CP&L

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10 CFR 50.90
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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

BRUNSWICK NUCLEAR PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 & 50-324/LICENSE NOS. DPR-71 & DPR-62
REQUEST FOR LICENSE AMENDMENTS
ELIMINATION OF MSLRM SCRAM AND ISOLATION FUNCTIONS

Gentlemen:

In accordance with the Code of Federal Regulations, Title 10, Parts 50.90 and 2.101, Carolina Power & Light Company hereby requests amendments to the Technical Specifications for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2. The proposed amendments would eliminate the scram and isolation functions of the main steam line radiation monitors (MSLRM). This change is based on both plant unique design and a BWR Owners' Group Topical Report, NEDO-31400, "Safety Evaluation for Eliminating the BWR Main Steam Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor."

Enclosure 1 provides a detailed description and the basis for the proposed amendments.

Enclosure 2 details the basis for the Company's determination that the proposed amendments do not involve a significant hazards consideration.

Enclosure 3 provides an environmental evaluation which demonstrates that the proposed amendments meet the eligibility for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental assessment needs to be prepared in connection with the issuance of the amendment.

Enclosure 4 provides the marked-up Technical Specification pages for Unit 1.

Enclosure 5 provides the marked-up Technical Specification pages for Unit 2.

Typed Technical Specification pages will be provided at a later date.

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Carolina Power & Light Company is providing, in accordance with 10 CFR 50.91(b), Mr. Dayne H. Brown of the State of North Carolina with a copy of the proposed license amendments.

Carolina Power & Light Company considers this request to be a cost-beneficial licensing action for the Brunswick Plant, with potential savings in excess of $100,000 over the life of the plant. CP&L therefore requests that the proposed amendments be given appropriate review priority. To facilitate implementation of these changes during the Unit 1 Spring, 1995 B110R1 refueling outage, CP&L requests that the amendments be issued in April, 1995, to be effective and implemented for Unit 1 upon completion of the B110R1 outage (Refuel Outage 9) and for Unit 2 upon completion of the B212R1 outage (Refuel Outage 11).

Please refer any questions regarding this submittal to Mr. R. P. Lopriore at (910) 457-2212.

Very truly yours,

Roy A. Anderson

KAH/

Enclosures:

1. Basis for Change Request
2. 10 CFR 50.92 Evaluation
3. Environmental Considerations
4. Marked-up Technical Specification Pages - Unit 1
5. Marked-up Technical Specification Pages - Unit 2

Roy A. Anderson, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge and belief; and the sources of his information are officers, employees, and agents of Carolina Power & Light Company.

Gwendolyn R. Milazzo
Notary (Seal)

My commission expires: August 12, 1996

cc: Mr. D. H. Brown, State of North Carolina
Mr. S. D. Ebneter, Regional Administrator, Region II
Mr. P. D. Milano, NRR Senior Project Manager - Brunswick Units 1 and 2
Mr. C. A. Patterson, Brunswick NRC Senior Resident Inspector
The Honorable H. Wells, Chairman - North Carolina Utilities Commission

ENCLOSURE 1

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1 AND 2
NRC DOCKETS 50-325 & 50-324
OPERATING LICENSES DPR-71 & DPR-62
ELIMINATION OF MSLRM SCRAM AND ISOLATION FUNCTIONS

BASIS FOR CHANGE REQUEST

Proposed Change

The proposed amendment would revise the Brunswick Unit 1 and Unit 2 Technical Specifications to reflect elimination of the MSLRM scram and isolation functions. As a result of the proposed amendment, the reactor scram, MSIV closure, MSL drain valve closure reactor water sample line isolation, mechanical vacuum line isolation on "High - High" radiation will be removed. The reactor water sample line isolation will be replaced with a low condenser vacuum signal. An isolation signal from the main stack radiation monitor will be provided for the mechanical vacuum pump line.

The MSLRMs will have both "High" and "High-High" radiation alarms. The setpoint for the MSLRM "High Radiation" alarm will be set at or below 1.5 times the nominal full power background adjusted for Hydrogen Water Chemistry operation. The setpoint for the condenser off-gas radiation monitor will be set at a value of 1.5 times background, but not less than 1.5 R/hr.

Basis For Proposed Change

The primary function of the Main Steam Line Radiation Monitoring (MSLRM) system is to provide early indication of gross fuel cladding failures. Four gamma-sensitive ion-chamber detectors located near the main steam lines (MSLs), just downstream of the outboard Main Steam Isolation Valves (MSIVs), monitor MSL radiation levels. As activity increases, the current design of the MSLRM logic initiates a reactor scram and a Group 1 isolation at three times the normal, full-power background radiation levels in order to contain the fission products released from the fuel. Group 1 valves consist of: 1) four inboard and four outboard MSIVs, 2) two MSL drain isolation valves, and 3) two reactor water sample isolation valves. This same MSLRM trip signal is also used to de-energize and isolate the main condenser mechanical vacuum pump line.

Carolina Power and Light believes removing the reactor scram and MSIV closure trips on high radiation in the MSLs will significantly reduce the possibility of spurious scrams and engineered safety feature (ESF) actuations. Removing these trips provides a reduction in transient initiating events which represents an estimated 0.3% reduction in core damage frequency. The reduction in scram frequency has economical benefit in avoiding an unnecessary scram and the associated plant recovery lost time. The BWROG has estimated that elimination of the main steam line scram and isolation functions would result in an annual cost savings of $25,000 per utility. The proposed change also reduces the amount of testing required for these monitors due to the elimination of the functions.

further increasing the estimated savings.

Safe operation of the plant is enhanced by elimination of the unnecessary scram and isolation of the reactor vessel. With implementation of these changes, 1) the primary heat sink (main condenser) remains available, 2) large transients on the reactor vessel, as well as challenges to the ESF, are avoided, and 3) the offgas system remains available to control the pathway of potential releases. The basis for each element of the proposed change follows.

I. ELIMINATION OF MSLRM SCRAM AND MSIV CLOSURE FUNCTIONS

The BWR Owners' Group (BWROG) issued NEDO-31400, "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor" to the NRC staff for review. In a Safety Evaluation dated May 15, 1991, the NRC staff approved the conclusions presented in NEDO-31400. CP&L has reviewed the BWROG submittal and NRC Safety Evaluation for this issue. The results of this review are summarized below.

No design basis accident (DBA) takes credit for a reactor scram attributed to high radiation in the MSLs. The only DBA which takes credit for an automatic MSLRM MSIV closure is the control rod drop accident (CRDA) when the mechanical vacuum pumps are not running (UFSAR 15.4.6). NEDO-31400 demonstrates that the occurrence of a CRDA, with the MSL high radiation isolation removed, results in offsite radiological exposures that are small fractions of 10 CFR 100 guidelines. Furthermore, the dose rates resulting from the CRDA for Brunswick Units 1 and 2, with the elimination of the scram and MSIV isolation functions, are also small fractions of the 10 CFR 100 limits. Table B1 provides the calculated dose rates for Brunswick Units 1 and 2.

The NRC Safety Evaluation for NEDO-31400 stipulated the following conditions for referencing NEDO-31400 in support of a licensing application:

1) The applicant demonstrates that the assumptions with regard to input values (including power per assembly, Chi/Q, and decay times) that are made in the generic analysis bound those for the plant.

2) The applicant includes sufficient evidence (implemented or proposed operating procedures or equivalent controls) to provide reasonable assurance that increased significant levels of radioactivity in the main steam lines will be controlled expeditiously to limit both occupational doses and environmental releases.

3) The applicant standardizes the MSLRM [Main Steam Line Radiation Monitor] and offgas [pretreatment] radiation monitor alarm setpoint at 1.5 times the nominal nitrogen-16 background dose rate at the monitor locations, and commits to promptly sample the reactor coolant to determine possible contamination levels in the plant reactor coolant and the need for additional corrective actions, if the MSLRM or offgas [pretreatment] radiation monitors or both exceed their alarm setpoints.

CONDITION 1

The Brunswick UFSAR discusses the CRDA Design Bases Accident (DBA) analysis as

presented in the original FSAR with minor revisions to reflect the use of 8X8 fuel. Subsequent to CP&L's analysis of the CRDA DBA, the Commission issued Standard Review Plan (SRP) 15.4.9, Reference 3. This SRP provided standardized guidelines for a conservative CRDA analysis. Some specific areas of difference between the SRP and the referenced inputs used in the CP&L analysis include the following:

- The extent of fuel melting
- Iodine releases
- Noble gas releases

The BWROG safety analysis methodology is presented in Reference 1. This analysis used values which enveloped data from all the participating utilities (Section 2, Paragraph a, Reference 1). Therefore, the analysis in Reference 1, performed to support this proposed amendment, has added to the conservatism provided by Reference 3.

Summarized in Table B1 are the following post CRDA DBA doses calculated for the exclusion area boundary:

- The CRDA doses as presently shown in the Brunswick UFSAR

- The SRP 15.4.9 guideline values

- The GE generic CRDA doses (with MSL isolation)

- The Brunswick doses using the GE model and Brunswick's X/Q (with MSL isolation)

- The GE generic CRDA doses (without MSL isolation)

- The Brunswick doses using the GE model and Brunswick's X/Q (without MSL isolation)

The values in Table B1 demonstrate the following:

- The values calculated by GE are well below the guideline values of the Standard Review Plan.

- When using Brunswick site specific X/Q's, the GE calculated doses become even lower.

- The new CRDA doses applicable to Brunswick for greater than 5% power are higher than those originally presented in the Brunswick FSAR. They are, however, lower than the "enveloping doses" calculated by GE and are only a small fraction of the 10 CFR 100 limits. (Item IV addresses doses at power levels less than 5%)

The parameters used in the original (FSAR) Brunswick CRDA dose analysis are enveloped by the parameters used in Reference 1, based on the comparative summary of dose parameters presented in Table B2.

Since the Brunswick specific CRDA doses are lower than the "enveloping doses" in NEDO-31400 and the dose parameters in NEDO-31400 envelope those used for the Brunswick analysis, it is concluded that the NRC's finding that the radiological released consequence is within the staff's acceptance criteria, even without the automatic MSIV trip, is applicable to the Brunswick Plant.

CONDITION 2

Existing annunciator procedures, APP UA-23 for the MSLRM "High Radiation" annunciator and APP UA-03 for Condenser Off-gas "High Radiation" annunciator, initiate actions through Emergency Operating Procedure EOP-04-RRCP which ensure that significant levels of radiation in the main steam lines are controlled expeditiously to limit both occupational doses and environmental releases.

CONDITION 3

CP&L will maintain the MSLRM alarm setting at or below 1.5 times normal N-16 background (either with or without Hydrogen Water Chemistry in operation) and therefore complies with the SER stipulation. CP&L will reduce the Condenser Off-gas Monitor Alarm Setpoint to 1.5 times the nominal N-16 background but not less than 1.5 R/hr. The minimum of 1.5 R/hr will provide for detection of releases resulting in occupational doses due to significant fuel defects, including releases associated with postulated Fuel Blockage Events discussed in the NRC Safety Evaluation (SE) for NEDO-31400. Detection of minor fuel defects which result in off-gas radiation levels of less than 1.5 R/hr will continue to be detected by the requirements of TS 3.4.5.b.2 and TS 4.11.2.7.2.c as discussed further below. CP&L believes these actions provide protection against significant occupational doses equivalent to that provided by the NRC specified reduction of the off-gas monitor setpoint to 1.5 times nominal N-16 background.

Reduction of the off-gas setpoint to 1.5 times the nominal N-16 background would result in frequent setpoint adjustments and nuisance alarms from operational changes. The Brunswick Plant normally operates with both SJAE trains in service at 50% capacity. In the event that one train becomes unavailable, the remaining train operates at 100% capacity. Single train operation results in an increase in activity seen at the monitor sample point locations of a magnitude that would cause the Condenser Off-gas Monitor to alarm if it were initially set at 1.5 times nominal N-16 background. The average background for dual SJAE train operation at Brunswick 2 for the most recent cycle has been approximately 50 mr/hr, with variations of 50 mR/hr to 100 mR/hr. For Unit 1 the average background for dual SJAE train operation for the most recent cycle has been approximately 60 - 80 mr/hr, with variations of 120 mR/hr to 180 mR/hr. The range of these variations also include the effects of changes in power level and changes in control rod patterns. Reduction of the condenser off-gas alarm setpoint to 1.5 times nominal N-16 background would necessitate unnecessary reestablishment of the alarm setpoint.

The condenser off-gas monitor setpoint is important in the prompt identification of failed fuel conditions that might not cause the MSLRM alarm to be reached and yet contribute to significant plant contamination levels. The MSLRMs provide prompt detection of gross fuel failures resulting in activity levels on the order of several kilocuries/sec at the location of the MSLRMs. Smaller releases to the reactor coolant would be masked at the MSLRM due to the high N-16 background at that location. These smaller releases are significant in

terms of plant contamination and occupational exposures.

The Fuel Blockage Event (FBE) has been postulated to cause significant fuel defects that do not cause the MSLRM setpoint to be reached and is described in detail in NEDO-10174, Rev. 1, "Consequences of a Postulated Flow Blockage Incident in a Boiling Water Reactor." The analysis in NEDO-10174 assumes a background activity level of 1 Ci/sec. Such high background activity levels at the location of the off-gas monitors would not be due to N-16 since N-16 decays during the transport time from the vessel to the condenser. These levels would more typically be the result of prior significant cladding defects and/or fuel failures.

Using the NEDO-10174 assumptions, a background activity of 1 Ci/sec would result in an off-gas exposure rate of 42,000 mr/hr or 42 R/hr. The minimum alarm setpoint proposed by CP&L for the off-gas monitor is 1.5 R/hr and corresponds to a background activity level of about 36,000 μCi/sec using the assumptions of NEDO-10174. A review of historical offgas radiation monitor data indicates that with background activity levels approaching 60,000 microcuries/sec, the off-gas monitor was reading approximately 1 R/hr. A setpoint of 1.5 R/hr would be equivalent to an activity level of about 90,000 μCi/sec. This level is less than 10% of the background activity level of 1 Ci/sec evaluated in NEDO-10174.

