

## PURPOSE

The purpose of this calculation is to determine if the control room air treatment system is required during a control rod drop accident (CRDA) to maintain control room doses within 10CFR50 Appendix A, GDC 19 acceptance criteria.

REASON FOR REVISION 1: The purpose of revision 1 is to incorporate independent reviewers comments, design inputs relating to control room normal ventilation intake flow rate, control room air volume, and to evaluate a ground level release to be consistent with SRP 15.4.9 (loss of off-site power).

## METHODOLOGY

The UFSAR CRDA analysis assumed a maximum of 1% of the noble gas activity and 0.5% of the halogen activity in a fuel rod are released to the coolant. The SRP assumes 10% of the noble gas and iodines are released to the coolant. The resultant halogen and noble gas radioactivity described in the UFSAR assumed to be released to the coolant as a result of a CRDA is compared to the calculated reactor coolant source term using SRP 15.4.9 methodology. For conservatism, the higher of the two coolant source terms is input to the Stone and Webster computer code DRAGON.

Two cases are analyzed. Case 1 assumes elevated release to be consistent with the UFSAR (REF 1.d) and case 2 assumes ground level release to be consistent with SRP 15.4.9 (loss of offsite power).

## DATA/ASSUMPTIONS

Nominal Reactor Power is 1850 MWt. A 2 % uncertainty is added. Therefore the Reactor Power used in this calculation is 1887. (REF 1a, 6)

850 fuel rods exceed 170 cal/gm, which is the enthalpy limit for eventual cladding perforation. Since U1's peak enthalpy will be less than 280 cal/gm, melting does not occur.(REF 2,3, and 4)

Fuel assumptions.(REF 1.b, 1.c, and 7)

532 fuel assemblies.

62 rods/fuel assembly.

32,632 total rods. This number was taken from radiological calculation 1H-009, Control Rod Drop Accident. The total rods were based on the actual fuel type in the Unit 1 core in 1991. Using only 8x8 fuel the total number of rods would be 62 by 532 = 32,984. This would result in an approximate 1 % decrease in source term (850/32,984 versus 850/32,632). Therefore, the dose contribution using the 1991 total number of rods are conservative and also negligible.

Case 1: elevated release. (REF 1.d). Case 2: ground level release (REF 4). (see METHODOLOGY section).

Release fractions (REF 4)

Amount of fuel gap activity is 10% of total activity in the rod (REF 5)

Fuel rods presumed failed are assumed to have operated at 1.5 times that of the average power level in the core.

All of noble gas and iodine gas activity instantaneously and uniformly mix in reactor coolant in the pressure vessel at the time of the accident. (REF 5)

10% of the iodines and 100% of the noble gases are released from the reactor pressure vessel to the turbine and condenser.

All noble gases in the turbine and condenser are available for release.

90% of the iodines are assumed to be removed by plateout and partitioning in the turbine and the condenser leaving only 10 % airborne and available for leakage.

The turbine and condensers leak to the atmosphere at 1% / day for a period of 24 hours, at which time the leak terminates.

Main Condenser volume:  $5.00E+04$  ft<sup>3</sup> (REF 7) - Not needed for this calculation since the DRAGON computer code uses fractions per day times the activity available for release to determine a release rate.

Control Room free volume :  $1.31E+05$  ft<sup>3</sup>. (REF 8)

The control room normal ventilation intake flow rate is  $2250 \text{ cfm} \pm 10\%$ .  $2250 \text{ cfm} + 10\% = 2475 \text{ cfm}$ , use  $2500 \text{ cfm}$ . (REF 14) . An additional  $30 \text{ cfm}$  in-leakage is to account for  $10 \text{ cfm}$  inleakage to the control room assumed in accordance with SRP 6.4, section III.3.d.(2).(ii) and  $20 \text{ cfm}$  is assumed to account for an unfiltered inleakage from a drain (REF 11).

Credit is not taken for Control Room Air Treatment System initiation to determine if filtration of the intake air is required to meet 10CFR50 Appendix A GDC 19 criteria.

U1 stack 0-2 hour X/Q is  $3.12E-04 \text{ sec/m}^3$  and the 2-720 hour X/Q is  $1.22E-08 \text{ sec/m}^3$ .

