
cont alur are	act with sper ninum/alumin the main con	ium alloys. The finement bound	canister shel	I, baseplate, lid, nts. All confinem	vent and drain port cover ient boundary componer	plates, and closure ring
cont alur are	act with sper ninum/alumin the main con	ium alloys. The finement bound	canister shel	l, baseplate, lid,	vent and drain port cover ient boundary componer	plates, and closure ring
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cont alur are	act with sper ninum/alumin the main con	ium alloys. The finement bound	canister shel	I, baseplate, lid, nts. All confinem	vent and drain port cover ient boundary componer	plates, and closure ring

NRC FORM	1954 U.S.N	UCLEAR REG		COMM	REION
(3-1999)		Certificate I	1032		
10 CFR 72	FOR SPENT FUEL STORAGE CASKS			10- (· - ·
JI		Amendmen	·		
 	Supplemental Sheet	Page	2	of	
	DESCRIPTION (continued)				
	There are two types of MPCs: the MPC-37 and MPC-89. The number suffix i fuel assemblies permitted to be loaded in the MPC. Both MPC models have t				ber of
	The HI-TRAC VW transfer cask provides shielding and structural protection of unloading, and movement of the MPC from the cask loading area to the storag a multi-walled (carbon steel/lead/carbon steel) cylindrical vessel with a neutron exterior and a retractable bottom lid used during transfer operations.	ge overpack	. The tra	ansfer ca	
	The HI-STORM FW storage overpack provides shielding and structural protect The overpack is a heavy-walled steel and concrete, cylindrical vessel. Its side reinforced) concrete that is enclosed between inner and outer carbon steel sh at the bottom and air outlets at the top to allow air to circulate naturally through MPC. The inner shell has supports attached to its interior surface to guide the removal and provide a means to protect the MPC confinement boundary again loadings. A loaded MPC is stored within the HI-STORM FW storage overpact	e wall consis ells. The ov h the cavity MPC durin hst impactive	ts of plai erpack h to cool th g insertio e or impl	in (un- nas air ir ne store on and ulsive	nlets
CONDITI	ONS A A				. 1
1. C	PERATING PROCEDURES	A .			
	Written operating procedures shall be prepared for handling, loading movemer maintenance. The user's site-specific written operating procedures shall be c described in Chapter 9 of the FSAR	ent, surveilla onsistent wi	ince, and th the teo	d chnical t	pasis
2. A	CCEPTANCE TESTS AND MAINTENANCE PROGRAM Written acceptance tests and a maintenance program shall be prepared consi described in Chapter 10 of the ESAR. At completion of welding the MPC shell confinement weld hellum leak test shall be performed using a hellum mass so boundary welds leakage rate test shall be performed using a hellum mass so boundary welds leakage rate test shall be performed using a hellum mass so boundary welds leakage rate test shall be performed using a hellum mass of the area repaired per ASME Code Section (1) Subsection NE, Article NB, 4450 be performed until the leakage rate acceptance criterion is met.	eakage shall	be dete	rmined a	and
3. C					
	Activities in the areas of design, purchase, fabrication, assembly, inspection, trepair, modification of structures, systems and components, and decommission shall be conducted in accordance with a Commission-approved quality assurate applicable requirements of 10 CFR Part 72, Subpart G, and which is establish with regard to the storage system	oning that ar ance progra	e import n which	ant-to-si satisfies	afety s the
· 4. H	IEAVY LOADS REQUIREMENTS				
	Each lift of an MPC, a HI-TRAC VW transfer cask, or any HI-STORM FW ove accordance to the existing heavy loads requirements and procedures of the limade. A plant-specific review of the heavy load handling procedures (under as applicable) is required to show operational compliance with existing plant s Lifting operations outside of structures governed by 10 CFR Part 50 must be Appendix A.	censed facil 10 CFR 50.5 specific hea	ity at whi 59 or 10 vy loads	ich the I CFR 72 requirer	.48, nents.