The current alarm setpoint of the condenser offgas radiation monitor of both Brunswick units is 11.7 R/hr. This setpoint is based on a fraction of 10 CFR 100 limits and is determined in accordance with ODCM methodology. The current setpoint is low enough to detect and annunciate increases in activity levels of the magnitude postulated to occur following the flow blockage events evaluated in NEDO-10174; however, in order to provide greater assurance that fuel damage is detected and plant contamination minimized, the condenser offgas radiation monitor alarm setpoint will be reduced.

A minimum setpoint of 1.5 R/hr for the off-gas monitor was selected and would ensure that significant releases, such as those postulated due to a FBE, are promptly detected. Prompt detection of fuel cladding defects caused by a postulated FBE is the basis for the conditions placed on the off-gas monitor setpoints by the NRC in the Safety Evaluation of NEDO-31400. Very small releases to the reactor coolant from fuel cladding defects would continue to be detected by the operator per the provisions of TS 3.4.5.b.2 and TS 4.11.2.7.2.c as discussed below. Use of a minimum setpoint of 1.5 R/hr would minimize unnecessary alarms resulting from operational transients and also minimize the need for setpoint changes.

Technical Specification Action Statement 3.4.5.b.2 requires reactor coolant sampling and isotopic analysis for iodine within 2 to 6 hours following a change in thermal power (>15% in one hour) or a change in offgas activity level at the SJAE (>10,000 μCi/s increase in one hour at activity levels <75,000 μCi/s, or >15% increase in one hour at activity levels >75,000 μCi/s).

Technical Specification Surveillance Requirement 4.11.2.7.2.c requires offgas sampling and analysis within 4 hours following an increase of greater than 50% in the Condenser Off-gas Monitor activity level, after factoring out increases due to changes in thermal power level. Sampling offgas rather than reactor coolant provides essentially the same level of indication of failed fuel as that provided by establishing the Condenser Off-gas Monitor setpoint at 1.5 times nominal N-16 background.

BNP Operating Instructions OI-03.1 and OI-03.2 require that the offgas activity level be recorded frequently to ensure timely detection of increased radiation levels and compliance with the above Technical Specifications. The chart recorders for the Condenser Off-gas Radiation Monitors are located on the Reactor-Turbine-Generator Board (RTGB), thereby ensuring trends in offgas activity are determined on a more frequent basis than every six hours.

In summary, CP&L will maintain the MSLRM alarm setting at or below 1.5 times normal N-16 background. CP&L will reduce the Condenser Off-gas Monitor Alarm Setpoint to 1.5 times the nominal N-16 background but not less than 1.5 R/hr. The minimum of 1.5 R/hr will provide for detection of releases resulting in significant occupational doses due to fuel defects including those due to the postulated Fuel Blockage Events as discussed in the NRC Safety Evaluation (SE) for NEDO-31400. Detection of possible fuel defects which result in off-gas radiation levels of less than 1.5 R/hr would occur by the requirements of TS 3.4.5.b.2 and TS 4.11.2.7.2.c. CP&L believes these actions provide protection against significant occupational doses equivalent to that provided by the NRC specified reduction of the off-gas monitor setpoint to 1.5 times nominal N-16 background without the potential for nuisance alarms caused by operational transients and frequent, unnecessary setpoint changes.

II. REMOVAL OF MSLRM MAIN STEAM LINE DRAIN ISOLATION FUNCTION

The Main Steam Line Drain valves, like the MSIVs, discharge to the main condenser. Both paths are therefore processed by the offgas system; however, the main steam line drain valves discharge to one 3-inch line while the MSIVs exhaust to the condenser through four 24-inch lines. The exhaust from the main steam line drain lines is minimal compared to the MSIVs and any releases are accounted for in the MSIV analysis.

III. REMOVAL OF MSLRM REACTOR WATER SAMPLE LINE ISOLATION FUNCTION

The discharge from the Reactor Water Sample line is routed to the Reactor Building Sampling Panel on the 50' elevation. The 3/4" sample line to the Reactor Building Sampling Panel provides a continuous 900 cc/min reactor water sample (Dwg. D-70070, sh. 1). By the time the sample has reached the analyzers, it has passed through two coolers (typically lowered to a temperature of 77°F). From the sample panel the sample is routed to the floor drain sump from where it is routed to the liquid radwaste system for processing.

The volume of reactor water flowing to the sample panel (900 cc/min) is insignificant when compared to the main steam flow rate of approximately 10.5 million pounds per hour. Additionally, a basic assumption is that noble gases remain with the steam phase and not the liquid phase (see UFSAR Section 11.1). Thus the noble gas source term from this sample line will be negligible when compared to that reaching the condenser.

The sample line will, however, contain iodine. At 77°F little iodine would be released from solution. Based on data in Reference 4, the iodine concentration in the air above the liquid in the floor drain sump will reach an equilibrium value of 0.032 percent (i.e., 0.00032) of that in the liquid phase. Since the sample inputs to the sump will be diluted with other floor drain sump inputs, the equilibrium airborne iodine concentration in this sump would

be correspondingly reduced. Thus, based on the sample volume and physical characteristics of the sample stream, any radiological source term resulting from this pathway will be negligible.

The MSLRM trip signal for the reactor water sample isolation line will be replaced with the Low Condenser Vacuum signal. The use of the Low Condenser Vacuum signal will prevent the flow of reactor water to the condenser when the steam flow path is lost. The reactor water sample function is not essential to plant safety and is isolated from the reactor pressure vessel during conditions of low reactor water level. The combination of Low Reactor Water Level 3 and Low Condenser Vacuum isolation signals for the reactor water sample line allows water samples to be taken over the largest range of plant conditions and maintains a redundant Group 1 Primary Containment Isolation System signal.

IV. REMOVAL OF MSLRM MECHANICAL VACUUM PUMP LINE ISOLATION FUNCTION

The mechanical vacuum pumps are used only when the reactor is at low power (less than 5%) and there is insufficient steam flow to operate the Steam Jet Air Ejectors. Radioactivity transported to the condenser by the steam flow as a result of a Control Rod Drop Accident could be transported to the main stack by the mechanical vacuum pumps. The increase in radiation would be detected by the MSLRMs and annunciated in the Main Control Room. Operators will be instructed, in the annunciator response procedures, to take action to stop the Mechanical Vacuum Pump(s) and isolate the Mechanical Vacuum Pump line. NUS Calculation 8T12-M-01 (Attachment B1) shows that, allowing for a 10 minute operator response time, the amount of radiation released at the site boundary (18.69 rem thyroid and 2.80 rem whole body) is within Standard Review Plan 15.4.9 limits of 75 rem thyroid and 6 rem whole body. NUS Calculation 8T12-M-02 (Attachment B2) documents that the dose received in the Main Control Room as a result of this accident is 27.2 rem thyroid and 0.18 rem whole body. This dose is less than the GDC 19 / SRP 6.4 limits of 30 rem thyroid and 5 rem whole body.

The modification associated with the proposed change would provide a non-safety related, automatic isolation of the mechanical vacuum pump line from the main stack radiation monitor. This added automatic trip of the mechanical vacuum pump(s) and isolation of the mechanical vacuum pump line will not be included in the Brunswick Technical Specifications since 1) the signal is considered to be a redundancy for the manual operator actions, and 2) the signal does not meet any of the criteria for incorporation in Technical Specifications as stated in the "Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors", 10 CFR 50, published in the Federal Register on July 22, 1993.

References

1. "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor", Prepared by General Electric, NEDO-31400, May 1987.

2. NRC letter from Ashok C. Thadani, Division of Systems Technology, to George J. Beck, BWR Owners' Group, c/o Philadelphia Electric Company, May 15, 1991, Subject - Acceptance for Referencing of Licensing Topical Report NEDO-31400, "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor".

3. U.S. NRC Standard Review Plan, NUREG-0800, 15.4.9 Radiological Consequences of Control Rod Drop Accident (BWR) Appendix A, Rev. 2 - July 1981.

4. "Vapor-Water Partition Coefficient of Iodine and Organic Iodides" Yokinori Nishizawa et. al.: Mitsubishi Atomic Power Ind., Inc. : Nippon Gensheryoku Gakkaishi, 11 (205-10) (1969); ORNL-TR-2255.

TABLE B1

SUMMARY OF PROJECTED EXCLUSION AREA DOSES
FOLLOWING A POSTULATED CRDA DBA
DOSES IN REM

ANALYSIS BASIS	THYROID	WHOLE BODY
CP&L original CRDA DBA Results[a]	2.1×10^{-4}	3.06×10^{-4}
SRP Generic Values (with MSIV isolation)	75	6
GE Generic Values (with MSIV isolation)	4.3	0.31
Brunswick Doses Based on GE Model but using Brunswick's X/Q[b] (with MSIV isolation) NEDO-31400 Fig. 2	1.4	0.1
GE Generic Values[c] (without MSIV isolation)	0	0.55
Brunswick Doses Based on GE Model using Brunswick's X/Q[d] and Brunswick's Charcoal Delay Times[e] (without MSIV isolation) NEDO-31400 Fig. 3 & 4	0	0.045
Brunswick Doses Based on MVP operation[f] (< 5% power)	18.69	2.17

NOTES:

a) Adjusted (in the UFSAR) to reflect 8x8 fuel
b) 8.4×10^{-4} Sec/m^3 (UFSAR Table 15.7.1-1)*
c) 18 hours delay for Krypton and 15 days for Xenon
d) 2×10^{-6} sec/m^3 (UFSAR Table 15.7.1-1)*
e) 16.8 hours delay for Krypton and 16 days for Xenon (System Des. SD-33)
f) NUS Calculation 8T12-M-01

*UFSAR values are significantly greater than the values used in the Offsite Dose Calculation Manual and are, therefore, conservative.

TABLE B2

COMPARISON OF UFSAR AND NEDO VALUES FOR BNP U1 & U2

PARAMETER	NEDO-31400	UFSAR
Power level	0.12 MWt/rod	0.114 MWt/rod[b]
No. of failed fuel rods	850	330 (7X7 fuel) 850 (8X8 fuel)
Decay Times		
Krypton	8 Hours[b]	16.8 Hours[b]
Xenon	8 Hours[c]	16 Days[c]
Chi/Q for release at ground level, s/m^3	2.5E-3	8.4E-4[d]
Chi/Q for stack release, s/m^3	3.0E-4	2.0E-5[e]

Notes:

a) ((2436 MWt)(1.05)(1.5)) / ((560 fuel bundles)(60 rods/bundle))

b) Minimum values required to meet the SRP generic values assuming a Chi/Q of 3×10^{-4}. (Refer to section 6.3 of NEDO 31400)

c) System Description SD-30

d) UFSAR Table 15.7.1-1 (ground level)

e) UFSAR Table 15.7.1.1 (stack release)

ATTACHMENT B1

NUS CALCULATION 8T12-M-01

ENGINEERING CALCULATION

CLIENT/PROJECT CP&L - BSEP CALC. NO. 8T12-M-01 REV. 1

TITLE Manual Isolation of the Condenser Mechanical Vacuum Pump upon a High MSLRM Signal

AUTHOR/DATE	VERIFIED BY/DATE	APPROVED BY/DATE
John W. Johnson 12/17/93	Stephen Talley 12/17/93	[signature] 12-17-93

PURPOSE:

To determine if the 0-2 hour site boundary dose would be below SRP 15.4.9 guideline values, assuming a control rod drop accident (CRDA) while the condenser is being evacuated, using the mechanical vacuum pumps (MVPs), and these pumps are manually isolated on a high radiation signal.

RESULTS The SRP 15.4.9 guideline values for the 0-2 hour dose at the site boundary, following a postulated CRDA and the corresponding doses for the BSEP (assuming the operator manually trips the MVPs ten minutes following the postulated CRDA) are as follows:

	Thyroid **	Whole body *
SRP 15.4.9	75 REM	6 REM
BSEP Calculated values	18.7 REM	2.8 REM

* Shorter isolation times would result in a lower dose, with the net effect being between a linear and exponential relationship between dose and time because of the short half lives of the dose controlling nuclides.

** The NEDO evaluation stated that the Iodine source term at low power when the MVPs were in use would not be significant. This is the calculated dose using full NEDO source term.

SUPERCEDED BY	QUALITY CLASS	DISTRIBUTION	VERIFICATION METHOD
REV. ______	■ SAFETY RELATED	■ PROJECT ☐ DCC	■ REVIEW
SUPPLEMENTED BY CALC. NO. ______	☐ NON-SR	☐ OTHER ______	☐ ALT. ANALYSIS
	☐ OTHER ______		

Halliburton NUS Corporation

STANDARD CALCULATION SHEET

CLIENT:	FILE NO.:	BY:	
CP&L - BSEP	8T12-M-01	R. A. MARTINEIT	PAGE 2 OF 27
SUBJECT: CRDA - MVP MANUAL ISOL.		CHECKED BY: [signature]	DATE: 12/5/93

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R

PURPOSE:

FOR SEVERAL REASONS DETAILED IN NEDO-31400 (REF 1) THE BWR OWNER'S GROUP REQUESTED PERMISSION OF THE NRC TO ELIMINATE THE MAIN STEAM VALVE CLOSURE FUNCTION AND SCRAM FUNCTION OF THE MAIN STEAM LINE RADIATION MONITOR.

THE NRC HAS APPROVED THIS REQUEST VIA REF. 2. CP&L IS TAKING ADVANTAGE OF THIS OPPORTUNITY TO REMOVE THE SUBJECT SCRAM AND ISOLATION FUNCTION FROM THE MSLRM.

HOWEVER, "NOTE d" IN TABLE 4.3.2-1 TO TECHNICAL SPECIFICATION (TS) 3/4.3.2 REQUIRES THAT TESTING BE PERFORMED TO VERIFY THAT THE MVP TRIPS AND THE MVP DISCHARGE LINE

VALVE CLOSES ON A HIGH RADIATION SIGNAL FROM THE MSLRM. IT HAS BEEN REQUESTED THAT AN ASSESSMENT BE PERFORMED TO DETERMINE IF THIS AUTOMATIC TRIP AND VALVE CLOSURE CAN BE ELIMINATED.

