Ref: LCR 89-13

ATTACHMENT 2

INSERTS AND MARKED-UP PAGES

INSERTS FOR LCR 89-13

INSERT 1

This item intentionally blank

INSERT 2

This ACTION is deleted

INSERT 3

ACTION 28

INSERT 4

$\leq 13^{(a)}$**

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS
(continued)

INSERT 1

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
6. ~~Main Steam Line Radiation - High, High#~~	~~≤ 3.0 x full power background~~	~~≤ 3.6 x full power background~~
7. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
8. Scram Discharge Volume Water Level - High		
a. Float Switch	Elevation 110' 11.5"	Elevation 111' 0.5"
b. Level Transmitter/Trip Unit	Elevation 110' 10.5"*	Elevation 111' 4.5"*
9. Turbine Stop Valve - Closure	≤ 5% closed	≤ 7% closed
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥ 530 psig	≥ 465 psig
11. Reactor Mode Switch Shutdown Position	NA	NA
12. Manual Scram	NA	NA

*80.5" above instrument zero EL 104' 2" for Level Transmitter/Trip Unit A&B (South Header) 83.25" above instrument zero EL 103' 11.25" for Level Transmitter/Trip Unit C&D (North Header)

~~#The hydrogen water chemistry (HWC) system shall not be placed in service until reactor power reaches 20% of RATED THERMAL POWER. After reaching 20% of RATED THERMAL POWER, and prior to operating the HWC system, the normal full power background radiation level and associated trip setpoints may be increased to levels previously measured during full power operation with hydrogen injection. Prior to decreasing below 20% of RATED THERMAL POWER and after the HWC system has been shutoff, the background level and associated setpoint shall be returned to the normal full power values. If a power reduction event occurs so that the reactor power is below 20% of RATED THERMAL POWER without the required setpoint change, control rod motion shall be suspended (except for scram or other emergency actions) until the necessary setpoint adjustment is made.~~

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

4. Reactor Vessel Water Level-Low

The reactor vessel water level trip setpoint has been used in transient analyses dealing with coolant inventory decrease. The scram setting was chosen far enough below the normal operating level to avoid spurious trips but high enough above the fuel to assure that there is adequate protection for the fuel and pressure limits.

5. Main Steam Line Isolation Valve-Closure

The main steam line isolation valve closure trip was provided to limit the amount of fission product release for certain postulated events. The MSIV's are closed automatically from measured parameters such as high steam flow, ~~high steam line radiation,~~ low reactor water level, high steam tunnel temperature, and low steam line pressure. The MSIV's closure scram anticipates the pressure and flux transients which could follow MSIV closure and thereby protects reactor vessel pressure and fuel thermal/hydraulic Safety Limits.

INSERT 1

6. ~~Main Steam Line Radiation-High~~

~~The main steam line radiation detectors are provided to detect a gross failure of the fuel cladding. When the high radiation is detected, a trip is initiated to reduce the continued failure of fuel cladding. At the same time the main steam line isolation valves are closed to limit the release of fission products. The trip setting is high enough above background radiation levels to prevent spurious trips yet low enough to promptly detect gross failures in the fuel cladding.~~

7. Drywell Pressure-High

High pressure in the drywell could indicate a break in the primary pressure boundary systems or a loss of drywell cooling. The reactor is tripped in order to minimize the possibility of fuel damage and reduce the amount of energy being added to the coolant and the primary containment. The trip setting was selected as low as possible without causing spurious trips.

8. Scram Discharge Volume Water Level-High

The scram discharge volume receives the water displaced by the motion of the control rod drive pistons during a reactor scram. Should this volume fill up to a point where there is insufficient volume to accept the displaced water at pressures below 65 psig, control rod insertion would be hindered. The reactor is therefore tripped when the water level has reached a point high enough to indicate that it is indeed filling up, but the volume is still great enough to accommodate the water from the movement of the rods at pressures below 65 psig when they are tripped. The trip setpoint for each scram discharge volume is equivalent to a contained volume of approximately 35 gallons of water.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
INSERT 1 6. ~~Main Steam Line Radiation - High, High~~	~~$1, 2^{(f)}$~~	~~2~~	~~5~~
7. Drywell Pressure - High	$1, 2^{(h)}$	2	1
8. Scram Discharge Volume Water Level - High			
a. Float Switch	1, 2 $5^{(i)}$	2 2	1 3
b. Level Transmitter/Trip Unit	1, 2 $5^{(i)}$	2 2	1 3
9. Turbine Stop Valve - Closure	$1^{(j)}$	$4^{(k)}$	6
10. Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure - Low	$1^{(j)}$	$2^{(k)}$	6
11. Reactor Mode Switch Shutdown Position	1, 2 3, 4 5	2 2 2	1 7 3
12. Manual Scram	1, 2 3, 4 5	2 2 2	1 8 9

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION

ACTION 1 - Be in at least HOT SHUTDOWN within 12 hours.