U1 turbine building blowout panel 0-2 hour X/Q is  $1.93E-03 \text{ sec/m}^3$ . This value is conservatively used for the postulated 24 hour release duration. (REF 9)

Radioactivity assumed in the coolant as a result of a postulated CRDA as described in the UFSAR is: Halogens =  $5.62E+04 \text{ Ci}$  and Noble Gas =  $6.64E+04 \text{ Ci}$ . (REF 1.d)

Breathing rate of  $3.47E-04 \text{ m}^3 / \text{sec}$  is conservatively assumed for the duration (0-720 hours) of the accident. (REF 12)

10CFR50 Appendix A, GDC 19 dose limit of 5 rem whole body or equivalent. This equates to 30 rem thyroid and 30 rem beta (skin) (REF 13).

## CALCULATION

### 1.0 COOLANT SOURCE TERM BASED ON SRP METHODOLOGY

ISOTOPE	Ci/MWt ①	TABLE 2		TOTAL CORE ACTIVITY ③ Ci	ACTIVITY IN COOLANT ④ Ci
		MWt ②			
I-131	2.90E+04	1887		5.47E+07	2.14E+05
I-132	4.20E+04	1887		7.93E+07	3.10E+05
I-133	4.80E+04	1887		9.06E+07	3.54E+05
I-134	6.20E+04	1887		1.17E+08	4.57E+05
I-135	4.90E+04	1887		9.25E+07	3.61E+05
				<b>TOTAL</b>	<b>1.70E+06</b>
KR-83M	3.00E+03	1887		5.66E+06	2.21E+04
KR-85M	6.50E+03	1887		1.23E+07	4.79E+04
KR-85	3.00E+02	1887		5.66E+05	2.21E+03
KR-87	1.20E+04	1887		2.26E+07	8.85E+04
KR-88	1.70E+04	1887		3.21E+07	1.25E+05
KR-89	2.00E+04	1887		3.77E+07	1.47E+05
XE-131M	1.80E+02	1887		3.40E+05	1.33E+03
XE-133M	2.00E+02	1887		3.77E+05	1.47E+03
XE-133	5.60E+04	1887		1.06E+08	4.13E+05
XE-135M	1.70E+04	1887		3.21E+07	1.25E+05
XE-135	9.80E+03	1887		1.85E+07	7.23E+04
XE-138	4.40E+04	1887		8.30E+07	3.24E+05
				<b>TOTAL</b>	<b>1.37E+06</b>

① GE BWR 6, Technical Description of a single cycle Boiling Water Reactor Nuclear System, January 1, 1974 (Table F.2.3-9)

② 1850 MWt \* 1.02 (DATA/ASSUMPTION # 1)

③ Column ① \* Column ②

④ Column ③ \* (850 failed rods / 32,632 total rods)\*1.5 peaking factor \* 0.1 (DATA/ASSUMPTIONS #2, 3, 5.a, 5.b, and 5.c)

As can be seen by comparing the total halogen and noble gas activities in Table 2 Column 4 to those described in the UFSAR (DATA / ASSUMPTIONS #11, the activity in the coolant using SRP methodology is greater than that the UFSAR coolant source term. Therefore the activity in the coolant using SRP methodology will be for this calculation.

### 2.0 ACTIVITY AVAILABLE IN CONDENSER

As stated in DATA/ASUMPTIONS 5.d all the noble gases and 10% of the iodines are released to the turbine and condenser. Therefore, Table 1 Column ④ is multiplied by 0.1 for iodines and 1.0 for noble gases to determine activity in the condenser.

TABLE 3	
ISOTOPE	ACTIVITY TO CONDENSER
I-131	2.14E+04
I-132	3.10E+04
I-133	3.54E+04
I-134	4.57E+04
I-135	3.61E+04
KR-83M	2.21E+04
KR-85M	4.79E+04
KR-85	2.21E+03
KR-87	8.85E+04
KR-88	1.25E+05
KR-89	1.47E+05
XE-131M	1.33E+03
XE-133M	1.47E+03
XE-133	4.13E+05
XE-135M	1.25E+05
XE-135	7.23E+04
XE-138	3.24E+05

The above condenser activity is input to DRAGON runs # 9014 and 8946 dated 5/16/98. The card inputs to these runs are included in Appendix A.