RATHER THAN AUTOMATIC PUMP TRIP AND VALVE CLOSURE BEING INITIATED VIA THE MSLRM THEY COULD BE INITIATED BY EITHER

1) MANUAL INITIATION IN 10 MINUTES OR LESS FOLLOWING A MAIN STEAM LINE HIGH RADIATION SIGNAL IN THE CONTROL ROOM.

2) AUTOMATIC INITIATION BY THE MAIN

STACK MONITOR ON A HIGH RADIATION SIGNAL IN ABOUT 4 MINUTES. IN THAT THIS MONITOR MAY BE TEMPORARILY OUT OF SERVICE FOR MAINTENANCE OR OTHER REASONS, THE PREFERRED APPROACH IS TO BASE THE ASSESSMENT ON MANUAL INITIATION SINCE THIS WOULD PROVIDE CP&L - BSEP THE GREATEST FLEXIBILITY IN ADDRESSING THIS ISSUE.

THE BASIS FOR DEMONSTRATING THE ACCEPTABILITY OF THIS APPROACH IS COMPLIANCE WITH THE DOSE GUIDELINES OF SRP 15.4.9 WHICH FOR A CRDA ARE 75 REM THYROID AND 6 REM WHOLE BODY FOR THE 0-2 HR DOSE AT THE SITE BOUNDARY. R1

ASSUMPTIONS:

1- ALL THE NOBLE GAS ACTIVITY IMMEDIATELY REACHES THE CONDENSER.

2- EVEN THOUGH THE REACTOR HAS BEEN SHUTDOWN LONG ENOUGH TO LOSE CONDENSER VACUUM, THE FULL POWER, END OF LIFE, NOBLE GAS SOURCE TERM GIVEN IN TABLE 1 OF REF. 1 WILL BE USED. THIS IS VERY CONSERVATIVE SINCE A MAJORITY OF THE OFF-SITE DOSE WILL BE RECEIVED FROM SHORT LIVED NOBLE GASES SUCH AS Kr-87 (76 MIN), Kr-88 (2.8 HOURS), AND XE-138 (14 MIN).

3- IODINE RELEASES ARE NOT A POTENTIAL CONCERN BASED ON THE FOLLOWING STATEMENT GIVEN IN SUBSECTION 6.3.2.1 OF REF 1.

"IF THE EVENT OCCURS AT LOW POWER WITHOUT THE SJAE OPERATING, THE ADDITIONAL IODINE ACTIVITY DUE TO CARRYOVER WOULD NOT BE EXPECTED TO BE SIGNIFICANT."

HOWEVER: THE SITE BOUNDARY THYROID DOSE WILL ALSO BE CALCULATED USING THE REF. 1 TABLE 1 IODINE SOURCE TERM. (THIS IS THE MAXIMUM SOURCE TERM)

4- CONDENSER LEAKAGE = 1% AS USED IN REF. 1

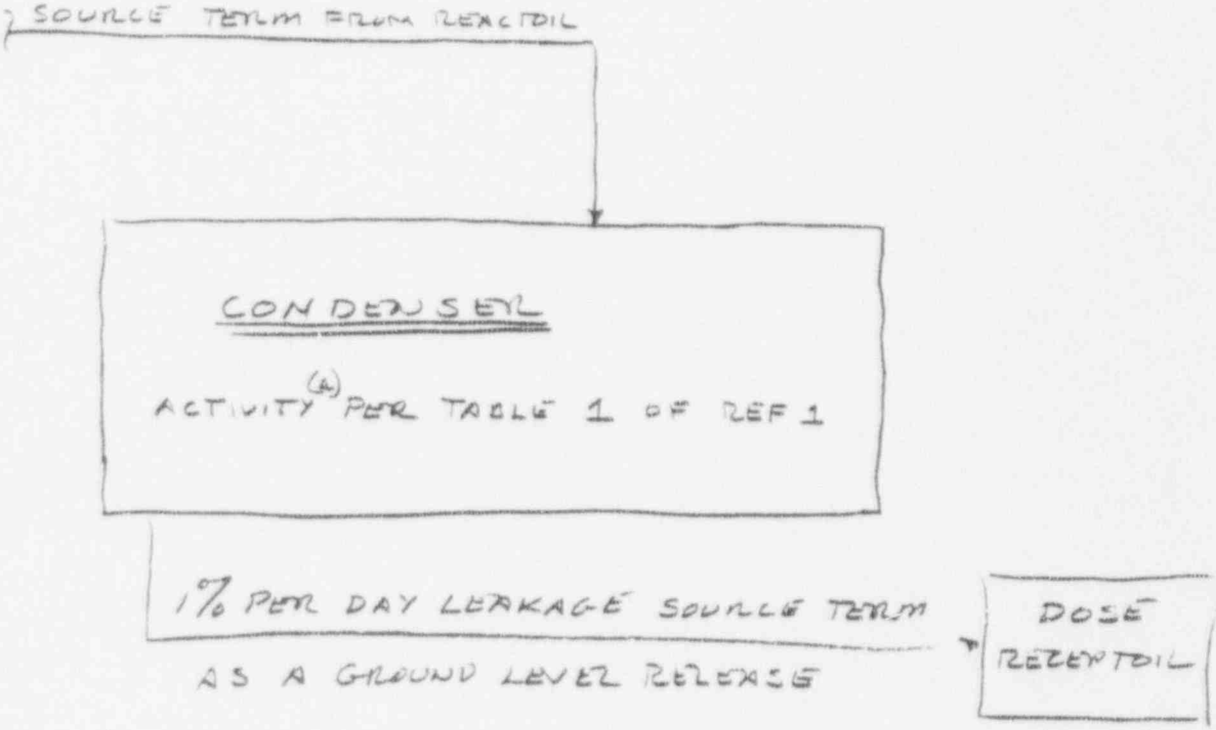
5- THE CONDENSER FREE VOLUME IS 86,700 FT^3 BASED ON ATTACHMENT A. R1

6- BOTH MECHANICAL VACUUM PUMPS ARE OPERATING (1810 CFM / MVP - REF 3)

7- OTHER ASSUMPTIONS AS DETAILED ON PAGES 10, 11, AND 12

TECHNICAL APPROACH:

APPROACH AS USED IN REF 1 (GE NEDO REPORT)



THIS NEDO EVALUATION ASSUMED THE CONDENSER STARTED LEAKING AT T=0

APPROACH USED IN THIS EVALUATION

AS STATED IN THE DISCUSSION UNDER THE "PURPOSE" SECTION OF THIS CALCULATION, THE OBJECTIVE IS TO DEMONSTRATE THAT OPERATOR ACTION TO ISOLATE THE CONDENSER 10 MINUTES FOLLOWING A POSTULATED CRDA WILL NOT RESULT IN DOSES IN EXCESS OF THE GUIDELINE VALUES OF SRP 15.4.9. THEREFORE, THIS CALCULATION HAS TWO SEGMENTS. THE FIRST IS FOR ZERO TO 10 MINUTES. DURING THIS PHASE OF THE CALCULATION, BOTH MVP'S ARE ASSUMED TO BE OPERATING. THE EXHAUST OF THE MVPs ARE ROUTED TO THE STATION'S MAIN STACK WHICH RESULTS IN AN ELEVATED RELEASE OF THE RADIOACTIVE GASES.

THE TOTAL DISCHARGE OVER 10 MINUTES WILL BE CALCULATED AS FOLLOWS:

λ_d ≡ DECAY CONSTANT = $0.693/T_{1/2}$

λ_p ≡ CONDENSER PURGE CONSTANT = $\frac{\text{PURGE RATE}}{\text{CONDENSER VOL.}}$

$\lambda_\tau \equiv \lambda_d + \lambda_p$

$T_{1/2}$ ≡ RADIONUCLIDE HALF LIFE (T^{-1})

CST ≡ CONDENSER SOURCE TERM

CST_d ≡ AFTER 10 MINUTES DECAY ≡ $(CST)(e^{-\lambda_d t})$

CST_τ ≡ AFTER 10 MINUTES DECAY AND PURGE ≡ $(CST)(e^{-\lambda_\tau t})$

THE STACK RELEASE = $CST_d - CST_\tau$

<u>FOR $T_{1/2}$ > 100 MINUTES IT IS ASSUMED</u> THAT λ_d IS NEGLIGIBLE IN COMPARISON TO λ_p AND THE STACK RELEASE IS CALCULATED AS FOLLOWS:

STACK RELEASE = $CST(1 - e^{-\lambda_p t})$

THE CST PROVIDED IN TABLE 1 OF REF 1 STARTS AT T = 1 MINUTE. FOR CALCULATING THE STACK RELEASE THE CST FOR T=0 WILL BE USED. THIS WILL BE CALCULATED AS FOLLOWS:

$$CST_{=0} = CST_{1\,MIN} / e^{-\lambda t}$$

$$\lambda = \lambda_d + \lambda_L^*$$

* = THE NEDO CALCULATION ASSUMES THE LEAKAGE STARTS AT T=0

λ_L = LEAKAGE REMOVAL CONSTANT FOR 1%/day

$$= \frac{0.01}{day} = \frac{0.01}{day} \times \frac{day}{24\,hrs} \times \frac{hr}{60\,min} = \frac{6.94 \times 10^{-6}}{min}$$

DURING THE 10 MINUTE PERIOD THAT THE MVP'S ARE ASSUMED TO REMAIN OPERATING, THERE IS NO ASSUMED LEAKAGE FROM THE CONDENSER. CONDENSER OUT LEAKAGE WHILE UNDER A VACUUM IS NOT A REASONABLE ASSUMPTION.

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IT IS ASSUMED THAT UPON SECURING THE MVPs, THE CONDENSER VACUUM IS TOTALLY LOST AND OUT LEAKAGE STARTS INSTANTLY.

THE CONDENSER OUT LEAKAGE FROM T = 10 MINUTES TO 2 HOURS WILL BE CALCULATED AS FOLLOWS:

CL ≡ CONDENSER LEAKAGE AS GIVEN IN TABLE 2 OF REF 1

CL_{10} ≡ TOTAL CONDENSER LEAKAGE (RELEASE) FOR T=0 TO T = 10 MINUTES

CL_{2hrs} = TOTAL CONDENSER LEAKAGE (RELEASE) FOR T=0 TO T = 2 HOURS

CL_{AST} ≡ CONDENSER LEAKAGE ACCIDENT SOURCE TERM (GROUND LEVEL RELEASE)

$$CL_{AST} = \left[\frac{CST_I}{CST_{10\,REF}}\right]\left[CL_{2Hrs} - CL_{10}\right]$$

$CST_{10\,REF}$ = CONDENSER SOURCE TERM AT T=10 MINUTES AS GIVEN IN TABLE-1 OF REFERENCE 1.

THE WHOLE BODY DOSE (REM) IS CALCULATED AS FOLLOWS:

$$D_{WB} = 0.25\,(X/Q) \cdot R_i \cdot \overline{E}_\gamma \quad \text{(Equ. 43 of Ref. 4)}$$

D_{WB} ≡ WHOLE BODY DOSE

X/Q ≡ ATMOSPHERIC DISPERSION FACTOR

R_i ≡ TOTAL RELEASE IN CURIES OF i^{th} NUCLIDE

$\overline{E}_\gamma$ ≡ AVERAGE GAMMA ENERGY OF i^{th} NUCLIDE

THE THYROID DOSE (REM) IS CALCULATED AS FOLLOWS:

$$D_T = BR \cdot (X/Q) \cdot I_i \cdot DCF_i$$

D_T ≡ THYROID DOSE

DCF_i ≡ DOSE CONVERSION FACTOR FOR THE i^{th} RADIOIODINE

I_i ≡ TOTAL RELEASE IN CURIES OF THE i^{th} RADIOIODINE

$BR = 3.47 \times 10^{-4}\ m^3/sec$ (0-8 Hrs) REF.-4

SUMMARY OF EQUATIONS

- STACK RELEASE (SR) FOR T=0 TO T=10 MINUTES

$$SR = CST_d - CST_\tau \quad \text{(PAGE 10)}$$

SR FOR NUCLIDES WITH $T_{1/2} > 100$ MINUTES

$$SR = CST\left(1 - e^{-\lambda_d \tau}\right) \quad \text{(PAGE 10)}$$

- CST AT T=0

$$CST = CST_{2\,min} \,/\, e^{-(\lambda_d + \lambda_L)\tau} \quad \text{(PAGE 11)}$$

- CONDENSER LEAKAGE ACCIDENT SOURCE TERM T > 10 MINUTES

$$CL_{AST} = \left[\frac{CST_\tau}{CST_{10\,REF}}\right]\left[CL_{2\,HRS} - CL_{10\,REF}\right] \quad \text{(PAGE 12)}$$

($CL_{2\,HRS} - CL_{10\,REF}$) → FROM TABLE 2 OF REF 1 GROUND LEVEL RELEASE

- DOSE EQUATIONS

$$D_{WB} = (0.25)(\chi/Q) \cdot R_i \cdot \bar{E}_\gamma$$

$$D_T = BR \cdot \chi/Q \cdot I_i \cdot DCF_i$$

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CALCULATIONAL INPUTS:

CONDENSER AIRBORNE ACTIVITY - CURIES (TABLE 1 - REF 1)

NUCLIDE	1 - MIN	10 - MIN	2 - HOURS
I-131	2.77×10^3	2.76×10^3	2.74×10^3
I-132	4.02×10^3	3.85×10^3	2.21×10^3
I-133	5.78×10^3	5.75×10^3	5.41×10^3
I-134	6.29×10^3	5.58×10^3	1.31×10^3
I-135	5.45×10^3	5.37×10^3	4.43×10^3
Kr-83m	3.40×10^4	3.21×10^4	1.60×10^4
Kr-85m	7.32×10^4	7.16×10^4	5.38×10^4
Kr-85	3.29×10^3	3.29×10^3	3.28×10^3
Kr-87	1.40×10^5	1.29×10^5	4.73×10^4
Kr-88	1.99×10^5	1.92×10^5	1.22×10^5
Kr-89	1.99×10^5	2.79×10^4	1.00×10^{-6}
Xe-131m	1.72×10^3	1.72×10^3	1.72×10^3
Xe-133m	2.51×10^4	2.51×10^4	2.45×10^4
Xe-133	6.03×10^5	6.02×10^5	5.90×10^5
Xe-135m	1.09×10^5	7.30×10^4	5.58×10^2
Xe-135	7.78×10^4	7.09×10^4	6.69×10^4
Xe-137	4.41×10^5	8.66×10^4	1.58×10^{-4}
Xe-138	4.79×10^5	3.08×10^5	1.42×10^3