ACTION 2 - Verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within one hour.

ACTION 3 - Suspend all operations involving CORE ALTERATIONS* and insert all insertable control rods within one hour.

ACTION 4 - Be in at least STARTUP within 6 hours.

ACTION 5 - ~~Be in STARTUP with the main steam line isolation valves closed within 6 hours or in at least HOT SHUTDOWN within 12 hours.~~

INSERT 2

ACTION 6 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure to less than the automatic bypass setpoint within 2 hours.

ACTION 7 - Verify all insertable control rods to be inserted within one hour.

ACTION 8 - Lock the reactor mode switch in the Shutdown position within one hour.

ACTION 9 - Suspend all operations involving CORE ALTERATIONS*, and insert all insertable control rods and lock the reactor mode switch in the SHUTDOWN position within one hour.

*Except replacement of LPRM strings provided SRM instrumentation is OPERABLE per Specification 3.9.2.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

INSERT 1

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High	NA
b. Inoperative	NA
2. Average Power Range Monitor*:	
a. Neutron Flux - Upscale, Setdown	NA
b. Flow Biased Simulated Thermal Power - Upscale	≤ 0.09**
c. Fixed Neutron Flux - Upscale	≤ 0.09
d. Inoperative	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ 0.55
4. Reactor Vessel Water Level - Low, Level 3	≤ 1.05
5. Main Steam Line Isolation Valve - Closure	≤ 0.06
6. ~~Main Steam Line Radiation - High, High~~	~~NA~~
7. Drywell Pressure - High	NA
8. Scram Discharge Volume Water Level - High	NA
a. Float Switch	NA
b. Level Transmitter/Trip Unit	NA
9. Turbine Stop Valve - Closure	≤ 0.06
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≤ 0.08#
11. Reactor Mode Switch Shutdown Position	NA
12. Manual Scram	NA

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel.

**Not including simulated thermal power time constant, 6 ± 0.6 seconds.

#Measured from start of turbine control valve fast closure.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	S/U[b],S S	S/U[c], W W	R R	2 3, 4, 5
b. Inoperative	NA	W	NA	2, 3, 4, 5
2. Average Power Range Monitor[f]:				
a. Neutron Flux - Upscale, Setdown	S/U[b],S S	S/U[c], W W	SA SA	2 3, 4, 5
b. Flow Biased Simulated Thermal Power - Upscale	S,D[g]	S/U[c], Q	W[d][e],SA,R[h]	1
c. Fixed Neutron Flux - Upscale	S	S/U[c], Q	W[d], SA	1
d. Inoperative	NA	Q	NA	1, 2, 3, 4, 5
3. Reactor Vessel Steam Dome Pressure - High	S	Q[k]	R	1, 2
4. Reactor Vessel Water Level - Low, Level 3	S	Q[k]	R	1, 2
5. Main Steam Line Isolation Valve - Closure	NA	Q	R	1
6. ~~Main Steam Line Radiation - High, High~~	~~S~~	~~Q~~	~~R~~	~~1, 2[i]~~
7. Drywell Pressure - High	S	Q[k]	R	1, 2

INSERT 1

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High				
a. Float Switch	NA	Q	R	1, 2, 5[(j)]
b. Level Transmitter/Trip Unit	S	Q[k)]	R	1, 5[(j)]
9. Turbine Stop Valve - Closure	NA	Q	R	1
10. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	NA	Q	R	1
11. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	W	NA	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) The IRM and SRM channels shall be determined to overlap for at least ½ decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least ½ decades during each controlled shutdown, if not performed within the previous 7 days.

(c) Within 24 hours prior to startup, if not performed within the previous 7 days.

(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER > 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.

(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.

(g) Verify measured core flow (total core flow) to be greater than or equal to established core flow at the existing recirculation loop flow (APRM % flow).