NRC FO	RM 651	J.S. NUCLEAR REGULATO	RY COMMISSION	
(3-1999)	CERTIFICATE OF COMPLIANCE	Certificate No.	1032	
10 CFR 72	FOR SPENT FUEL STORAGE CASKS	Amendment No.	0	
eren -	Supplemental Sheet	Page 3	of 4	
5.	APPROVED CONTENTS Contents of the HI-STORM FW MPC Storage System must meet the fuel	specifications given in A	Appendix B to	
6.	this certificate. DESIGN FEATURES			
	Features or characteristics for the site or system must be in accordance w	vith Appendix B to this c	ertificate.	
7.	CHANGES TO THE CERTIFICATE OF COMPLIANCE			
	The holder of this certificate who desires to make changes to the certificat (Technical Specifications) and Appendix B (Approved Contents and Design application for amendment of the certificate			
9.	 SPECIAL REQUIREMENTS FOR FIRST-SYSTEMS IN PLACE The air mass flow rate through the cask system will be determined by dire overpack cooling passages for the first HI-STORM FW MPC Cask System a heat load equal to or greater than 30 kW. The velocity will be measured MPC shell and the overpack inner shell. An analysis shall be performed validate the analytic methods and thermal performance predicted by the line Chapter 4 of the FSAR. A letter report summarizing the results of the thermal validation test and a in accordance with 10 CFR 72.4. Cask users may satisfy this requirement report submitted to the NRO by another cask user. PRE-OPERATIONAL TESTING AND TRAINING EXERCISE. 	blaced into service by Lin the annulus formed that demonstrates the n censing-basis thermal r Construction nalysis shall be submitt	any user with between the neasurements models in red to the NRC	
	A dry run training exercise of the loading closure, handling, unloading, a HI-STORM FW MPC Storage System shall be conducted by the licensee load spent fuel assemblies. The training exercise shall not be conducted run may be performed in an alternate step sequence from the actual pro performed. The dry run shall include, but is not limited to the following; a. Moving the MPC and the transfer cask into the spent fuel pool or cas	p priof to the first use of with spent fuel in the N cedures, but all steps m	IPC. The dry	
	b. Preparation of the HI-STORM FW MPC Storage System for fuel load			
	c. Selection and verification of specific fuel assemblies to ensure type of	conformance.		
	d. Loading specific assemblies and placing assemblies into the MPC (ι including appropriate independent verification.	ising a dummy fuel asso	embly),	
	 Remote installation of the MPC lid and removal of the MPC and tran cask loading pool. 	sfer cask from the spen	t fuel pool or	
	f. MPC welding, NDE inspections, pressure testing, draining, moisture helium dehydration, as applicable), and helium backfilling. (A mocku exercise.)			

g. Transfer of the MPC from the transfer cask to the overpack.

NRC FORM 651 (3-1999) 10 CFR 72 U.S. NUCLEAR REGULATORY CERTIFICATE OF COMPLIANCE FOR SPENT FUEL STORAGE CASKS Supplemental Sheet Certificate No. Amendment No. Amendment No. Page 4 h. Placement of the HI-STORM FW MPC Storage System at the ISFSI. i. HI-STORM FW MPC Storage System unloading, including flooding MPC cavity and removing welds. (A mockup may be used for this dry-run exercise.)	103 0	
ID CFR 72 FOR SPENT FUEL STORAGE CASKS Supplemental Sheet Amendment No. h. Placement of the HI-STORM FW MPC Storage System at the ISFSI. Amendment No. i. HI-STORM FW MPC Storage System unloading, including flooding MPC cavity and removing	0	
Supplemental Sheet Page 4 h. Placement of the HI-STORM FW MPC Storage System at the ISFSI. i. HI-STORM FW MPC Storage System unloading, including flooding MPC cavity and removing	······	4
 h. Placement of the HI-STORM FW MPC Storage System at the ISFSI. i. HI-STORM FW MPC Storage System unloading, including flooding MPC cavity and removing 	of	4
i. HI-STORM FW MPC Storage System unloading, including flooding MPC cavity and removing		
	MPC lid	
Any of the above steps can be omitted if they have already been successfully carried out at a site t STORM 100 System (USNRC Docket 72-1014).	o load a	i Hl-
10. AUTHORIZATION		
The HI-STORM FW MPC Storage System, which is authorized by this certificate, is hereby approv general use by holders of 10 CFR Part 50 licenses for nuclear reactors at reactor sites under the op license issued pursuant to 10 CFR 72.210, subject to the conditions specified by 10 CFR 72.212, or certificate, and the attached Appendices A and B The HI, STORM FW MPC Storage System may fabricated and used in accordance with any approved amendments to Coc No. 1032 listed in 10 CF Each of the licensed HI-STORM FW-MPC Storage System components (i.e., the MPC, overpack, cask), if fabricated in accordance with any of the approved CoC Amendments, may be used with or provided an assessments performed by the CoC holder that demonstrates design compatibility. STORM FW MPC Storage System may be installed on an ISFSI pad with the HI-STORM 100 Cas (USNRC Docket 72-1014) provided an assessment is performed by the CoC holder that demonstr compatibility.	eneral be FR 72.21 and tran ne anot The HI- k Syster	isfer her m
Dated July 14, 2011		
1. Appendix A 2. Appendix B		