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RELEASES FROM THE CONDENSER - CURIES
GROUND LEVEL RELEASES - TABLE 2 REF 1

NUCLIDE	10 MIN	2 HOURS
I-131	1.92×10^{-1}	2.30×10^{0}
I-132	2.74×10^{-1}	2.53×10^{0}
I-133	4.01×10^{-1}	4.60×10^{0}
I-134	4.14×10^{-1}	2.67×10^{0}
I-135	3.76×10^{-1}	4.11×10^{0}
Kr-83m	2.30×10^{0}	2.00×10^{1}
Kr-85m	5.03×10^{0}	5.26×10^{1}
Kr-85	2.28×10^{-1}	2.74×10^{0}
Kr-87	9.35×10^{0}	7.15×10^{1}
Kr-88	1.30×10^{1}	1.32×10^{2}
Kr-89	7.00×10^{0}	7.88×10^{0}
Xe-131m	1.20×10^{-1}	1.43×10^{0}
Xe-133m	1.74×10^{0}	2.07×10^{1}
Xe-133	4.18×10^{1}	4.99×10^{2}
Xe-135m	6.27×10^{0}	1.77×10^{1}
Xe-135	5.38×10^{0}	6.02×10^{1}
Xe-137	1.70×10^{1}	2.03×10^{1}
Xe-138	2.76×10^{1}	7.11×10^{1}

MAIN STACK χ/Q TO SITE BOUNDARY = 2×10^{-5} SEC/m^3
GROUND LEVEL χ/Q TO SITE BOUNDARY = 8.4×10^{-4} SEC/m^3

VALUES FROM UFSAR TABLE 15.7.1-1

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NUCLIDE	T½ (min)	λ_d (MIN^{-1})	$\bar{E}_\gamma$ [a]	DCF [a]
I-131	1.159×10^4	5.98×10^{-5}	0.371	1.48×10^6
I-132	138	5.02×10^{-3}	2.400	5.35×10^4
I-133	1260	5.5×10^{-4}	0.477	4.00×10^5
I-134	52	1.33×10^{-2}	1.939	$2.3? \times 10^4$
I-135	402	1.72×10^{-3}	1.779	1.24×10^5
Kr-83m	111.6	6.21×10^{-3}	0.005	
Kr-85m	264	2.63×10^{-3}	0.156	
Kr-85	5.655×10^6	1.23×10^{-7}	0.0021	
Kr-87	76	9.12×10^{-3}	1.375	
Kr-88	168	4.13×10^{-3}	1.743	
Kr-89	3.18	2.18×10^{-1}	2.219 [b]	
Xe-131m	1.699×10^4	4.08×10^{-5}	0.022	
Xe-133m	3254	2.13×10^{-4}	0.033	
Xe-133	7589	9.13×10^{-5}	0.030	
Xe-135m	15.6	4.44×10^{-2}	0.422	
Xe-135	548	1.26×10^{-3}	0.246	
Xe-137	3.8	1.82×10^{-1}	0.192 [b]	
Xe-138	14	4.95×10^{-2}	2.870	

λ_p (CONDENSER PURGE) = (2)(1870 CFM)/86,700 CF
= 4.18×10^{-2}/min | R1

λ_L (LEAKAGE) 6.94×10^{-6}/min (PAGE 11)

a) REF. 4 b) REF 5

CALCULATIONS:

NUCLIDE	CONC AT $T=0^{(a)}$	$e^{-\lambda\tau}$ (b)	$1-e^{-\lambda\tau}$	STATIC (c) RELEASE
I-131	2.77×10^{3}	0.659	0.341	9.46×10^{2}
I-132	4.04×10^{3}	"	"	1.38×10^{3}
I-133	5.78×10^{3}	"	"	1.97×10^{3}
I-134	6.37×10^{3}	—	—	(d)
I-135	5.45×10^{3}	0.659	0.341	1.86×10^{3}
Kr-83m	3.42×10^{4}	"	"	1.17×10^{4} R1
Kr-85m	7.34×10^{4}	"	"	2.51×10^{4}
Kr-85	3.29×10^{3}	"	"	1.12×10^{3}
Kr-87	1.41×10^{5}	—	—	(d)
Kr-88	2.00×10^{5}	0.659	0.341	6.82×10^{4}
Kr-89	2.47×10^{5}	—	—	(d)
Xe-131m	1.72×10^{3}	0.659	0.341	5.87×10^{2}
Xe-133m	2.51×10^{4}	"	"	8.57×10^{3}
Xe-133	6.03×10^{5}	"	"	2.06×10^{5}
Xe-135m	1.14×10^{5}	—	—	(d)
Xe-135	7.79×10^{4}	0.659	0.341	2.66×10^{4}
Xe-137	5.29×10^{5}	—	—	(d)
Xe-138	5.03×10^{5}	—	—	(d)

a) BASES GIVEN ON PAGE 19
b) $\lambda = \lambda_p$ ON PAGE 17 $\tau = 10$ min
c) (CONC AT $T=0$)$(1-e^{-\lambda\tau})$
d) $T_{1/2}$ < 100 MINUTES - CALCULATED ON PAGE 19

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CONDENSER AIRBORNE ACTIVITY - CURIES

NUCLIDE		T = 1 min	T = 0 (a)	CST_d (b)	CST_T (c)	$CST_d - CST_T$
I-131		2.77×10^3	2.77×10^3			THIS IS THE STACK RELEASE FOR THESE RADIONUCLIDES
I-132		4.02×10^3	4.04×10^3			
I-133		5.78×10^3	5.78×10^3			
I-134	•	6.29×10^3	6.37×10^3	5.58×10^3	3.68×10^3	1.90×10^3
I-135		5.45×10^3	5.45×10^3			
Kr-83m		3.40×10^4	3.42×10^4			
Kr-85m		7.32×10^4	7.34×10^4			
Kr-85		3.29×10^3	3.29×10^3			
Kr-87	•	1.40×10^5	1.41×10^5	1.27×10^5	8.49×10^4	4.40×10^4
Kr-88		1.99×10^5	2.00×10^5			
Kr-89	•	1.99×10^5	2.47×10^5	2.79×10^4	1.84×10^4	9.55×10^3
Xe-131m		1.72×10^3	1.72×10^3			
Xe-133m		2.51×10^4	2.51×10^4			
Xe-133		6.03×10^5	6.03×10^5			
Xe-135m	•	1.09×10^5	1.14×10^5	7.31×10^4	4.81×10^4	2.49×10^4
Xe-135		7.78×10^4	7.79×10^4			
Xe-137	•	4.41×10^5	5.29×10^5	8.57×10^4	5.65×10^4	2.93×10^4
Xe-138	•	4.79×10^5	5.03×10^5	3.07×10^5	2.02×10^5	1.05×10^5

a) (ACTIVITY AT T = 1 MIN / $e^{-\lambda t}$) $t = 1$ min, $\lambda = \lambda_d + \lambda_L$ GIVEN ON PAGE 17

b) (CST AT T = 0)($e^{-\lambda_d t}$) λ_d GIVEN ON PAGE 17, $t = 10$ min

c) (CST AT T=0)($e^{-\lambda_T t}$) $\lambda_T = \lambda_d + \lambda_p$ VALUES GIVEN ON PAGE 17

(b and c: ONLY FOR T1/2 LESS THAN 100 MINUTES)

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(1) NUCLIDE	(2) CONDENSOR CONC AT T = 0	(3) CONDENSOR CONC AT T = 10 MIN	(4) REF 1 CONC AT T = 10 MIN	(5) CONC. AT 10 MIN ÷ REF 1 CONC AT 10 MIN	(6) REF 1 RELEASE AT 2 HRS — REF 1 REL. AT 10 MIN (c)	(7) GROUND LEVEL RELEASE COL 5 × COL 6
I-131	2.77×10^3	1.82×10^3 (a)	2.76×10^3	0.661	2.11	1.39
I-132	4.04×10^3	2.66×10^3 (a)	3.85×10^3	0.691	2.26	1.56
I-133	5.78×10^3	3.81×10^3 (a)	5.75×10^3	0.662	4.2	2.78
I-134	6.37×10^3	3.68×10^3 (b)	5.58×10^3	0.659	2.26	1.49
I-135	5.45×10^3	3.60×10^3 (a)	5.37×10^3	0.670	3.73	2.50
Kr-83m	3.42×10^4	2.25×10^4 (a)	3.21×10^4	0.702	17.7	12.43
Kr-85m	7.34×10^4	4.83×10^4 (a)	7.16×10^4	0.675	47.6	32.12
Kr-85	3.29×10^3	2.17×10^3 (a)	3.29×10^3	0.659	2.51	1.65
Kr-87	1.41×10^5	8.44×10^4 (b)	1.29×10^5	0.659	62.2	40.93
Kr-88	2.00×10^5	1.32×10^5 (a)	1.92×10^5	0.68[illegible]	119.0	81.58
Kr-89	2.47×10^5	1.84×10^4 (b)	2.79×10^4	0.660	0.88	0.58
Xe-131m	1.72×10^3	1.13×10^3 (a)	1.72×10^3	0.659	1.31	0.86
Xe-133m	2.51×10^4	1.65×10^4 (a)	2.51×10^4	0.659	19.0	12.49
Xe-133	6.03×10^5	3.97×10^5 (a)	6.02×10^5	0.660	457	301.67
Xe-135m	1.14×10^5	4.81×10^4 (b)	7.3×10^4	0.660	11.4	7.54
Xe-135	7.79×10^4	5.13×10^4 (a)	7.09×10^4	0.724	54.8	39.67
Xe-137	5.29×10^5	5.65×10^4 (b)	8.66×10^4	0.652	3.30	2.15
Xe-138	5.03×10^5	2.02×10^5 (b)	3.08×10^5	0.656	43.5	28.54

a) (CONC AT T = 0)($e^{-\lambda t}$) PAGE 18; b) FROM PAGE 19; c) PAGE 16

GROUND LEVEL RELEASES IN CURIES

R1

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SUMMARY OF THE
SITE BOUNDARY SOURCE TERMS

SOURCE TERMS IN CURIES

NUCLIDE	STACK RELEASES	GROUND LEVEL RELEASES
I-131	1.46×10^{2}	1.39
I-132	1.38×10^{2}	1.56
I-133	1.47×10^{3}	2.78
I-134	1.90×10^{2}	1.49
I-135	1.86×10^{3}	2.50
Kr-83m	1.17×10^{4}	12.43
Kr-85m	2.51×10^{4}	32.12
Kr-85	1.12×10^{3}	1.65
Kr-87	4.40×10^{4}	40.93
Kr-88	6.82×10^{4}	81.58
Kr-89	9.55×10^{3}	0.58
Xe-131m	5.87×10^{2}	0.86
Xe-133m	8.57×10^{3}	12.49
Xe-133	2.06×10^{5}	301.67
Xe-135m	2.49×10^{4}	7.54
Xe-135	2.66×10^{4}	39.67
Xe-137	2.93×10^{4}	2.15
Xe-138	1.05×10^{5}	28.54

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WHOLE BODY DOSE - SITE BOUNDARY - STACK RELEASES

NUCLIDE	0.25	× X/Q	× CURIES	× $\overline{E}_\gamma$	=	REM
I-131	0.25	2×10^{-5}	4.46×10^{2}	0.371	=	0.0018
I-132	"	"	1.38×10^{3}	2.4	=	0.0165
I-133	"	"	1.97×10^{3}	0.477	=	0.0047
I-134	"	"	1.90×10^{3}	1.939	=	0.0185
I-135	"	"	1.86×10^{3}	1.779	=	0.0166
Kr-83m	"	"	1.17×10^{4}	0.005	=	0.0003
Kr-85m	"	"	2.51×10^{4}	0.156	=	0.0195
Kr-85	"	"	1.12×10^{3}	0.0021	=	NEG
Kr-87	"	"	4.40×10^{4}	1.375	=	0.303
Kr-88	"	"	6.82×10^{4}	1.743	=	0.594
Kr-89	"	"	9.55×10^{3}	2.219	=	0.106
Xe-131m	"	"	5.87×10^{2}	0.022	=	NEG
Xe-133m	"	"	8.57×10^{3}	0.033	=	0.0014
Xe-133	"	"	2.06×10^{5}	0.03	=	0.0309
Xe-135m	"	"	2.49×10^{4}	0.422	=	0.0526
Xe-135	"	"	2.66×10^{4}	0.246	=	0.0327
Xe-137	"	"	2.93×10^{4}	0.192	=	0.0281
Xe-138	"	"	1.05×10^{5}	2.87	=	1.50

SUBTOTAL = 2.7244 REM

R1 R1

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WHOLE BODY DOSE - SITE BOUNDARY - GROUND LEVEL RELEASES

NUCLIDE	0.25 ×	X/Q ×	CURIES ×	$\bar{E}_\gamma$	= REM
I-131	0.25	8.4×10^{-4}	1.39	0.371	0.0001
I-132	"	"	1.56	2.4	0.0008
I-133	"	"	2.78	0.477	0.0003
I-134	"	"	1.49	1.939	0.0006
I-135	"	"	2.50	1.779	0.0009
Kr-83m	"	"	12.43	0.005	NEG
Kr-85m	"	"	32.12	0.156	0.0011
Kr-85	"	"	1.65	0.0021	NEG
Kr-87	"	"	40.93	1.375	0.0118
Kr-88	"	"	81.58	1.743	0.0299
Kr-89	"	"	0.58	2.219	0.0003
Xe-131m	"	"	0.86	0.022	NEG
Xe-133m	"	"	12.49	0.033	0.0001
Xe-133	"	"	301.67	0.03	0.0019
Xe-135m	"	"	7.54	0.422	0.0007
Xe-135	"	"	39.67	0.246	0.0021
Xe-137	"	"	2.15	0.192	0.0001
Xe-138	"	"	28.54	2.87	0.0172