DRAGON inputs for case 1:

Main Condenser volume: 5.00E+04 ft<sup>3</sup> (DATA/ASSUMPTION # 6)  
 Main Condenser release rate: 0.01 fractions per day (DATA/ASSUMPTION #5.g)  
 Unit 1 Control Room (DATA/ASSUMPTIONS # 7 - 9)  
     volume: 1.31 E+05 ft<sup>3</sup>  
     intake rate 0 to 720 hours: 2530 cfm  
     filter efficiencies 0 - 720 hours = 0  
 Breathing rate 0 - 720 hours is 3.47E-04 m<sup>3</sup> /sec (DATA/ASSUMPTION #12)  
 0-2 hour X/Q: 3.12E-04 sec/m<sup>3</sup> 2 - 24 hour X/Q: 1.22E-08 sec/m<sup>3</sup>  
 (DATA/ASSUMPTION #10).  
 Fraction of iodine inventory available for release is 0.1  
 Fraction of noble gas inventory available for release is 1.0

DRAGON inputs for case 2 is same as for case 1 with the exception of 0-24 hour X/Q = 1.93E-3 sec/m<sup>3</sup>

## RESULTS

The 0 - 720 hour Unit 1 Control Room Doses are as follows

	UNIT 1 CR DOSES, REM	GDC 19 LIMIT, REM
THYROID	4.50E-01	30
GAMMA	7.10E-04	5
BETA	9.21E-03	30

	UNIT 1 CR DOSES, REM	GDC 19 LIMIT, REM
THYROID	2.76E+01	30
GAMMA	1.44E-02	5
BETA	3.37E-01	30

## CONCLUSIONS

The revision to control room volume and control room ventilation intake flow rate had negligible impact on the control room operator doses. The control room doses resulting from both a ground level and elevated release are within the dose guidelines of 10CFR50 Appendix A GDC 19 assuming no control room air treatment system initiation. Therefore, the control room air treatment system is not required to meet the guideline doses.

## COMPUTER RUN LOG

<u>JOB #</u>	<u>DATE</u>	<u>DESCRIPTION OF RUN</u>
9014	5/16/98	DRAGON (REF 10) CRDA to U1 Control Room-no filters,
		elevated release
8946	5/16/98	DRAGON (REF 10) Same as above except ground
		level release.

Card images (2) are given in Appendix A

## REFERENCES

Nine Mile Point 1 Final Safety Analysis Report Revision 14

Table XV-9

I.B.4.0

XV.C.3.2

XV.C.4.5.1

III.B.2.2

General Electric Standard Application for Reactor Fuel, Licensing Topical Report, NEDE-24011-P-A-13 Class III, August 1996.

Engineering Report for Application of GE11 to Nine Mile Point Nuclear Station Unit 1 Reload 12, GENE-770-31-1292, revision 2, April, 1993

NUREG 0800, Standard Review Plan 15.4.9, Radiological Consequences of Control Rod Drop Accident (BWR).

Regulatory Guide 1.77, Assumptions used for evaluating a control rod ejection accident for pressurized water reactors, Appendix B.1.b, 1.c, and 2.c

Technical Specifications section 1.14, Amendment 142, page 5

Calculation 1H-009, Control Rod Drop Accident, revision 00

S10-210-HV12, Control Room & Auxiliary Control Room, revision 00, pages 45 and attachment I-3 and NMPC Drawings: C18810C, sheet 1; C18812C, sheet 1; C18804C, sheet 1.

Letter dated March 19, 1984, from T. E. Lempges (NMPC) to D. B. Vassallo (NRC)

DRAGON computer code, SWEC Number NU-115, Version 5, Level 0

NMPC Calculation S10-CR277.A-U1.210.

SRP 15.6.5, Loss-of Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary, and Regulatory Guide 1.3, Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors.

SRP 6.4, Control Room Habitability System

Internal Correspondence, file code M98-014, from T. Mogren to T. Kulczycky dated 5/18/98, "Outside Air Flow Rate for Control Room Ventilation."

APPENDIX A ( 2 ATTACHED)  
CARD IMAGES OF COMPUTER RUNS

From: <kulczyckyt@nimo.com>  
To: WND2.WNP3 (DSH)  
Date: 5/5/98 7:23am  
Subject: NMP Unit 1 Control Rod Drop Accident

Darl,

My original e-mail did not go through, apparently because of the size of the documents. Therefore I will send them to you one at a time.