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CERTIFICATE OF COMPLIANCE-NO. 1032 --

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APPENDIX A

TECHNICAL SPECIFICATIONS

FOR THE HI-STORM FW MPC STORAGE SYSTEM

5.0 ADMINISTRATIVE CONTROLS AND PROGRAMS (continued)

5.3 Radiation Protection Program

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- 5.3.1 Each cask user shall ensure that the Part 50 radiation protection program appropriately addresses dry storage cask loading and unloading, as well as ISFSI operations, including transport of the loaded OVERPACK or TRANSFER CASK outside of facilities governed by 10 CFR Part 50. The radiation protection program shall include appropriate controls for direct radiation and contamination, ensuring compliance with applicable regulations, and implementing actions to maintain personnel occupational exposures As Low As Reasonably Achievable (ALARA). The actions and criteria to be included in the program are provided below.
- 5.3.2 As part of its evaluation pursuant to 10 CFR 72.212(b)(2)(i)(C), the licensee shall perform an analysis to confirm that the dose limits of 10 CFR 72.104(a) will be satisfied under the actual site conditions and ISFSI configuration, considering the planned number of casks to be deployed and the cask contents.
- 5.3.3 Based on the analysis performed pursuant to Section 5.3.2, the licensee shall establish individual cask surface dose rate limits for the TRANSFER CASK and the OVERPACK to be used at the site. Total (neutron plus gamma) dose rate limits shall be established at the following locations:
 - a. The top of the OVERPACK.
 - b. The side OVERPACK
 - c. The side of the TRANSFER CASK
 - d. The inlet and outlet ducts on the OVERPACK
- 5.3.4 Notwithstanding the limits established in Section 5.3.3, the measured dose rates on a loaded OVERPACK or TRANSFER CASK shall not exceed the following values:
 - a. 30 mrem/hr (gamma + neutron) on the top of the OVERPACK
 - b. 300 mrem/hr (gamma + neutron) on the side of the OVERPACK, excluding inlet and outlet ducts
 - c. 3500 mrem/hr (gamma + neutron) on the side of the TRANSFER CASK
- 5.3.5 The licensee shall measure the TRANSFER CASK and OVERPACK surface neutron and gamma dose rates as described in Section 5.3.8 for comparison against the limits established in Section 5.3.3 or Section 5.3.4, whichever are lower.

Certificate of Compliance No. **1032** Appendix A

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			0ERP01-ZV-SH03	Rev. 12	Page 2 of 13				
1	Acting Security Manager								
1.0	Purpo	ose and Sc	ope						
	1.1		cedure specifies the actions to be completed declared emergency.	by the Acting Secu	arity Manager				
	1.2	Unusual	cedure implements the necessary Security e Event and for initial immediate response fo leved by the Security Manager.						
	1.3		cedure implements the requirements of the ing Station (STPEGS) Emergency Plan spec						
2.0	Respo	onsibilities	3						
	2.1		urity Force Supervisor assumes the responsi ieved. Those responsibilities include:	bilities of the Actin	g Security Manag				
		2.1.1	Directing the implementation of on-site	security emergency	response activitie				
		2.1.2	Implementing assembly and accountability	ity efforts.					
		2.1.3	Assisting with Protected and Owner Con	trolled Area evacua	ution.				
		2.1.4	Establishing special access controls.						
		2.1.5	Providing for the expedient entry/exit of	emergency vehicle	S.				
		2.1.6	Directing changes to security operations	based on radiologic	al conditions.				
		2.1.7	Determining level of compliance with cu	urrent security proce	edures.				
	Preca	utions-and	-Limitations						
	3.1	Classifie	-ZV-IN04, Assembly and Accountability ar cation or greater unless to do so would put s at anytime as dictated by conditions may o	ite personnel at risk	. The Emergency				
ı	3.2	greater	-ZV-IN05, Site Evacuation is required at a unless to do so would put site personnel at ri as dictated by conditions may order site Ev	isk. The Emergency					