SUBTOTAL 0.0677 REM

STACK RELEASES = 2.73 REM
GROUND LEVEL = 0.07 REM
2.80 REM

Halliburton NUS Corporation — STANDARD CALCULATION SHEET

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CP&L - BSEP	8T12-M-01	[signature]	24 OF 27
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R1

THYROID DOSE - SITE BOUNDARY - STACK RELEASE

NUCLIDE	BR	×	X/Q	×	CURIES	×	DCF	=	REM
I-131	3.47×10^{-4}		2×10^{-5}		9.46×10^{2}		1.48×10^{6}	=	9.71
I-132	"		"		1.38×10^{3}		5.35×10^{4}	=	0.51
I-133	"		"		1.97×10^{3}		4.00×10^{5}	=	5.48
I-134	"		"		1.90×10^{3}		2.50×10^{4}	=	0.33
I-135	"		"		1.86×10^{3}		1.24×10^{5}	=	1.60

SUBTOTAL = 17.64 REM

THYROID DOSE - SITE BOUNDARY - GROUND LEVEL RELEASES

R1

NUCLIDE	BR	×	X/Q	×	CURIES	×	DCF	=	REM
I-131	3.47×10^{-4}		8.4×10^{-4}		1.39		1.48×10^{6}	=	0.60
I-132	"		"		1.56		5.35×10^{4}	=	0.02
I-133	"		"		2.78		4.00×10^{5}	=	0.32
I-134	"		"		1.49		2.50×10^{4}	=	0.01
I-135	"		"		2.50		1.24×10^{5}	=	0.09

SUBTOTAL = 1.05 REM

TOTAL = 18.69 REM

R R

Halliburton NUS Corporation	STANDARD CALCULATION SHEET

CLIENT: CP&L - BSEP	FILE NO.: 8T12-M-01	BY: [signature]	PAGE 25 OF 22
SUBJECT: CRDA - MVP MANUAL ISOL.		CHECKED BY: [signature]	DATE: 12/5/93

SUMMARY:

THE TOTAL WHOLE BODY DOSE IS 2.8 REM. THIS IS 46.7% OF THE DOSE GUIDELINE VALUES OF SRP 15.4.9. THIS ASSUMES THE MAXIMUM END OF LIFE SOURCE TERM. OF THIS TOTAL DOSE, 54.2% IS FROM Xe-138 WITH A 14 MINUTE HALF LIFE. ALSO, 94.4% OF THE TOTAL DOSE IS FROM RADIONUCLIDES WITH A HALF LIFE OF ABOUT 3 HOURS OR LESS. CONSIDERING THAT THE PLANT IS IN A STARTUP MODE USING THE MVP, THEN IT IS REASONABLE TO ASSUME THAT DECAY WOULD HAVE SIGNIFICANTLY REDUCED THE SOURCE TERM. ALSO, THE PLANT WOULD BE AT 7% POWER OR LESS SINCE THE MVPS ARE IN USE. THEREFORE, HAD THE REALISTIC PARAMETERS BEEN USED FOR THIS SCENARIO, THE CALCULATED DOSES WOULD BE

CONSIDERABLY LOWER.

THE CALCULATED THYROID DOSE IS 24.9 % OF GUIDELINE VALUES. AGAIN, USING REALISTIC PARAMETERS WOULD MEANINGFULLY REDUCE THIS DOSE. R1

Halliburton NUS Corporation

STANDARD CALCULATION SHEET

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SUBJECT: CRDA - MVP MANUAL ISOL.		CHECKED BY: [signature]	DATE: 12/5/93

REFERENCES :

1 - SAFETY EVALUATION FOR ELIMINATING THE BOILING WATER REACTOR MAIN STEAM ISOLATION VALVE CLOSURE FUNCTION AND SCRAM FUNCTION OF THE MAIN STEAM LINE RADIATION MONITOR. NEDO-31400, MAY 1987 | R1

2 - NRC LETTER TO GEORGE J. BECK, CHAIRMAN BWR OWNER'S GROUP, FROM ASHOK C. THADANI, DIRECTOR DIVISION OF SYSTEMS TECHNOLOGY, DATED MAY 15, 1991

3 - BSEP/VOL II/SD-30 REV. 7 SECTION 3.2.2 MECHANICAL VACUUM PUMP | R1

4 - AXIDENT - A DIGITAL COMPUTER DOSE CALCULATION MODEL, NUS-1554, Rev. 3

5 - SUMMARY OF GAMMA AND BETA ENERGY AND INTENSITY DATA, MEEK AND GILBERT, NEDO-12037 69-NED-41 CLASS I, JANUARY 1970

Attachment A

Purpose:

The purpose of the Attachment A calculation is to determine the free air volume of both Low Pressure Turbine Condensers at Brunswick Nuclear Station, i.e., to determine the total condenser free air volume of an individual unit.

Description:

This calculation will include the volume of the condenser, the Steam Dome Interconnect, and the LP Turbine connection. The volumes of the Feed Water Heaters, the Normal Water Level, the Condenser tubes, and miscellaneous structure will be deducted from the total volume.

References:

FP-02450, Sh.1&2, Rev.A,	General Arrangement of Surface Condenser
FP-08504, Sh.1&2, Rev.C,	Cross Section Of Main Turbine
FP-02528, Rev.B	Condenser Vendor Manual
F-02004, Rev.10,	Turbine Building General Arrangement El. 38 & 45
F-02007, Rev.7,	Turbine Building General Arrangement Cross Section B-B
SD-28, Rev.6	Main Condenser

Assumptions:

1. The dimensions at the condenser surface used in the volume calculations will be reduced by a 1" for wall thickness.

2. The Condenser and associated volumes will be divided into parts for calculation of the internal volume.

3. For the dimensions of the condenser used in the calculation see 'Attachment A - Figure.'

Condenser Parts:

A. Lower Condenser (LC) - Condenser portion containing the condensing water tubes.

B. Steam Dome Section 1 (SD1) - Condenser portion up to the expansion joint containing the opening for the Steam Dome Interconnect.

C. Steam Dome Section 2 (SD2) - Condenser portion up to the expansion joint containing the openings for the Feed Water Heaters.

D. Steam Dome Section 3 (SD3) - Condenser portion up to the expansion joint containing the connection for the Feed Water Pump Turbine Exhaust Line.

E. Steam Dome Section 4 (SD4) - Condenser portion above and including the expansion joint.

F. Steam Dome Interconnect (SDI) - The passage connecting the two Condensers.

G. LP Turbine Connection (LPC) - The connection between the LP Turbine and the Condenser.

H. Feed Water Heaters (FWH) - The two Feed Water Heaters that are located within both Condensers (two heaters per condenser).

I. Normal Water Level (NWL) - The normal water level within the condenser.

J. Condenser Tubes Volume (CT) - The tubes located within both condensers.

Calculation:

A. Lower Condenser Volume

$V = \{[(23' - 4^7/_8") + (22' - 9^7/_8")] \times [(12' - 2^1/_2") + (8' - 3^1/_2")] \times (30' - 0")$

$V = (46.23') \times (20.50') \times (30.00') = 28{,}431\ ft^3$

B. Steam Dome Section 1 Volume

$V = (23.79') \times (18.75') \times (30.00') = 13{,}382\ ft^3$

C. Steam Dome Section 2 Volume

$V = (^1/_2) \times (11.22') \times (18.75') \times (30.00') = 3156\ ft^3$

D. Steam Dome Section 3 Volume

$V = (^1/_2) \times (11.22') \times (18.75') \times (30.00') = 3156\ ft^3$

E. Steam Dome Section 4 Volume

$V = (23.79') \times (5.75') \times (30.00') = 4104\ ft^3$

F. Steam Dome Interconnect Volume

The Steam Dome Interconnect length is 6.5 feet.

$V = (6.50') \times (10.83') \times (6.50') = 458\ ft^3$

G. LP Turbine Connection Volume

The distance between the top of the Condenser and Elevation 70'- 0" is 6'-6". The diameter of the largest part of the LP Turbine is 10 feet (see page 8). A five foot radius below the Turbine centerline of 74'- 6" is an elevation of 69'- 6". The top of the Condenser is at 63'- 6". The distance between the bottom of the Turbine and the top of the condenser is 6'- 0". For conservatism the distance used in the volume calculation will be 5'- 4" (5.33').

$V = (23.79') \times (5.33') \times (30.00') = 3804\ ft^3$

H. Feed Water Heater Volume (Two Heaters per Condenser)

Heater length is measured along the centerline of the heater from the surface of the Condenser through half way along the dish at the heater dish end. This length is 26'-8" = 26.67'.

The heater diameters are taken as 64" = 5'- 4" = 5.33' and 69" = 5'- 9" = 5.75'
The heater radii are 2'- 8" = 2.67' and 2'- 10.5" = 2.88'

$V = \pi\,(2.67')^2 \times (26.67') = 597\ ft^3$

$V = \pi\,(2.88')^2 \times (26.67') = 695\ ft^3$

$V_H = 597 + 695 = 1292\ ft^3$

I. Normal Water Level Volume

The normal water level is $36^5/_8$" from the bottom of the Condenser.

$V = (46.23') \times (3.05') \times (30.00') = 4230\ ft^3$

J. The Condenser Tubes Volume

There are 46,236 tubes total in both Condensers. These tubes are 48' long and 1" OD.

$V = \pi\,(^{0.5}/_{12}\ ft)^2 \times (48\ ft) \times (46{,}236) = 12{,}105\ ft^3$

Total Free Air Volume for the Two Condensers Together

V = (2xLC) + (2xSD1) + (2xSD2) + (2xSD3) + (2xSD4) + (SDI) + (2xLPC) - (2xFWH) - (2xNWL) - (CT)

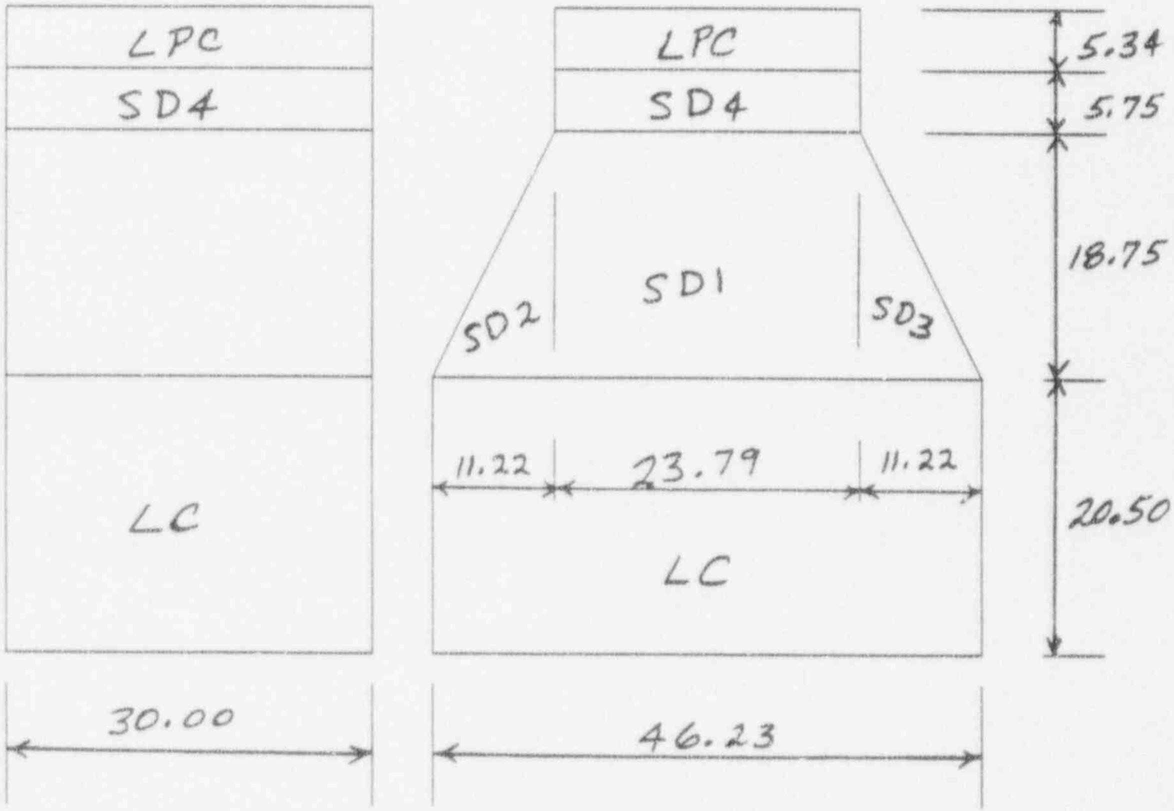
V = (2 x 28,431) + (2 x 13,382) + (2 x 3156) + (2 x 3156) + (2 x 4104) + (458) + (2 x 3804) - (2 x 1292) - (2 x 4230) - (12,105)

V = 89,375 ft^3

A 97% factor will be applied to the total volume to account for miscellaneous internal structure.

V = (0.97) x (89,375) ≈ 86,700 ft^3

ATTACHMENT A - FIGURE



Note: All dimensions are in feet.

NUS AND SUBSIDIARIES		TELECON NOTE
Control Number: 8T12	**Date:** 12/16/93	**Time:**
Distribution: File		
Between: Dave Canady	**Of:** BNP Tech Support	
	Phone: 910-457-2738	
And: Lamar Castles		**(NUS)**

Discussion: Lamar asked Dave to describe the volume below the LP Turbine which enters into the Main Condenser. This region is the connection between the LP Turbine (located on Elevation 70'- 0") and the top of the Main Condenser (located at Elevation 63'- 6"). Dave stated that the combined length of the largest rotor blade and shaft combination for the LP Turbine is approximately 10 feet. Dave stated that the LP Turbine blade would be below Elevation 70'- 0" given that the Turbine shaft centerline is at Elevation 74'- 6". Dave stated that the use of 10 feet as the diameter is acceptable.