Attached is one of the 4 calculations for the Unit 1 Control Room Air Treatment system you requested yesterday during our telecon.

Unit 1 Control Rod Drop Accident  
(See attached file: ulcrda.doc)

The documents are in WORD for Windows version 7.0 format - if unable to read please call. My number is 315-349-1949 or try my pager 1-800-732-4365, pager # 1072.

Ted Kulczycky



## TABLE OF CONTENTS

	<u>page #</u>
Title Page	1
Table of Contents	2
Purpose	3
Background	3
Methodology	3
Data / Assumptions	4
Calculation	5
Results	8
Conclusion	8
Computer Output Log	8
References	9
APPENDIX A - Card Image of Computer Run	10

## **PURPOSE**

The purpose of this calculation is to determine if the control room air treatment system is required during a control rod drop accident (CRDA) to maintain control room doses within 10CFR50 Appendix A, GDC 19 acceptance criteria.

## **BACKGROUND**

In March, 1984 NMP1 submitted to the staff the results of their control room habitability study in response to NUREG 0737 TMI Task Action Item III.D.3.4, Control Habitability Requirements. The design basis accidents identified in the submittal were the main steam line break and the loss of coolant accidents. The staff required that the licensee use Standard Review Plan (SRP) 6.4, Control Room Habitability System, as one of the documents to determine if the control room habitability acceptance criteria was met. SRP 6.4, section II.6, acceptance criteria states that "In accordance with GDC 19 (Ref. 3), these doses (5 rem or equivalent) to an individual in the control room should not be exceeded for any postulated design basis accident." Furthermore, SRP 15.4.9, Radiological Consequences of Control Rod Drop Accident (BWR), requires a specific evaluation of the CRDA for the first application involving a particular standardized design to establish a reference point for comparison of future applications incorporating the design. The Safety Evaluation Report (SER) received from the staff in May, 1984 stated "... The staff's conclusion is that control room operator doses following design basis accidents would be within GDC-19 guidelines." As stated in the UFSAR section III.B.1.5, Shielding and Access Control, "The most limiting accidents are the main steam line (MSL) break accident and the loss-of-coolant accident (LOCA) without core spray..." However, supporting documentation could not be found that supported the assumption that the control rod drop and fuel handling (FHA) accident were also evaluated as part of the study. As stated in the PURPOSE section, this calculation is for CRDA only. The FHA has been evaluated in calculation H21C045.

Although it has not been determined if the CRDA was required to be evaluated as part of the Task Action Item. This calculation has been performed to determine if the control room air treatment system would be required to mitigate the radiological consequences in the control room to ensure an individual in the control room will not received greater that 5 rem whole body dose or equivalent for any postulated design basis accident.

## **METHODOLOGY**

The resultant halogen and noble gas radioactivity described in the UFSAR assumed to be released to the coolant as a result of a CRDA is compared to a calculated reactor coolant source term based on the number of fuel rods that are assumed to fail as a result of a CRDA. The higher of the two coolant source terms is input to the Stone and Webster computer code DRAGON for conservatism. The method used for release assumptions from the coolant to the environment are consistent with the Standard Review Plan 15.4.9 (REF 4) with the exception that an elevated release is assumed to be consistent with the

UFSAR (REF 1.d).

## DATA/ASSUMPTIONS

Nominal Reactor Power is 1850 MWt. A 2 % uncertainty is added. Therefore the Reactor Power used in this calculation is 1887.

850 fuel rods reach 170 cal/gm, which is the enthalpy limit for eventual cladding perforation. Since U1's peak enthalpy will be less than 280 cal/gm, melting does not occur.

Fuel assumptions.

532 fuel assemblies.

62 rods/fuel assembly.

32,632 total rods. This number was taken from radiological calculation 1H-009, Control Rod Drop Accident. The total rods were based on the actual fuel type in the Unit 1 core in 1991. Using only 8x8 fuel the total number of rods would be 62 by 532 = 32,984. This would result in an approximate 1 % decrease in source term (850/32,984 versus 850/32,632). Therefore, the dose contribution using the 1991 total number of rods are conservative and also negligible.