(h) This calibration shall consist of verifying the 6 ± 0.6 second simulated thermal power time constant.

INSERT 2

(i) ~~This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.~~

(j) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(k) Verify the tripset point of the trip unit at least once per 92 days

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE ACTUATION GROUPS OPERATED BY SIGNAL[(d)]	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[(a)]	APPLICABLE OPERATIONAL CONDITION	ACTION
3. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level - Low Low Low, Level 1	6 1	2	1, 2, 3	21
b. Main Steam Line Radiation - High, High	1, 2[(b)]	2	1, 2, 3##	21 INSERT 3
c. Main Steam Line Pressure - Low	1	2	1	22
d. Main Steam Line Flow - High	1	2/line	1, 2, 3	20
e. Condenser Vacuum - Low	1	2	1, 2**, 3**	21
f. Main Steam Line Tunnel Temperature - High	1	2/line	1, 2, 3	21
g. Manual Initiation	1, 2, 17	2	1, 2, 3	25
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. RWCU Δ Flow - High	7	1/Valve[(e)]	1, 2, 3	23
b. RWCU Δ Flow - High, Timer	7	1/Valve[(e)]	1, 2, 3	23
c. RWCU Area Temperature - High	7	6/Valve[(e)]	1, 2, 3	23
d. RWCU Area Ventilation Δ Temperature-High	7	6/Valve[(e)]	1, 2, 3	23
e. SLCS Initiation	7[(f)]	1/Valve[(e)]	1, 2, 5#	23
f. Reactor Vessel Water Level - Low Low, Level 2	7	2/Valve[(e)]	1, 2, 3	23
g. Manual Initiation	7	1/Valve[(e)]	1, 2, 3	25

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TABLE NOTATION

TRIP FUNCTION	VALVES CLOSED BY SIGNAL
3. MAIN STEAM LINE ISOLATION	
a. Reactor Vessel Water Level - Low Low Low, Level 1	1 (HV-F022A, B, C & D, HV-F028A, B, C & D, HV-F067A, B, C & D, HV-F016, HV-F019)
b. Main Steam Line Radiation - High, High	1 (as above), 2
c. Main Steam Line Presure - Low	1 (as above)
d. Main Steam Line Flow - High	1 (as above)
e. Condenser Vacuum - Low	1 (as above)
f. Main Steam Line Tunnel Temperature - High	1 (as above)
g. Manual Initiation	1 (as above), 2, 17 (SV-J004A-1, 2, 3, 4 & 5)
4. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. RWCU Δ Flow - High	7
b. RWCU Δ Flow - High, Timer	7
c. RWCU Area Temperature - High	7

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level	
1) Low Low, Level 2	NA
2) Low Low Low, Level 1	NA
b. Drywell Pressure - High	NA
c. Reactor Building Exhaust Radiation - High	NA
d. Manual Initiation	NA
2. SECONDARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level-Low Low, Level 2	NA
b. Drywell Pressure - High	NA
c. Refueling Floor Exhaust Radiation - High(b)	$\leq$ 4.0
d. Reactor Building Exhaust Radiation - High(b)	$\leq$ 4.0
e. Manual Initiation	NA
3. MAIN STEAM LINE ISOLATION	
a. Reactor Vessel Water Level - Low Low Low, Level 1	$\leq$ 1.0*/$\leq$ 13(a)**
b. Main Steam Line Radiation - High, High(a)(b)	~~[illegible]~~ INSERT 4
c. Main Steam Line Pressure - Low	$\leq$ 1.0*/$\leq$ 13(a)**
d. Main Steam Line Flow-High	$\leq$ 0.5*/$\leq$ 13(a)**
e. Condenser Vacuum - Low	NA
f. Main Steam Line Tunnel Temperature - High	NA
g. Manual Initiation	NA
4. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. RWCU Δ Flow - High	NA
b. RWCU Δ Flow - High, Timer	NA
c. RWCU Area Temperature - High	NA
d. RWCU Area Ventilation Δ Temperature - High	NA
e. SLCS Initiation	NA
f. Reactor Vessel Water Level - Low Low, Level 2	NA
g. Manual Initiation	NA
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION	
a. RCIC Steam Line Δ Pressure (Flow) - High	NA
b. RCIC Steam Line Δ Pressure (Flow) - High, Timer	NA
c. RCIC Steam Supply Pressure - Low	NA
d. RCIC Turbine Exhaust Diaphragm Pressure - High	NA