Action Items:

ATTACHMENT B2

NUS CALCULATION 8T12-M-02

ENGINEERING CALCULATION

CLIENT/PROJECT CP&L - BSEP CALC. NO. 8T12-M-02 REV. 1

TITLE Control Room Doses Following Manual Isolation of the Condenser Mechanical Vacuum Pumps Following a Control Rod Drop Accident

AUTHOR/DATE	VERIFIED BY/DATE	APPROVED BY/DATE
John W. Johnson 12/17/93	[signature] 12/17/93	[signature] 12-17-93

PURPOSE:

NUS calculation 8T12-M-01 evaluated the 0-2 hour site boundary doses assuming a control rod drop accident (CRDA) while the condenser is being evacuated, using the mechanical vacuum pumps (MVPs), and these pumps are manually isolated on a high radiation signal. Calculation 8T12-M-01 demonstrated that the site boundary doses for this scenario were well within the doses guideline values of SRP 15.4.9.

This evaluation determines if the control room doses for the above scenario are within the dose limits of GDC-19.

RESULTS:

The post DBA control room dose limits established by GDC-19 and the calculated control room doses are as follows:

	Thyroid	Whole body
GDC-19	30 REM	5 REM
Calculated values	27.2 REM	.18 REM

SUPERCEDED BY REV. ____ SUPPLEMENTED BY CALC. NO. ____	QUALITY CLASS	DISTRIBUTION	VERIFICATION METHOD
	■ SAFETY RELATED	■ PROJECT ☐ DCC	■ REVIEW
	☐ NON-SR	☐ OTHER ____	☐ ALT. ANALYSIS
	☐ OTHER ____		

NUS 378 REVISED 0688

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CLIENT: CP&L - BSEP	FILE NO.: 8T12-M-02	BY: [signature]	PAGE 3 OF 25
SUBJECT: CONT. RM. DOSES - MVPs - MANUAL ISOL		CHECKED BY: [signature]	DATE: 12/9/93

PURPOSE:

NUS CALCULATION 8T12-M-01 (REF 3) CALCULATED THE SITE BOUNDARY DOSES FOR THE FOLLOWING ACCIDENT SCENARIO:

- A CONTROL ROD DROP ACCIDENT HAS OCCURRED
- THE ACCIDENT SOURCE TERM IS IMMEDIATELY TRANSFERRED TO THE CONDENSER
- THE MSLRM INITIATES A HIGH RADIATION ALARM SIGNAL
- BOTH THE MECHANICAL VACUUM PUMPS ARE IN OPERATION AND REMAIN IN OPERATION UNTIL MANUALLY TRIPPED (10 MINUTES)
- FOLLOWING MANUAL ISOLATION OF THE CONDENSER, THE CONDENSER LEAKS AT 1% PER DAY
- IN ADDITION, THIS CALCULATION USED THE MAXIMUM END OF CORE INVENTORY. HOWEVER, FOR THE MECHANICAL VACUUM PUMPS TO BE IN USE, THE NSSS WOULD HAVE TO HAVE BEEN SHUTDOWN LONG ENOUGH FOR THE CONDENSER TO LOSE VACUUM, AND IN ORDER FOR THE MECHANICAL VACUUM PUMPS TO BE IN OPERATION THE NSSS WOULD BE LIMITED TO 7% POWER. THIS MEANS THAT RADIOACTIVE DECAY WOULD HAVE REDUCED THE INVENTORIES OF

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SUBJECT: CONT. RM. DOSES - MVPs - MANUAL ISOL.		CHECKED BY: [signature]	DATE: 12/12/93

SHORT LIVED RADIONUCLIDES. (THE SOURCE TERM IS TAKEN FROM REF. 1.)

THE PURPOSE OF THIS CALCULATION IS TO QUANTIFY THE RADIOLOGICAL CONSEQUENCES FOR THE ABOVE ACCIDENT SCENARIO IN THE CONTROL ROOM AND TO DETERMINE IF THESE DOSES ARE IN COMPLIANCE WITH 10 CFR 50, APPENDIX A, GDC-19.

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SUBJECT: CONT. RM DOSES - MVPs - MANUAL IOL		CHECKED BY: [signature]	DATE: 12/12/93

ASSUMPTIONS:

THE ASSUMPTIONS AFFECTING THIS CALCULATION ARE AS DEFINED IN REFERENCES 2 & 3. THESE ASSUMPTIONS HAVE ALREADY BEEN REVIEWED DURING THE INDEPENDENT VERIFICATION OF THOSE REFERENCES.

ONE NEW ASSUMPTION OF THIS CALCULATION IS THAT THE GROUND LEVEL RELEASE OF IODINES IS A NEGLIGABLE CONTRIBUTOR TO THE CONTROL RM DOSE. THE BASIS FOR THIS ASSUMPTION IS THE RELATIVELY SMALL GROUND LEVEL RELEASE.

THE STACK & GROUND LEVEL SOURCE TERMS ARE AS FOLLOWS:

IODINE RELEASE IN CURIES

NUCLIDE	STACK	GROUND LEVEL	
I-131	946	1.39	RELEASE FOR T=10 MIN TO T=30 DAYS
I-132	1380	1.56	
I-133	1970	2.78	
I-134	1900	1.49	
I-135	1860	2.50	
	(10 MIN RELEASE)		

THE SAME ASSUMPTION IS BEING USED WITH RESPECT TO THE RADIOACTIVE NOBLE GAS SOURCE TERM.

OTHER ASSUMPTIONS ARE GIVEN IN THE NOTES ON PAGE 15.

TECHNICAL APPROACH:

ALL OTHER FACTORS BEING EQUAL, THE CALCULATED DOSE FOR ANY GIVEN SCENARIO WILL BE LINEAR WITH RESPECT TO THE SOURCE TERM. THEREFORE, THE APPROACH USED HERE IS TO BASE THE CALCULATED CONTROL RM DOSES ON A CALCULATION USING IDENTICAL CALCULATION PARAMETERS AND ADJUSTING THIS ALREADY CALCULATED DOSE USING THE FOLLOWING RELATIONSHIP:

$$\frac{\text{SOURCE TERM FROM 8T12-M-01}}{\text{SOURCE TERM FROM REF. CALC.}} \left[\begin{matrix}\text{CONTROL ROOM DOSE} \\ \text{FROM THE REF. CALC}\end{matrix}\right]$$

THE REF. CALC. IS 5T73-M-04 (REF. 2). THIS CALCULATION IS BASED ON A STACK RELEASE AS IS THE RELEASE FOR THE CRDA. IN ADDITION, REF 2 INCLUDES THE FOLLOWING

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CALCULATIONAL PARAMETERS:

- CONTROL ROOM ISOLATION TAKES PLACE AT T = 5 SECONDS
- THE IODINE FRACTIONS ARE:

ORGANICS	=	4%
ELEMENTAL	=	91%
PARTICULATE	=	5%

- CONTROL ROOM FILTER EFF

ORGANICS	=	90%
ELEMENTAL	=	95%
PARTICULATE	=	95%

THE DURATION OF THE STACK RELEASE RESULTING FROM THE CRDA SCENARIO IS 10 MINUTES.

FOR THIS CALCULATION ONLY TWO SEGMENTS OF THE REF. 2 ANALYSIS ARE REQUIRED TO BE USED, THESE ARE T = 0 TO 5 SECONDS, AND 5 SECONDS TO ½ HOUR.

TO ACCOUNT FOR THE DIFFERENT FILTER

EFFICIENCIES, BASED ON THE "TYPE" OF IODINE, THE REF 3* SOURCE TERM WILL BE PARTITIONED BETWEEN THE THREE TYPES OF IODINES USING THE ABOVE FRACTIONS.

THE T=0 TO 5 SECOND SOURCE TERM FOR THE REF. 3 SOURCE TERM WILL BE CALCULATED AS FOLLOWS:

$$\left[\text{REF. 3 SOURCE TERM}\right] \times \left[\frac{5\ \text{SEC.}}{10\ \text{min.} \times 60\ \frac{\text{sec}}{\text{min}}}\right]$$

$$\left[\text{REF. 3 SOURCE TERM}\right] \times \left[8.33 \times 10^{-3}\right]$$

THE REMAINING QUANTITY REPRESENTS THE T=5 SEC TO 10 min RELEASE.

* REF. 3 IS THE CROA CALCULATION

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SUBJECT: CONT. RM. DOSES - MUPS - MANUAL ISOL.		CHECKED BY: [signature]	DATE: 12/12/93

CALCULATIONAL INPUTS:

SOURCE TERM FROM REF. 2 APPENDIX A

IODINE T=0 TO 5 SEC - RELEASE IN CURIES

IODINE	ELEMENTAL	PARTICULATE	ORGANIC
I-131	6.11×10^{-6}	6.72×10^{-8}	2.69×10^{-7}
I-132	9.09×10^{-6}	9.99×10^{-8}	4.00×10^{-7}
I-133	1.4×10^{-5}	1.54×10^{-7}	6.17×10^{-7}
I-134	1.64×10^{-5}	1.80×10^{-7}	7.19×10^{-7}
I-135	1.30×10^{-5}	1.84×10^{-7}	5.72×10^{-7}

CORRESPONDING CONTROL ROOM THYROID DOSES IN REM

I-131	3.07×10^{-10}	3.37×10^{-12}	1.37×10^{-11}
I-132	1.65×10^{-11}	1.81×10^{-13}	7.37×10^{-13}
I-133	1.90×10^{-10}	2.09×10^{-12}	8.51×10^{-12}
I-134	1.39×10^{-11}	1.52×10^{-13}	6.20×10^{-13}
I-135	5.47×10^{-11}	6.01×10^{-13}	2.45×10^{-12}

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FROM REF. 2

NOBLE GAS SOURCE TERM T = 0 TO 5 SEC IN CURIES

NUCLIDE	RELEASE	DOSES (REM)
Xe-131m	4.06×10^{-6}	7.48×10^{-16}
Xe-133m	3.14×10^{-5}	2.73×10^{-15}
Xe-133	1.24×10^{-3}	2.48×10^{-13}
Xe-135m	3.31×10^{-4}	2.93×10^{-13}
Xe-135	1.16×10^{-3}	6.04×10^{-13}
Xe-138	1.09×10^{-3}	1.67×10^{-12}
Kr-83m	5.60×10^{-5}	8.99×10^{-15}
Kr-85m	2.4×10^{-4}	8.42×10^{-14}
Kr-85	8.05×10^{-6}	3.58×10^{-17}
Kr-87	4.61×10^{-4}	6.38×10^{-13}
Kr-88	6.57×10^{-4}	1.97×10^{-12}

NOBLE GAS SOURCE TERM T = 5 SEC TO 1/2 HOUR - IN CURIES

NUCLIDE	RELEASE	DOSES (REM)
Xe-131m	5.22×10^{-1}	3.14×10^{-8}
Xe-133m	4.02×10^{0}	1.14×10^{-7}
Xe-133	1.59×10^{2}	1.04×10^{-5}
Xe-135m	1.86×10^{1}	4.73×10^{-6}
Xe-135	1.46×10^{2}	2.46×10^{-5}
Xe-138	6.62×10^{1}	2.99×10^{-5}
Kr-83m	1.09×10^{1}	3.28×10^{-7}
Kr-85m	2.93×10^{1}	3.33×10^{-6}
Kr-85	1.04×10^{0}	1.48×10^{-9}
Kr-87	4.92×10^{1}	2.19×10^{-5}
Kr-88	7.79×10^{1}	7.51×10^{-5}

IODINE SOURCE TERM T = 5 SEC TO 1/2 HOUR - IN CURIES
→ FROM REF. 2

IODINE	ELEMENTAL	PARTICULATE	ORGANIC
I-131	7.86×10^{-1}	8.63×10^{-3}	3.45×10^{-2}
I-132	1.06	1.16×10^{-2}	4.66×10^{-2}
I-133	1.79	1.97×10^{-2}	7.86×10^{-2}
I-134	1.62	1.78×10^{-2}	7.13×10^{-2}
I-135	1.62	1.78×10^{-2}	7.12×10^{-2}

CORRESPONDING CONTROL ROOM THYROID DOSES - IN REM

I-131	1.26×10^{-2}	1.38×10^{-4}	5.62×10^{-4}
I-132	6.05×10^{-4}	6.65×10^{-6}	2.71×10^{-5}
I-133	7.72×10^{-3}	8.49×10^{-5}	3.46×10^{-4}
I-134	4.25×10^{-4}	4.67×10^{-6}	1.9×10^{-5}
I-135	2.16×10^{-3}	2.38×10^{-5}	9.68×10^{-5}

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CRDA SOURCE TERM - FROM REV. 3 (CRDA CALC.)