Elevated release (see METHODOLOGY section)

Release fractions

Amount of fuel gap activity is 10% of total activity in the rod

Fuel rods presumed failed are assumed to have operated at 1.5 times that of the average power level in the core.

10 % of the gap activity for noble gas and iodines instantaneously and uniformly mix in reactor coolant in the pressure vessel at the time of the accident. Although Regulatory Guide 1.25 was used to define the gap activity released to the coolant, only 10% of the krypton 85 activity was assumed to be released instead of the 30% stated in RG1.25. Since Kr-85 had a negligible impact on dose. The computer run was not repeated.

10% of the iodines and 100% of the noble gases are released from the pressure vessel to the turbine.

All noble gases in the turbine are available for release.

90% of the iodines are assumed to be removed by plateout and partitioning in the turbine and the condenser leaving only 10 % airborne and available for leakage.

The turbine and condensers leak to the atmosphere at 1%/day for a period of 24 hours, at which time the leak terminates.

Main Condenser volume: 5.00E+04 ft<sup>3</sup>

Control Room free volume : 1.36E+05 ft<sup>3</sup>. The total volume, 1.69E+05 ft<sup>3</sup>, is multiplied by 0.8 to account for equipment located in the control room envelope (main control room, auxiliary control room, instrument shop, and computer room).

Control Room normal ventilation intake flow rate: 3550 cfm + 30 cfm in-leakage.

This is calculated by taking the maximum flow rate of 16,300 cfm minus the minimum recirculation flow rate of 12,750 cfm (REF 1.e). The additional 30 cfm is to account for 10 cfm inleakage to the control room assumed in accordance with SRP 6.4, section III.3.d.(2).(ii). An additional 20 cfm is

assumed to account for an unfiltered inleakage.  
 Credit is not taken for Control Room Air Treatment System initiation.  
 U1 stack 0-2 hour X/Q is 3.12E-04 sec/m<sup>3</sup> and the 2-720 hour X/Q is 1.22E-08 sec/m<sup>3</sup>.

Radioactivity assumed in the coolant as a result of a postulated CRDA is  
 Halogens = 5.62E+04 Ci and Noble Gas = 6.64E+04 Ci.

Breathing rate of 3.47E-04 m<sup>3</sup> / sec for the duration (0-720 hours) of the accident.

10CFR50 Appendix A, GDC 19 dose limit of 5 rem whole body or equivalent.

This equates to 30 rem thyroid and 30 rem beta (skin).

## CALCULATION

### UNIT 1 COOLANT ACTIVITY

ISOTOPE	Ci/MWt ①	MWt ②	TABLE 1		
			TOTAL CORE ACTIVITY ③ Ci	ACTIVITY IN COOLANT FROM RODS ④ Ci	ACTIVITY RELEASED TO THE CONDENSER ⑤ Ci
I-131	2.90E+04	1887	5.47E+07	7.09E+03	7.09E+02
I-132	4.20E+04	1887	7.93E+07	1.03E+04	1.03E+03
I-133	4.80E+04	1887	9.06E+07	1.17E+04	1.17E+03
I-134	6.20E+04	1887	1.17E+08	1.51E+04	1.51E+03
I-135	4.90E+04	1887	9.25E+07	1.20E+04	1.20E+03
		<b>TOTAL</b>	<b>4.34E+08</b>	<b>5.62E+04</b>	<b>5.62E+03</b>
KR-83M	3.00E+03	1887	5.66E+06	1.07E+03	1.07E+03
KR-85M	6.50E+03	1887	1.23E+07	2.33E+03	2.33E+03
KR-85	3.00E+02	1887	5.66E+05	1.07E+02	1.07E+02
KR-87	1.20E+04	1887	2.26E+07	4.30E+03	4.30E+03
KR-88	1.70E+04	1887	3.21E+07	6.09E+03	6.09E+03
KR-89	2.00E+04	1887	3.77E+07	7.16E+03	7.16E+03
XE-131M	1.80E+02	1887	3.40E+05	6.45E+01	6.45E+01
XE-133M	2.00E+02	1887	3.77E+05	7.16E+01	7.16E+01
XE-133	5.60E+04	1887	1.06E+08	2.01E+04	2.01E+04
XE-135M	1.70E+04	1887	3.21E+07	6.09E+03	6.09E+03
XE-135	9.80E+03	1887	1.85E+07	3.51E+03	3.51E+03
XE-138	4.40E+04	1887	8.30E+07	1.58E+04	1.58E+04
		<b>TOTALS</b>	<b>3.51E+08</b>	<b>6.66E+04</b>	