ATTACHMENT 3

COMPLIANCE WITH NRC CONDITIONS FOR REFERENCING NEDO 31400A

CONDITIONS

The NRC staff concluded that the removal of the MSLRM trips that automatically shut down the reactor and close the MSIVs is acceptable and that the Licensing Topical Report, NEDO 31400A, could be referenced in support of our amendment request provided that:

1. The assumptions with regard to input values made in the generic analysis of the LTR are bounding for the plant...

Table 1 of this attachment provides a comparison of key input parameters and Tables 2a, and 2b compare dose assessment between the Hope Creek Generating Station (HCGS) UFSAR and NEDO 31400A analysis assumptions.

---

2. Reasonable assurance is provided that significantly increased levels of radioactivity in the main steam lines will be controlled expeditiously to limit both occupational and environmental releases...

HCGS has, in place, procedures that ensure that any significant increase in the levels of radioactivity in the main steam lines is promptly controlled to limit environmental releases and on-site (occupational) exposures. Those procedures have been reviewed and will be upgraded, upon receipt of the requested amendment, to ensure their continued applicability and correctness.

---

3. The MSLRM and offgas radiation monitor setpoints are standardized at 1.5 times the nitrogen-16 background dose rate at the monitor locations and should either or both exceed their alarm setpoint, the reactor coolant will be promptly sampled to determine activity levels and the possible need for additional corrective actions...

The MSLRM setpoint is 1.5 times the N^{16} background at the monitor location. That alarm would trigger entry into the abnormal procedure, OP-AB.ZZ-203, which requires a reactor coolant sample be obtained and analyzed. The Offgas Radiation Monitor alarm is set to satisfy HCGS TS 4.11.2.7.2.b by alarming at 50% increase (1.5 times)* the nominal steady-state fission gas release from the reactor coolant, after factoring out any increases due to changes in thermal power leve representative gas sample taken from near the discharge of the main condenser air ejector and would trigger entry into one or more of the above abnormal procedures - which, in turn, prescribe further additional corrective actions.

Attachment 3, (cont'd)

* The offgas pre-treatment radiation monitor alarm is set at 1.5 times background or 10 mr/hr, whichever is greater. This 10 mr/hr caveat has been found necessary to eliminate numerous spurious alarms (with their attendant distractions of the control room operators) due to current background levels so low (4 to 5 mr/hr) that circuit noise or minor changes in offgas flowrate can initiate an alarm. The **10 mr/hr** alarm setpoint corresponds to .05% of the limit of 330 millicuries/second specified in TS 3.11.2.7. It is in accordance with this TS that the offgas radiation monitor alarm is set. Historically, as a point of reference, one leaking fuel pin has produced several thousand mr/hr levels on the offgas radiation monitor at HCGS. Therefore, the current alarm set point of 10 mr/hr provides conservative indication. As background levels increase with plant age, the 10 mr/hr alarm will eventually be supplanted by the 1.5 times background alarm setpoint.

TABLE 1

COMPARISON OF KEY ANALYSIS INPUT VALUES
HCGS UFSAR VS. NEDO 31400A

PARAMETER	NEDO 31400A VALUE (*)	HCGS UFSAR VALUE
No. of failed fuel rods	850	770
Core average power (MWt)	3579	3458 (105%)
Relative power level of failed rods (fraction)	1.5	[same]
Power level of failed rods (MWt)	0.12	0.11
Fission Product (FP) release from failed rods		
Melted	100% NG / 50% Iodines	[same]
Non-Melted	10% NG / 10% Iodines	[same]
Mass fraction of melted fuel	0.0077	[same]
% of FP transported to Main Condenser	100% NG / 10% Iodines	[same]
% airborne of FP in Main Condenser	100% NG / 10% Iodines	[same]
For CRDA without MSIV isolation, 100% of the Noble Gases (NG) are held-up in the Off-Gas Treatment system charcoal beds for a time; the Iodines are retained indefinitely in the charcoal beds.		
Main Condenser leakage (#)	1% per day	[same]
Off-Gas Treatment System HCGS specific (##)	H_2 Recombiner/Charcoal Treatment System (322,000 lbs charcoal) NO BYPASS of charcoal is possible	
Charcoal bed holdup times: [65°F/40°F dewpoint]		Kr = 35.5 hours Xe = 34.1 days
[77°F/45°F dewpoint](**)		Kr = 20.7 hours Xe = 15.3 days
H_2 flow rate to recombiner - (design capability)	50-150scfm	154 scfm
Air/Noble Gas flow rate	site specific	75 scfm
Thyroid dose conversion factor	Reg.Guide 1.109	[same]
Breathing Rates	Reg.Guide 1.3	[same]
Whole Body Dose Conversion Factor (semi-infinite cloud)	Reg.Guide 1.109	[same]