NUCLIDES	TEN MINUTE RELEASE IN CURIES
I-131	9.46×10^2
I-132	1.38×10^3
I-133	1.97×10^3
I-134	1.90×10^3
I-135	1.86×10^3
Kr-83m	1.17×10^4
Kr-85m	2.51×10^4
Kr-85	1.12×10^3
Kr-87	4.40×10^4
Kr-88	6.82×10^4
Kr-89	9.55×10^3
Xe-131m	5.87×10^2
Xe-133m	8.57×10^3
Xe-133	2.06×10^5
Xe-135m	2.49×10^4
Xe-135	2.66×10^4
Xe-137	2.93×10^4
Xe-138	1.05×10^5

Halliburton NUS Corporation

STANDARD CALCULATION SHEET

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SUBJECT: CONT. Rm. DOSE - MVP's - MANUAL ISOL.		CHECKED BY: [signature]	DATE: 12/12/93

CALCULATIONS:

NOBLE GAS SOURCE TERMS FOR THE CRDA CONTROL DOSE ANALYSIS

NUCLIDE	REF. 3-10 MIN. RELEASE IN CURIES	CONT. Rm. DOSE - 5 SEC SOURCE TERM	CONT. Rm DOSE 5 sec - 1/2 HOUR SOURCE TERM
Kr-83m	1.17×10^4	9.73×10^1	1.16×10^4
Kr-85m	2.51×10^4	2.09×10^2	2.49×10^4
Kr-85	1.12×10^3	9.36	1.11×10^3
Kr-87	4.40×10^4	3.67×10^2	4.37×10^4
Kr-88	6.82×10^4	5.68×10^2	6.76×10^4
Kr-89	9.55×10^3	7.96×10^1	9.47×10^3
Xe-131m	5.87×10^2	4.89	5.82×10^2
Xe-133m	8.57×10^3	7.14×10^1	8.50×10^3
Xe-133	2.06×10^5	1.72×10^3	2.04×10^5
Xe-135m	2.49×10^4	2.08×10^2	2.47×10^4
Xe-135	2.66×10^4	2.22×10^2	2.64×10^4
Xe-137	2.93×10^4	2.44×10^2	2.90×10^4
Xe-138	1.05×10^5	8.73×10^2	1.04×10^5

Halliburton NUS Corporation — STANDARD CALCULATION SHEET

CLIENT: CP&L - BSEP	FILE NO.: 7T12-M-07	BY: [signature]	PAGE 15 OF 25
SUBJECT: Cont. Rm. Doses - MYB - Manual Isol.		CHECKED BY: [signature]	DATE: 12/12/93

NOBLE GAS CONTROL RM. WHOLE BODY DOSE T = 0 TO 5 SEC

NUCLIDE	CURIES REF. 3	CURIES REF. 2	REF. 3 / REF. 2	× REF. 2 T=0 TO 5 SEC CONTROL RM DOSE	= CRDA T=0 TO 5 SEC CONTROL RM DOSE
Kr-83m	9.73×10^{1}	9.6×10^{-5}	1.01×10^{6}	8.99×10^{-15}	9.11×10^{-9}
Kr-85m	2.09×10^{2}	2.4×10^{-4}	8.70×10^{5}	8.42×10^{-14}	7.32×10^{-8}
Kr-85	9.36	8.05×10^{-6}	1.16×10^{6}	3.58×10^{-17}	4.16×10^{-11}
Kr-87	3.67×10^{2}	4.61×10^{-4}	7.96×10^{5}	6.38×10^{-13}	5.08×10^{-7}
Kr-88	5.68×10^{2}	6.57×10^{-4}	8.65×10^{5}	1.97×10^{-12}	1.70×10^{-6}
Kr-89	7.96×10^{1}	NO DATA	N/A	NO DATA	2.65×10^{-7} (a)
Xe-131m	4.89	4.06×10^{-6}	1.21×10^{6}	7.48×10^{-16}	9.01×10^{-10}
Xe-133m	7.14×10^{1}	3.14×10^{-5}	2.27×10^{6}	2.73×10^{-15}	6.21×10^{-9}
Xe-133	1.72×10^{3}	1.24×10^{-3}	1.38×10^{6}	2.48×10^{-13}	3.43×10^{-7}
Xe-135m	2.08×10^{2}	3.31×10^{-4}	6.28×10^{5}	2.93×10^{-13}	1.84×10^{-7}
Xe-135	2.22×10^{2}	1.16×10^{-3}	1.91×10^{5}	6.04×10^{-13}	1.15×10^{-7}
Xe-137	2.44×10^{2}	NO DATA	N/A	NO DATA	3.44×10^{-8} (b)
Xe-138	8.73×10^{2}	1.09×10^{-3}	8.01×10^{5}	1.67×10^{-12}	1.34×10^{-6}

T = 0 TO 5 SEC WHOLE BODY DOSE = 4.58×10^{-6} REM

NOTES:

a) Kr-88 WAS 10.34% OF THE TOTAL KRYPTON NOBLE GAS DOSE FOR THE CALCULATED DOSE IN REF. 3. SINCE THE RADIONUCLIDE CONCENTRATIONS IS BASED ON REF 3 INVENTORIES, THIS % WAS ASSUMED TO REMAIN CONSTANT.

b) Xe-137 WAS 1.7% OF THE TOTAL XENON NOBLE GAS DOSE. SEE NOTE a FOR FURTHER EXPLAINATION OF THIS ESTIMATE

Halliburton NUS Corporation — STANDARD CALCULATION SHEET

CLIENT: CP&L - BSEP	FILE NO.: 8T12-M-02	BY: [signature]	PAGE 16 OF 25
SUBJECT: Cont. Rm. Doses - MVPs - Manual Isol.		CHECKED BY: [signature]	DATE: 12/12/93

NOBLE GAS CONTROL RM. WHOLE BODY DOSE T = 5 SEC TO ½ HOUR.

NUCLIDE	CURIES REF. 3	CURIES REF. 2	REF. 3 / REF. 2	× REF. 2 T = 5 SEC TO ½ HOUR CONTROL ROOM DOSE	= CRDA T = 5 SEC TO ½ HOUR CONTROL ROOM DOSE
Kr-83m	1.16×10^{4}	10.9	1.06×10^{3}	3.28×10^{-7}	3.48×10^{-4}
Kr-85m	2.48×10^{4}	29.3	8.48×10^{2}	3.33×10^{-6}	2.82×10^{-3}
Kr-85	1.11×10^{3}	1.04	1.07×10^{3}	1.48×10^{-9}	1.58×10^{-6}
Kr-87	4.37×10^{4}	49.6	8.80×10^{2}	2.19×10^{-5}	1.93×10^{-2}
Kr-88	6.76×10^{4}	77.9	8.68×10^{2}	7.51×10^{-5}	6.52×10^{-2}
Kr-89	9.47×10^{3}	NO DATA	N/A	N/A	1.01×10^{-2} (a)
Xe-131m	5.82×10^{2}	0.522	1.12×10^{3}	3.14×10^{-8}	3.50×10^{-5}
Xe-133m	8.50×10^{3}	4.02	2.11×10^{3}	1.14×10^{-7}	2.41×10^{-4}
Xe-133	2.04×10^{5}	159	1.28×10^{3}	1.04×10^{-5}	1.34×10^{-2}
Xe-135m	2.47×10^{4}	18.6	1.33×10^{3}	4.73×10^{-6}	6.29×10^{-3}
Xe-135	2.64×10^{4}	146	1.81×10^{2}	2.46×10^{-5}	4.44×10^{-3}
Xe-137	2.90×10^{4}	NO DATA	N/A	N/A	1.24×10^{-3} (b)
Xe-138	1.04×10^{5}	66.2	1.57×10^{3}	2.99×10^{-5}	4.69×10^{-2}

T = 5 SEC TO ½ HOUR WHOLE BODY DOSE = **0.17 REM**

NOTES:

a) SEE NOTE "a" ON PAGE 15

b) SEE NOTE "b" ON PAGE 15

Halliburton NUS Corporation	STANDARD CALCULATION SHEET

CLIENT:	FILE NO.:	BY:	PAGE 17 OF 25
CP&L - BSEP	8T12-M-02	[signature]	
SUBJECT: Cont. Rm. Doses - MVPs - Manual Isol.		CHECKED BY: [signature]	DATE: 12/12/93

IODINE SOURCE TERM FOR THE CRDA (T=0 TO 10 MIN.)

NUCLIDE	SOURCE TERM CURIES
I-131	9.46×10^2
I-132	1.38×10^3
I-133	1.97×10^3
I-134	1.90×10^3
I-135	1.86×10^3

NUCLIDE	SOURCE TERM CURIES
ELEMENTAL (91%)	
I-131	8.60×10^2
I-132	1.25×10^3
I-133	1.80×10^3
I-134	1.73×10^3
I-135	1.70×10^3
PARTICULATE (4%)	
I-131	3.78×10^1
I-132	5.52×10^1
I-133	7.90×10^1
I-134	7.62×10^1
I-135	7.45×10^1
ORGANIC (5%)	
I-131	4.73×10^1
I-132	6.90×10^1
I-133	9.87×10^1
I-134	9.52×10^1
I-135	9.32×10^1

Halliburton NUS Corporation	STANDARD CALCULATION SHEET

CLIENT: CP&L - BSEP	FILE NO.: 8T12-M-02	BY: [signature]	PAGE 18 OF 25
SUBJECT: Cont. Rm. Doses - MVIS - Manual Isol		CHECKED BY: [signature]	DATE: 12/12/93

NUCLIDE	REF. 3 - 10 MIN RELEASE IN CURIES	CONTROL Rm. T=0 TO 5 SEC SOURCE TERM	CONTROL Rm T=5 SEC TO 1/2 HOUR SOURCE TERM
ELEM.			
I-131	8.60×10^{2}	7.17	8.53×10^{2}
I-132	1.25×10^{3}	10.46	1.24×10^{3}
I-133	1.80×10^{3}	14.97	1.78×10^{3}
I-134	1.73×10^{3}	14.44	1.72×10^{3}
I-135	1.70×10^{3}	14.13	1.68×10^{3}
PARTIC.			
I-131	3.78×10^{1}	3.15×10^{-1}	3.75×10^{1}
I-132	5.52×10^{1}	4.60×10^{-1}	5.47×10^{1}
I-133	7.90×10^{1}	6.58×10^{-1}	7.83×10^{1}
I-134	7.62×10^{1}	6.35×10^{-1}	7.56×10^{1}
I-135	7.45×10^{1}	6.21×10^{-1}	7.39×10^{1}
ORGANIC			
I-131	4.73×10^{1}	3.94×10^{-1}	4.69×10^{1}
I-132	6.90×10^{1}	5.75×10^{-1}	6.84×10^{1}
I-133	9.87×10^{1}	8.22×10^{-1}	9.79×10^{1}
I-134	9.52×10^{1}	7.94×10^{-1}	9.44×10^{1}
I-135	9.32×10^{1}	7.76×10^{-1}	9.24×10^{1}

Halliburton NUS Corporation — STANDARD CALCULATION SHEET

CLIENT:	FILE NO.:	BY:	PAGE 19 OF 25
CP&L - BSEP	8T12-M-02	[signature]	
SUBJECT: CONT. Rm. DOSES - MYB - MANUAL II.01		CHECKED BY: [signature]	DATE: 12/12/93

IODINE CONTROL RM WHOLE BODY DOSE T = 0 TO 5 SEC

NUCLIDE	CURIES REF. 3 (a)	CURIES REF. 2	REF. 3 / REF. 2	× REF. 2 T=0 TO 5 SEC CONTROL RM DOSE (b)	= CRDA T=0 TO 5 SEC CONTROL RM WB DOSE
ELEMENT.					
I-131	7.17	6.11×10^{-6}	1.17×10^{6}	3.53×10^{-15}	4.14×10^{-9}
I-132	10.46	9.09×10^{-6}	1.15×10^{6}	2.87×10^{-14}	3.30×10^{-8}
I-133	14.97	1.4×10^{-5}	1.07×10^{6}	1.28×10^{-14}	1.37×10^{-8}
I-134	14.44	1.64×10^{-5}	8.81×10^{5}	5.53×10^{-14}	4.87×10^{-8}
I-135	14.13	1.3×10^{-5}	1.09×10^{6}	2.38×10^{-14}	2.59×10^{-8}
PARTICULATE					
I-131	3.15×10^{-1}	6.72×10^{-8}	4.69×10^{6}	3.88×10^{-17}	1.82×10^{-10}
I-132	4.60×10^{-1}	9.99×10^{-8}	4.60×10^{6}	3.15×10^{-16}	1.45×10^{-9}
I-133	6.58×10^{-1}	1.54×10^{-7}	4.27×10^{6}	1.41×10^{-16}	6.02×10^{-10}
I-134	6.35×10^{-1}	1.80×10^{-7}	3.53×10^{6}	6.07×10^{-16}	2.14×10^{-9}
I-135	6.21×10^{-1}	1.84×10^{-7}	3.38×10^{6}	2.62×10^{-16}	8.84×10^{-10}
ORGANIC					
I-131	3.94×10^{-1}	2.69×10^{-7}	1.46×10^{6}	1.58×10^{-16}	2.31×10^{-10}
I-132	5.75×10^{-1}	4.00×10^{-7}	1.44×10^{6}	1.28×10^{-15}	1.84×10^{-9}
I-133	8.22×10^{-1}	6.17×10^{-7}	1.33×10^{6}	5.74×10^{-16}	7.65×10^{-10}
I-134	7.94×10^{-1}	7.19×10^{-7}	1.10×10^{6}	2.47×10^{-15}	2.73×10^{-9}
I-135	7.76×10^{-1}	5.72×10^{-7}	1.36×10^{6}	1.06×10^{-15}	1.44×10^{-9}
				SUBTOTAL =	1.38×10^{-7} REM

NOTES:

a) FROM PAGE 18 OF THIS CALC. (b) REF. 2 App. "A" PAGE 3

IODINE CONTROL RM WHOLE BODY DOSE T=5 SEC TO 1/2 HOUR

NUCLIDE	CURIES REF. 3 (a)	CURIES REF. 2	REF. 3 / REF. 2	× REF. 2 T=5 SEC TO 1/2 HOUR CONTROL RM. DOSE (b)	= CR DOSE T=5 SEC TO 1/2 HOUR CONTROL RM W.B. DOSE
ELEMENT.					
I-131	8.53×10^{2}	7.86×10^{-1}	1.09×10^{3}	1.45×10^{-7}	1.57×10^{-4}
I-132	1.24×10^{3}	1.06	1.17×10^{3}	1.05×10^{-6}	1.23×10^{-3}
I-133	1.78×10^{3}	1.79	9.95×10^{2}	5.21×10^{-7}	5.18×10^{-4}
I-134	1.72×10^{3}	1.62	1.06×10^{3}	1.69×10^{-6}	1.79×10^{-3}
I-135	1.68×10^{3}	1.62	1.04×10^{3}	9.41×10^{-7}	9.77×10^{-4}
PARTICULATE					
I-131	3.75×10^{1}	8.63×10^{-3}	4.35×10^{3}	1.59×10^{-9}	6.91×10^{-6}
I-132	5.47×10^{1}	1.16×10^{-2}	4.72×10^{3}	1.16×10^{-8}	5.47×10^{-5}
I-133	7.83×10^{1}	1.97×10^{-2}	3.97×10^{3}	5.73×10^{-9}	2.28×10^{-5}
I-134	7.56×10^{1}	1.78×10^{-2}	4.24×10^{3}	1.86×10^{-8}	7.90×10^{-5}
I-135	7.39×10^{1}	1.78×10^{-2}	4.15×10^{3}	1.03×10^{-8}	4.28×10^{-5}
ORGANIC					
I-131	4.69×10^{1}	3.45×10^{-2}	1.36×10^{3}	6.47×10^{-9}	8.79×10^{-6}
I-132	6.84×10^{1}	4.66×10^{-2}	1.47×10^{3}	4.71×10^{-8}	6.91×10^{-5}
I-133	9.79×10^{1}	7.86×10^{-2}	1.25×10^{3}	2.33×10^{-8}	2.90×10^{-5}
I-134	9.44×10^{1}	7.13×10^{-2}	1.32×10^{3}	7.58×10^{-8}	1.00×10^{-4}
I-135	9.24×10^{1}	7.12×10^{-2}	1.30×10^{3}	4.21×10^{-8}	5.46×10^{-5}