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			0ERP01-ZV-SH03	Rev. 12	Page 3 of 13			
			Acting Security Manage					
4.0	Refer	ences						
	4.1	STPEGS En	nergency Plan					
	4.2	0ERP01-ZV-IN03, Emergency Response Organization Notification						
	4.3	0ERP01-ZV-IN04, Assembly and Accountability						
	4.4	0ERP01-ZV	-IN05, Site Evacuation	· · ·				
·	4.5	0ERP01-ZV-RE02, Documentation						
	4.6	0POP04-ZO-0007, Aircraft Crash Onsite						
	4.7	0PGP05-ZV-0004, Emergency Plan Implementing Procedure Users Guide						
	4.8	Security Inst	truction SI 2202, Owner Controlled Ar	ea Vehicle Patrol				
5.0	Proce	edure						
	5.1		al Event or higher emergency classific curity Manager Checklist. Use Checkl					
	5.2		by the Security Manager, provide a br vities underway using Data Sheet 2, Se					
	5.3	-	onsibilities have been transferred to th tion of Security procedures and discont	• • • •				
	5.4	During an A	lert or higher classification, ensure an	ERO Qualified EMT	is onsite.			
6.0	Supp	ort-Documents	<u>.</u>					
	6.1	Form 1, Mee	dical Emergency Information Data					
	6.2	Data Sheet 1	I, Acting Security Manager Checklist					
	6.3	Data Sheet 2	2, Security Briefing Checklist					

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				0ERP	01-ZV-TS08	3	Rev. 16	Page 3 of 20
				Sec	urity Manage	r		
. ,	3.2	0ERP01-ZV-IN05, Site Evacuation is required at a Site Area Emergency Classification or greater unless to do so would put site personnel at risk. Site Evacuation may be ordered by the Emergency Director at anytime as dictated by conditions.						
	3.3						mergency or high ncy Classification	gher classification : on.
		3.3.1			tor has ordered port response		vation of the Te	chnical
4.0	Refer	ences						
	4.1	STP Eme	ergency Pla	n ·				,
	4.2	0ERP01-	ZV-IN01, I	Emergency (Classification			
	4.3	0ERP01-	ZV-IN03, I	Emergency H	Response Orga	nization	Notification	
	4.4	_ 0ERP01-	ZV-IN04, A	Assembly an	d Accountabil	ity		
	4.5	0ERP01-	ZV-IN05, S	Site Evacuat	ion			· .
	4.6	0ERP01-	ZV-SH03,	Acting Secu	rity Manager			
	4.7	0ERP01-	ZV-RE01,	Recovery O	perations			
	4.8	0ERP01-	ZV-RE02,	Documentat	ion			
	4.9	0PGP05-	ZV-0004, I	Emergency I	Plan Implemen	ting Proc	edure Users Gu	ide
	4.10	0POP04-	ZO-0007, A	Aircraft Cras	sh Onsite			
	4.11	Security	Instruction	<u>2203, Owne</u>	er Controlled A	rea Cheo	kpoints	142
	4.12	-		-	y 2009-10, Cor gencies and Si		tions Between t Events.	he NRC and
5.0	Proce	dure						
	5.1	report to		l Unit's Tec				mergency Director Data Sheet 1, Step
	5.2	Complete	e Checklist	activities as	follows:			

- 5.2 Complete Checklist activities as follows:
 - 5.2.1 Use the right column to log the time an activity is performed.