① GE BWR 6, Technical Description of a single cycle Boiling Water Reactor Nuclear System, January 1, 1974 (Table F.2.3-9)

② 1850 \* 1.02 (DATA/ASSUMPTION # 11)

③ Column ① \* Column ②

④ Activity in coolant from rods = (Column ③ / Total Core Activity) \* Total Coolant Activity

⑤ Column ④ \* 0.1 for Iodines and 1.0 for Noble Gases

## 2.0 COOLANT SOURCE TERM ASSUMING 850 FAILED RODS

ISOTOPE	Ci/MWt ①	TABLE 2		TOTAL CORE ACTIVITY ③ Ci	ACTIVITY IN COOLANT ④ Ci
		MWt ②			
I-131	2.90E+04	1887		5.47E+07	2.14E+05
I-132	4.20E+04	1887		7.93E+07	3.10E+05
I-133	4.80E+04	1887		9.06E+07	3.54E+05
I-134	6.20E+04	1887		1.17E+08	4.57E+05
I-135	4.90E+04	1887		9.25E+07	3.61E+05
		<b>TOTAL</b>		<b>4.34E+08</b>	
KR-83M	3.00E+03	1887		5.66E+06	2.21E+04
KR-85M	6.50E+03	1887		1.23E+07	4.79E+04
KR-85	3.00E+02	1887		5.66E+05	2.21E+03
KR-87	1.20E+04	1887		2.26E+07	8.85E+04
KR-88	1.70E+04	1887		3.21E+07	1.25E+05
KR-89	2.00E+04	1887		3.77E+07	1.47E+05
XE-131M	1.80E+02	1887		3.40E+05	1.33E+03
XE-133M	2.00E+02	1887		3.77E+05	1.47E+03
XE-133	5.60E+04	1887		1.06E+08	4.13E+05
XE-135M	1.70E+04	1887		3.21E+07	1.25E+05
XE-135	9.80E+03	1887		1.85E+07	7.23E+04
XE-138	4.40E+04	1887		8.30E+07	3.24E+05
				<b>3.51E+08</b>	

① GE BWR 6, Technical Description of a single cycle Boiling Water Reactor Nuclear System, January 1, 1974 (Table F.2.3-9)

② 1850 MWt \* 1.02 (DATA/ASSUMPTION # 1)

③ Column ① \* Column ②

④ Column ③ \* (850 failed rods / 32,632 total rods)\*1.5 peaking factor \* 0.1

As can be seen by comparing Table 1 Column 4 to Table 1 Column 4, the activity assumed in the coolant as a result of 850 failed fuel rods is greater than the activity in the coolant using the UFSAR coolant source term. Therefore the activity from 850 failed fuel rods will be for this calculation.

### ACTIVITY AVAILABLE FOR RELEASE

As stated in DATA/ASUMPTIONS 5.d all the noble gases and 10% of the iodines are released to the turbine and condenser. Therefore, Table 2 Column ④ is multiplied by 0.1 for iodines and 1.0 for noble gases to

determine activity in the condenser.

ISOTOPE	ACTIVITY TO CONDENSER
I-131	2.14E+04
I-132	3.10E+04
I-133	3.54E+04
I-134	4.57E+04
I-135	3.61E+04
KR-83M	2.21E+04
KR-85M	4.79E+04
KR-85	2.21E+03
KR-87	8.85E+04
KR-88	1.25E+05
KR-89	1.47E+05
XE-131M	1.33E+03
XE-133M	1.47E+03
XE-133	4.13E+05
XE-135M	1.25E+05
XE-135	1.25E+05
XE-138	3.24E+05

The above condenser activity is input to DRAGON run # 5341 dated 4/29/98. The card input to this run is included in Appendix A.