SUBTOTAL = 5.15×10^{-3} REM

NOTES:
a) FROM PAGE 18 OF THIS CALC. b) REF. 2 APP "A" PAGE 3

Halliburton NUS Corporation	STANDARD CALCULATION SHEET

CLIENT: CP&L - BSEP	FILE NO.: 8T12-M-02	BY: [signature]	PAGE 21 OF 25
SUBJECT: CONT. RM DOSES - MVPs - MANUAL Isol		CHECKED BY: [signature]	DATE: 12/12/93

TOTAL CONTROL ROOM WHOLE BODY DOSE

- TOTAL NOBLE GAS WB DOSE (PAGE 16) — 1.70×10^{-1} REM
- IODINE WB DOSE T=0 TO 5 SEC (PAGE 19) — 1.38×10^{-7} REM
- IODINE WB DOSE T=5 SEC TO ½ Hr (PAGE 20) — 5.15×10^{-3} REM

TOTAL WHOLE BODY DOSE = 1.75×10^{-1} REM

CONTROL ROOM THYROID DOSE T = 0 TO 5 SEC

NUCLIDE	CURIES REF. 3[a]	CURIES REF. 2[b]	REF. 3 / REF. 2	×	REF. 2 T=0 TO 5 SEC CONTROL Rm DOSE[b]	=	CRDA T=0 TO 5 SEC CONT. Rm THYROID DOSE
ELEMENTAL							
I-131	7.17	6.11×10^{-6}	1.17×10^{6}		3.07×10^{-10}		3.60×10^{-4}
I-132	1.05×10^{1}	9.09×10^{-6}	1.15×10^{6}		1.65×10^{-11}		1.90×10^{-5}
I-133	1.50×10^{1}	1.4×10^{-5}	1.07×10^{6}		1.90×10^{-10}		2.03×10^{-4}
I-134	1.44×10^{1}	1.64×10^{-5}	8.81×10^{5}		1.39×10^{-11}		1.22×10^{-5}
I-135	1.41×10^{1}	1.3×10^{-5}	1.09×10^{6}		5.47×10^{-11}		5.95×10^{-5}
PARTICULATE							
I-131	3.15×10^{-1}	6.72×10^{-8}	4.69×10^{6}		3.37×10^{-12}		1.58×10^{-5}
I-132	4.60×10^{-1}	9.99×10^{-8}	4.60×10^{6}		1.81×10^{-13}		8.33×10^{-7}
I-133	6.58×10^{-1}	1.54×10^{-7}	4.27×10^{6}		2.09×10^{-12}		8.93×10^{-6}
I-134	6.35×10^{-1}	1.80×10^{-7}	3.53×10^{6}		1.52×10^{-13}		5.36×10^{-7}
I-135	6.21×10^{-1}	1.84×10^{-7}	3.38×10^{6}		6.01×10^{-13}		2.03×10^{-6}
ORGANIC							
I-131	3.94×10^{-1}	2.69×10^{-7}	1.46×10^{6}		1.37×10^{-11}		2.01×10^{-5}
I-132	5.75×10^{-1}	4.00×10^{-7}	1.44×10^{6}		7.37×10^{-13}		1.06×10^{-6}
I-133	8.22×10^{-1}	6.17×10^{-7}	1.33×10^{6}		8.51×10^{-12}		1.13×10^{-5}
I-134	7.94×10^{-1}	7.19×10^{-7}	1.10×10^{6}		6.20×10^{-13}		6.84×10^{-7}
I-135	7.76×10^{-1}	5.72×10^{-7}	1.36×10^{6}		2.45×10^{-12}		3.33×10^{-6}

7.19×10^{-4} REM

NOTES:

a) PAGE 18 OF THIS CALC

b) PAGE 10 OF THIS CALC

Halliburton NUS Corporation — STANDARD CALCULATION SHEET

CLIENT:	FILE NO.:	BY:	PAGE 23 OF 25
CP&L - BSEP	8T12-M-02	[signature]	
SUBJECT: CONT. RM. DOSES - MUPS - MANUAL ISOL.		CHECKED BY: [signature]	DATE: 12/12/93

CONTROL ROOM THYROID DOSE T~5 SEC TO 1/2 HOUR

NUCLIDE	CURIES REF. 3 (a)	CURIES REF. 2 (b)	REF. 3 / REF. 2	× REF. 2 T=5 SEC TO 1/2 HOURS CONTROL RM. DOSE (5)	= CRDR T=5 SEC TO 1/2 HOUR CONTROL ROOM THYROID DOSE
ELEMENTAL					
I-131	8.53×10^{2}	7.86×10^{-1}	1.09×10^{3}	1.26×10^{-2}	1.37×10^{1}
I-132	1.24×10^{3}	1.06	1.17×10^{3}	6.05×10^{-4}	7.10×10^{-1}
I-133	1.78×10^{3}	1.79	9.95×10^{2}	7.72×10^{-3}	7.68
I-134	1.72×10^{3}	1.62	1.06×10^{3}	4.25×10^{-4}	4.51×10^{-1}
I-135	1.68×10^{3}	1.62	1.04×10^{3}	2.16×10^{-3}	2.24
PARTICULATE					
I-131	3.75×10^{1}	8.63×10^{-3}	4.35×10^{3}	1.38×10^{-4}	6.00×10^{-1}
I-132	5.47×10^{1}	1.16×10^{-2}	4.72×10^{3}	6.65×10^{-6}	3.14×10^{-2}
I-133	7.83×10^{1}	1.97×10^{-2}	3.97×10^{3}	8.49×10^{-5}	3.37×10^{-1}
I-134	7.56×10^{1}	1.78×10^{-2}	4.24×10^{3}	4.67×10^{-6}	1.98×10^{-2}
I-135	7.39×10^{1}	1.78×10^{-2}	4.15×10^{3}	2.38×10^{-5}	9.88×10^{-2}
ORGANIC					
I-131	4.69×10^{1}	3.45×10^{-2}	1.36×10^{3}	5.62×10^{-4}	7.64×10^{-1}
I-132	6.84×10^{1}	4.66×10^{-2}	1.47×10^{3}	2.71×10^{-5}	3.98×10^{-2}
I-133	9.79×10^{1}	7.86×10^{-2}	1.25×10^{3}	3.46×10^{-4}	4.31×10^{-1}
I-134	9.44×10^{1}	7.13×10^{-2}	1.32×10^{3}	1.9×10^{-5}	2.52×10^{-2}
I-135	9.34×10^{1}	7.12×10^{-2}	1.30×10^{3}	9.68×10^{-5}	1.26×10^{-1}
					2.72×10^{1} REM

NOTES:

a) PAGE 18 OF THIS CALC

b) PAGE 12 OF THIS CALC

SUMMARY:

THE CALCULATED CONTROL ROOM DOSES FOLLOWING THE POSTULATED CRDA DEFINED IN THIS CALCULATION ARE AS FOLLOWS:

TOTAL WHOLE BODY DOSE	0.17 REM
TOTAL THYROID DOSE	27.2 REM

THESE VALUES ARE BELOW THE LIMITS OF 10 CFR 50 App. A, GDC-19.

ALSO THESE CALCULATED VALUES ARE BASED ON THE MAXIMUM END OF CORE LIFE SOURCE TERM TAKEN FROM REF. 1.

REFERENCES:

1- SAFETY EVALUATION FOR ELIMINATING THE BOILING WATER REACTOR MAIN STEAM ISOLATION VALVE CLOSURE FUNCTION AND SCRAM FUNCTION OF THE MAIN STEAM LINE RADIATION MONITOR; NEDO-31400 CLASS 1, MAY 1987; GENERAL ELECTRIC

2- NUS CALCULATION 5T73-M-04 REV. 0, IMPACT OF STANDBY GAS TREATMENT FLOWS ON LOCA ANALYSIS, 4/14/93

3- NUS CALCULATION 8T12-M-01 REV. 1, MANUAL ISOLATION OF THE CONDENSER MECHANICAL VACUUM PUMP UPON A HIGH MSLRM SIGNAL, 12/17/93 | R1

ENCLOSURE 2

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1 AND 2
NRC DOCKETS 50-325 & 50-324
OPERATING LICENSES DPR-71 & DPR-62
REQUEST FOR LICENSE AMENDMENTS
RESPONSE TIME TABLE RELOCATION

10 CFR 50.92 EVALUATION

The Commission has provided standards in 10 CFR 50.92(c) for determining whether a significant hazards consideration exists. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. Carolina Power & Light Company has reviewed this proposed license amendment request and believes that its adoption would not involve a significant hazards consideration. The basis for this determination follows.

1. The proposed amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated. The deletion of the MSLRM trip function from the reactor scram and the Group 1 isolation initiation logic removes a potential transient initiation and therefore decreases the probability of plant transients occurring due to inadvertent scrams resulting from this system.

   The deletion of the MSLRM trip function from the Main Steam Drain Valve, the Reactor Water Sample Isolation Valve, and the Mechanical Vacuum Pump line isolation logic, does not affect the initiators of any accident previously evaluated in the Safety Analysis Report. Therefore, the proposed change does not involve an increase in the probability of occurrence of any accident previously evaluated.

   The NRC staff acceptance criterion for the Control Rod Drop Accident is that the doses from the accident fall significantly below the limits given in 10 CFR Part 100. The releases calculated for accident during plant operations when the Steam Jet Air Ejectors (SJAE) are operating and when the Mechanical Vacuum Pumps are operating are within these acceptance limits.

   In NEDO-31400, GE shows that the occurrence of a CRDA, with the MSL high radiation isolation removed, and SJAE in operation, results in offsite radiological exposures that are small fractions of 10CFR100 guidelines. Since the Brunswick specific CRDA doses are lower than the calculated by GE and the GE dose parameters envelope those used for the Brunswick analysis, it is concluded that the NRC's findings that the radiological release consequence is within the staff's acceptance criteria, even without the automatic MSIV trip, is applicable to Brunswick.

   While not specifically addressed in the GE evaluation, Carolina Power and Light also proposes to eliminate the Main Steam Line Drain valves, the Reactor Water Sample Line isolation valves, and the mechanical vacuum line isolation valves from the MSLRM isolation logic. Main Steam Line Drain Valves B21-F016 and B21-F019 drain to the main condenser, which is

the same flow path as the MSIVs. The discharge of both the MSIV and MSL drain flow paths is processed through the offgas system. Any radiation released through the drain valves during a control rod drop accident will be negligible and, for Brunswick, is bounded by the NEDO analysis.

The reactor water sample line provides a small amount of reactor water to the Reactor Building Sample Panel. The discharge of the Reactor Building Sample Panel is routed through the floor drain sump to the liquid radwaste system. Any release through this path would be negligible and, for Brunswick, is bounded by the NEDO analysis.

The mechanical vacuum pumps are used only when the reactor is at low power (less than 5%) and there is insufficient steam flow to operate the Steam Jet Air Ejectors. The increase in radiation will be detected by the MSLRMs and annunciated in the Main Control Room. Operators will be instructed, in the annunciator response procedures, to take action to stop the Mechanical Vacuum Pump(s) and isolate the Mechanical Vacuum Pump line. The amount of radiation released prior to isolating the line would represent the most limiting case for this accident. However, it will still be well within 10 CFR Part 100 limits. Additionally, the dose received in the Main Control Room as a result of this accident is within General Design Criteria 19 (SRP 6.4) limits.

Therefore, since elimination of the MSIV scram and isolation functions would not result in an increase in exposure above NRC acceptance limits, the proposed changes will not significantly increase the consequences of a previously evaluated accident.

2. The proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated. The function of a MSLRM trip is to detect abnormal fission product release and isolate the steam lines, thereby stopping the transport of fission products from the reactor to the main condenser. The monitors do not perform a prevention function for any kind of accident. The existence of a MSLRM trip does not prevent the occurrence of a fuel failure event or any other type of event. The elimination of these signals, which served only in a mitigative function, does not create the possibility of a new or different kind of accident from those previously evaluated. Also, radiation monitors with alarm functions will remain installed in the plant to warn the operators of a high radiation condition in the main steam lines, or in the off-gas system. Thus no new or different accident can be postulated by the proposed changes.

3. The proposed amendments do not involve a significant reduction in a margin of safety. As shown in the topical report, the changes represent an overall improvement in plant safety. Safe operation of the plant is further enhanced by elimination of the unnecessary scram and isolation of the reactor vessel. With implementation of these changes, 1) the primary heat sink (main condenser) remains available, 2) large transients on the reactor vessel, as well as challenges to the ESF, are avoided, and 3) the Offgas system remains available to control the pathway of potential releases. As such, the margin of safety is enhanced by the proposed changes.

ENCLOSURE 3

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1 AND 2
NRC DOCKETS 50-325 & 50-324
OPERATING LICENSES DPR-71 & DPR-62
REQUEST FOR LICENSE AMENDMENTS
RESPONSE TIME TABLE RELOCATION

ENVIRONMENTAL CONSIDERATIONS

10 CFR 51.22(c)(9) provides criterion for and identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration, (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (3) result in an increase in individual or cumulative occupational radiation exposure. Carolina Power & Light Company has reviewed this request and believes that the proposed amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(c), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the amendment. The basis for this determination follows.

1. The amendments do not involve a significant hazards consideration, as shown in Enclosure 2.

2. The amendments do not result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite. As shown in NEDO-31400, and in CP&L's plant specific evaluation of the proposed change, the resulting releases from this change are within NRC acceptance limits. The change also does not affect the type of effluents released since the accident source term is unaffected.

3. The amendments do not result in an increase in individual or cumulative occupational radiation exposure. As shown in NEDO-31400, and in CP&L's plant specific evaluation of the proposed change, the resulting releases from this change are within NRC acceptance limits for individual exposures. In addition, reduction of the condenser offgas radiation monitor setpoint further minimizes plant contamination and personnel exposures which may result from a postulated fuel failure that would not be detected by the MSLRMs.