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0POP04-SY-0001

Seismic Event

<u>NOTE</u>

- <u>Operational Basis Earthquake (OBE)</u> is defined as "That earthquake which, considering the regional and local geology and seismology and specific characteristics of local subsurface material, could reasonably be expected to affect the plant site during the operating life of the plant; vibratory ground motion for which those features of the nuclear power plant necessary to continued operation without undue risk to the health and safety of the public are designed to remain functional (10CFR100Appendix A)."
- <u>Safe Shutdown Earthquake (SSE)</u> is defined as "That earthquake which is based upon an evaluation of the maximum earthquake potential considering the regional and local geology and seismology and specific characteristics of local subsurface material. It is that earthquake which produces the maximum vibratory ground motion for which certain safety-related structures, systems, and components are designed to remain functional (10CFR100 Appendix A)." (From USFAR Section 2.5.1)
- The station design basis values for an OBE is vibratory ground motion equal to or exceeding a horizontal acceleration of 0.05g, but less than an SSE. The station design basis values for an SSE is vibratory ground motion equal to or exceeding a horizontal acceleration of 0.10g.
- The accelerometer recorded information can be analyzed and displayed using a personal computer and software supplied with the machine. This software will display the measured response spectrum to be compared with the OBE and SSE response spectrum which will determine if the OBE or SSE has been exceeded.

1.0 <u>Purpose</u>

- 1.1 This procedure provides instructions for determining if a seismic event has occurred, and the appropriate actions to ensure plant safety following an actuation of the Seismic Monitoring System. Instructions are also included for determining if the Operational Basis Earthquake (OBE) or Safe Shutdown Earthquake (SSE) limits have been exceeded.
- 1.2 This procedure is applicable in all modes.

2.0 Symptoms and Entry Conditions

- 2.1 (Unit 1 Only) "SEISMIC EVENT" alarm. (Lampbox 9M01, Window E-8)
- 2.2 (Unit 1 Only) "SEISMIC TRIGGER" triggers the "SEISMIC EVENT" alarm.
 - 2.2.1 "SEISMIC TRIGGER" This alarm indicates that an acceleration signal greater than 0.02g in the vertical or horizontal direction has been detected from the RCB Foundation seismic trigger accelerometer (0-SY-XR-0011, - 37 ft RCB Tendon Gallery AZ 295°).
- 2.3 Physical symptoms of a seismic event have been observed. (e.g., ground motion felt by plant personnel).

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Rev. 8

Seismic Event

Initials

<u>NOTE</u>

Determination of OBE or SSE should be complete within 4 hours.

- 4.4 DIRECT I&C personnel to retrieve all seismic instrumentation data per 0PSP02-SY-0012, Seismic Monitoring Data Retrieval and System Functional Test.
- 4.5 DIRECT I&C to provide ERO TECH Support I&C Engineers with data.
- 4.6 DIRECT ERO TECH Support I&C Engineers to determine if OBE or SSE was exceeded per 0PSP09-SY-0001, Seismic Monitoring Data Analysis.

<u>NOTE</u>

The information from the National Earthquake Information Center is just for confirmation that a Seismic Event or strong explosion happened; not to be used to determine OBE or SSE unless on-site instrumentation not available.

4.7 CONTACT the National Earthquake Information Center in Denver Colorado at phone number (303) 273-8500 (voice) or (303) 273-8516 (tape) for confirmation that a seismic event has taken place.

NOTE

- Modified Mercalli Intensity VI as defined by the USGS: Felt by nearly everyone; many awakened. Some dishes, windows broken. Unstable objects overturned. Pendulum clocks may stop.
- National Earthquake Information Center will list magnitude of earthquakes and distance from nearby cities in km.
 - 4.8 <u>IF</u> Seismic Monitoring Data can <u>NOT</u> be obtained in Step 4.6, <u>THEN</u> any one of the following can be used as confirmation of exceeding an OBE:
 - The earthquake resulted in Modified Mercalli Intensity VI or greater within 5 km (3.1miles) of the plant,
 - The earthquake was felt within the plant and was of magnitude 6.0 or greater,
 - The earthquake was of magnitude 5.0 or greater and occurred within 200 km (124.3 miles) of the plant.

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3.4 WATER LEVEL (FLOOD) DESIGN

The methods of analysis used to determine the design basis flood are discussed in Section 2.4. These methods are consistent with the requirements of Regulatory Guide (RG) 1.59.

The protection measures used to accommodate static and dynamic flood loads on Category I structures generally fall under the category of "incorporated barriers" as specified in regulatory position C.1 of RG 1.102.

3.4.1 Flood Protection

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3.4.1.1 <u>External Flood Protection Measures for Seismic Category I Structures</u>. The flooding due to a postulated Main Cooling Reservoir (MCR) embankment breach produces the maximum water level around the power block structures as well as the controlling water elevations for buoyancy calculations. This is also the controlling phenomena in determining the maximum water level at the Essential Cooling Water Intake Structure (ECWIS). Studies and analyses on the MCR embankment have demonstrated that an adequate margin of safety can be maintained for all credible failure mechanisms (Section 2.5.6). Accordingly, mechanistic effects (such as scour and erosion) associated with a postulated failure of the MCR embankment need-not-be evaluated.