TABLE 1 - Continued

PARAMETER	NEDO 31400A VALUE (*)	HCGS UFSAR VALUE
Radiological Consequences Evaluation (***)	CONACO3	CONACO1
Dispersion Coefficient, X/Q		
0 - 2 hour Site Boundary	(###)	1.9E-04 (sec/m3)
8 - 24 hour LPZ	(###)	4.0E-05 (sec/m3)

FOOTNOTES:

(*) Except as noted in (#) and (##) below, values apply to the CRDA both with MSIV isolation and without MSIV isolation.

(#) Applies only to CRDA with MSIV isolation.

(##) Applies only to CRDA without MSIV isolation and 100% of Noble Gas source term processed through the Off-Gas Treatment System.

(**) NUREG 0016, Rev 1 values

(***) NEDO 31400A calculates the radiological consequences of a CRDA using the CONACO3 code while the HCGS UFSAR uses the earlier CONACO1 code. GE memo, DRR-89-07, dated 5/9/89, has provided fuel activity release fractions required to update the HCGS UFSAR to the CONACO3 code.

(###) NEDO 31400A uses bounding values of 2.5E-03 for CRDA analysis per SRP analysis and 5.0E-04 for CRDA w/o MSIV isolation. Dose calculations are done for the HCGS-specific X/Q values only.

TABLE 2a

CRDA DOSE COMPARISON
HCGS UFSAR VS. NEDO 31400A

ANALYSIS METHOD	2 HOUR SITE BOUNDARY DOSES (REM)			
	WHOLE BODY	% (*)	THYROID	% (*)
Present Design Basis HCGS UFSAR 15.4.9 (#)	3.11E-02	0.50	2.62E-01	0.35
NEDO 31400A Design Basis (# and ##)	2.50E-02	0.42	3.50E-01	0.47
NEDO 31400A - with NO MSIV ISOLATION (**, ##, and ***) [Charcoal Bed Temperatures]				
65°F	2.03E-02	0.34	N/A	N/A
77°F	3.50E-01	5.80	N/A	N/A

Footnotes:

(*) Percent of 25% of 10CFR100 (or 6 REM WB & 75 REM Thyroid)

(#) Design Basis is MSIV isolation w/ Noble Gas & Iodine leakage from Main Condenser.

(**) NO MSIV isolation with 100% of Noble Gas processed by Offgas Treatment System and all Iodine retained indefinitely in Charcoal Beds.

(##) HCGS-specific values used per Table 1 of this Attachment

(***) Krypton & Xenon doses obtained separately from Figures 3 & 4 of NEDO 31400A and given below. Whole Body dose is the sum of the Kr and Xe doses.

BED TEMP.	Xe DOSE (REM)	Kr DOSE (REM)
65°F	1.40E-02	0.63E-02
77°F	1.90E-01	1.60E-01

Doses were obtained from NEDO figures at a X/Q of 3.0E-04 and scaled to a X/Q of 1.9E-04 (multiplied by 0.633) to eliminate interpolation.

TABLE 2b

CRDA DOSE COMPARISON
HCGS UFSAR VS. NEDO 31400A

ANALYSIS METHOD	24 HOUR LOW POPULATION ZONE DOSES (REM)			
	WHOLE BODY	% (*)	THYROID	% (*)
Present Design Basis HCGS UFSAR 15.4.9	6.55E-03	0.11	5.52E-02	0.02
NEDO 31400A Design Basis	5.26E-03	0.09	7.37E-02	0.03
NEDO 31400A - with NO MSIV ISOLATION [Charcoal Bed Temperatures]				
65°F	4.27E-03	0.07	N/A	N/A
77°F	7.37E-02	1.20	N/A	N/A

Footnote:

(*) Percent of 25% of 10CFR100 (or 6 REM WB & 75 REM Thyroid)