The maximum water level on a vertical face at the south end of the plant structures is El. 50.8 ft mean sea level (MSL), which is El. 22.8 ft above plant grade. This maximum elevation occurs during a quasi-steady-state condition after a breach of the MCR embankment and is based on an instantaneous removal of approximately 2,000 ft of the embankment opposite the power block structures. This maximum elevation occurs on the south face of the Fuel-Handling Building (FHB) of Unit 1. The selection of postulated embankment breach widths and the assumptions made in determining the maximum flood elevations are described in Section 2.4.4.

Total inundation of the Essential Cooling Pond (ECP) occurs only under the condition of MCR embankment breach and does not affect the safe shutdown capability of the plant. The maximum water level calculated to occur at the ECWIS is El. 40.8 ft.

Safety-related structures, systems and components listed in Table 3.2.A-1 are protected against the effects of external flooding by:

1. Being designed to withstand the maximum flood level and associated effects and remain functional (such as seismic Category I structures and the Category I auxiliary feedwater storage tank) or

2. Being housed within seismic Category I structures which are designed as in item 1, above.

Flood protection of safety-related structures, systems, and components is provided for postulated flood levels and conditions described in Section 2.4.

Seismic Category I structures are designed to withstand the maximum flood levels by:

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Having external walls and slabs of structures designed to resist the hydrostatic and hydrodynamic forces associated with surge-wave runup and steady-state water level.

2. Ensuring the overall stability of the total structure against overturning and sliding due to the hydrostatic and hydrodynamic forces associated with surge-wave runup and steady state water level, and

3. Ensuring that the total structure will not float due to buoyancy forces.

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Figure 3.4-1 shows a general section through the plant. Figure 3.4-2 shows the seismic Category I Building maximum steady-state water surface profile, and the corresponding relationship of sill elevations for entrances to seismic Category I buildings.

Table 3.4-1 shows the results of hydraulic loading and buoyancy calculations which were done for the various safety-related facilities. The water depths shown on this table were developed from the maximum water surface elevations presented in Table 2.4.4-3.

An investigation of seismic Category I structures has been made for the flood levels and associated effects as previously described. The design for gross effects upon the structure incorporates safety factors greater than 1.1. All exterior seismic Category I bullding openings are located above the maximum steady-state flood level or are equipped with watertight doors when located below this profile, except as stated below.

Exceptions to the above-stated design basis for exterior building openings in seismic Category I structures are: (1) the opening for the truck bay in the radwaste loading area of the Mechanical-Electrical Auxiliaries Building (MEAB) and (2) the opening for the rail car access in the spent fuel cask loading area of the FHB. These areas are not protected from flooding because they do not have any safety-related systems and components located near or below the maximum flood level which is required to perform any essential function. In addition, the two areas are separated from the remainder of the building by walls which do not contain openings below the maximum water surface elevation corresponding to their location. The Tendon Gallery Access Shaftcover (TGAS) is provided with a watertight cover to prevent flood waters from entering the MEAB.

The safety-related equipment in the ECWIS is protected from the effects of the design basis flood. The personnel access doors on the west wall are provided with watertight doors; all other doors and openings are above the flood level. The dividing walls and doors between the ECWIS compartments minimize the potential for the propagation of flooding from one compartment to another.

The three maintenance knockout panels in the exterior walls of the Diesel-Generator Building (DGB), which are located below the maximum water surface elevation of 45.0 ft MSL, are watertight and designed for the hydrostatic forces. Each knockout panel allows access to only one of the three separate compartments within the structure, and only one panel may be removed at one time. The dividing walls between the compartments preclude propagation of flooding from one compartment to another.

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The maintenance knockout panels in the exterior wall of the room, housing the component cooling water heat exchangers in the MEAB are located below the maximum steady-state water level shown on Figure 3.4-2. These panels are watertight. Since mechanistic effects from the MCR breach need not be evaluated, there is adequate time to replace the knockout panels for the remaining flood events of concern.

All exterior seismic Category I building wall and slab surfaces below grade are waterproofed. This conservatively protects the substructure of seismic Category I buildings from groundwater, which is expected to stabilize between El. 17 ft and 26 ft (1 to 10 ft below grade) after decommissioning of the dewatering system. No waterproofing is provided on exterior wall or slab surfaces above grade to protect against the effects of surge-wave run-up because of its short duration. All construction joints in exterior walls and slabs (except for localized areas of blockouts) are provided with waterstops to the maximum flood level for that location and can withstand hydrostatic and hydrodynamic effects.

All seismic joints between Category I structures contain dual 9-in. water stops capable of withstanding potential seismic and hydrostatic effects. Cracks in concrete are minimized by imposing strict QA and QC procedures on the quality of concrete and construction techniques.

Drains are provided with check valves such that the external flooding would not result in internal flooding through the inadvertent introduction of water through these drains into seismic Category I structures.

The duct banks are sealed so as to prevent backflow into safety-related areas. The cable in the duct banks is designed/specified for submerged installations.

Leakage from groundwater into the FHB is prevented by the use of waterproofing on exterior wall and slab surfaces located below grade. Should groundwater inleakage occur, it is handled by the pumps in the FHB sump, the three-train compartment sumps, and the transfer cart area sump. For Unit 1 only, accumulated groundwater inleakage to the 64 degree tendon buttress area drains through a penetration in the RCB tendon gallery outer wall and is collected in the tendon gallery sump.

Leakage of groundwater into the MEAB is prevented by the use of waterproofing on exterior wall and slab surfaces located below grade. Should groundwater leakage occur, it will be collected in sumps. Discharge from non-radioactive sumps are routed to the reservoir via a circulating water discharge line. Potentially radioactive discharge is pumped to the Liquid Waste Processing System (LWPS).

3.4.2 Analysis Procedures

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3.4.2.1 <u>Phenomena Considered in Design Load Calculations</u>. For external flooding, the design basis events considered in design load calculations are as described in Section 3.4.1.

3.4.2.2 <u>Flood-Force Application</u>. The design flood conditions and elevations have been determined from an analysis of the phenomena discussed in Section 3.4.1.1.

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In order to establish the controlling load conditions resulting from the embankment breach, both instantaneous surge wave runup as well as the longer term, quasi-steady-state conditions were analyzed. The wave runup condition conservatively assumes that the maximum total force perpendicular to the south face of the plant structures includes a dynamic component in addition to the associated hydrostatic forces. The quasi-steady state condition assumes that only the hydrostatic component contributes to the development of the total force for this case. The latter condition resulted in higher water surface elevations and greater hydraulic loads on power block structures.

The vertical buoyant loading condition is the force equal to the weight of water displaced by a structure. The discussion of lateral and vertical loadings is presented in the following subsections. Table 3.4-1 shows a summary of different lateral loadings at various locations around plant and ECP structures, caused by their respective controlling flood conditions. Procedures used to determine flood loadings are identified in Sections 3.4.2.2.1 and 3.4.2.2.2.

3.4.2.2.1 Lateral Loading:

3.4.2.2.1.1 <u>Lateral Loading on the Power Block Structures</u> – The analysis of the lateral force on the power block structures considered both the instantaneous wave runup and the quasisteady state conditions. This analysis determined that the maximum total lateral force on the power block structures occurs when the maximum water level is reached during the quasi-steady state condition. Table 3.4-1 shows the controlling lateral forces (hydrostatic) exerted on different power block structures. These lateral forces are treated as triangular loadings on a vertical surface, varying at a rate of 62.4 lb/ft2/ft of structure depth. The procedures used to determine the dynamic and hydrostatic loadings for the above analysis conditions are discussed below:

1. Dynamic Force

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The dynamic force on the south side of the power block structures is determined by application of linear momentum principles. The flow from the MCR is assumed to be normal to the south side of the power block structures. Therefore, the dynamic force exerted on the structures can be expressed by the following momentum equation (Ref. 3.4-2);

$F = p \cdot Q \cdot V_0$

where:

F = dynamic force normal to plant structure<math>p = density of flow Q = flow rate $V_0 = velocity of flow$

The maximum value of pQv_0 during surge formation is calculated. This is the contribution of momentum flux to the dynamic force. The contribution of the unsteadiness of momentum field is insignificant.

2. Hydrostatic Force

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