DRAGON inputs:

Main Condenser volume: 5.00E+04 ft<sup>3</sup> (DATA/ASSUMPTION # 6)  
Main Condenser release rate: 0.01 fractions per day (DATA/ASSUMPTION #5.g)  
Unit 1 Control Room (DATA/ASSUMPTIONS # 7 - 9)  
    volume: 1.36 E+05 ft<sup>3</sup>  
    intake rate 0 to 720 hours: 3580 cfm  
    filter efficiencies 0 - 720 hours = 0  
Breathing rate 0 - 720 hours is 3.47E-04 m<sup>3</sup> /sec (DATA/ASSUMPTION #12)  
0-2 hour X/Q: 3.12E-04 sec/m<sup>3</sup> 2 - 24 hour X/Q: 1.22E-08 sec/m<sup>3</sup>  
(DATA/ASSUMPTION #10)

## RESULTS

The 0 - 720 hour Unit 1 Control Room Doses are as follows

TABLE 4  
CRDA TO U1 CONTROL ROOM

	UNIT 1 CR DOSES, REM	GDC 19 LIMIT, REM
THYROID	4.52E-01	30
GAMMA	7.65E-04	5
BETA	9.68E-03	30

### CONCLUSIONS

The control room doses are within the dose guidelines of 10CFR50 Appendix A GDC 19 assuming no control room air treatment system initiation. Therefore, the control room air treatment system is not required to meet the guideline doses.

### COMPUTER RUN LOG

<u>JOB #</u>	<u>DATE</u>	<u>DESCRIPTION OF RUN</u>
5341	4/29/98	DRAGON (REF 10) CRDA to U1 Control

Room-no filters  
Card image is given in Appendix A

## REFERENCES

Nine Mile Point 1 Final Safety Analysis Report Revision 14

Table XV-9

I.B.4.0

XV.C.3.2

XV.C.4.5.1

III.B.2.2

General Electric Standard Application for Reactor Fuel, Licensing Topical Report, NEDE-24011-P-A-13 Class III, August 1996.

Engineering Report for Application of GE11 to Nine Mile Point Nuclear Station Unit 1 Reload 12, GENE-770-31-1292, revision 2, April, 1993

NUREG 0800, Standard Review Plan 15.4.9, Radiological Consequences of Control Rod Drop Accident (BWR).

Regulatory Guide 1.25, Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility For A Boiling and Pressurized Water Reactor.

Technical Specifications section 1.14, Amendment 142, page 5

Calculation 1H-009, Control Rod Drop Accident, revision 00

S10-210-HV12, Control Room & Auxiliary Control Room, revision 00

Letter dated March 19, 1984, from T. E. Lempges (NMPC) to D. B. Vassallo (NRC)

DRAGON computer code, SWEC Number NU-115, Version 5, Level 0

NMPC Calculation S10-CR277.A-U1.210.



APPENDIX A ( 1 ATTACHED)  
CARD IMAGE OF COMPUTER RUNS

<b>NY NIAGARA MOHAWK</b> NUCLEAR ENGINEERING	<b>CALCULATION COVER SHEET</b>	Page 1 ( Next <input type="text"/> ) Total 11 Last 11
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NINE MILE POINT NUCLEAR STATION      Unit (1, 2 or 0=Both): 1    Discipline: ANALYSIS

Title <b>CONTROL ROD DROP ACCIDENT (CRDA) IMPACT ON CONTROL ROOM HABITABILITY</b>	Calculation No. <b>H21C046</b>		
(Sub)system(s) N/A	Building TB	Floor Elev. 277	Index No. N/A

Originator(s) T. M. KURTZ
Checker(s) / Approver(s) A. C. MOISAN / T. G. KULCZYCKY

Rev	Description	Design/Config Change No.	Prep'd By	Date	Chk	Date	App	Date
00	ORIGINAL	NA	TAK	5/1/98	COM	5/1/98	TGE	5/1/98

Computer Output/Microfilm Filed Separately (Yes / No / NA): Yes      Safety Class (SR / NSR / Qxx): SR

Superseded Document(s): NONE

Document Cross Reference(s) - For additional references see page(s) :9

Ref No	Document No.	Doc Type	Index	Sheet	Rev
	SEE PAGE 9				

General Reference(s):  
 See page 9

Remarks:  
 Calculation to determine Unit 1 Control Room doses from a design basis Unit 1 Control Rod Drop Accident (CRDA). This calculation determines that no Unit 1 Control Room Air Treatment System Initiation is required to reduce doses below GDC 19 limits.

Confirmation Required (Yes / No) : No See Page(s) : _____	Final Issue Status ( APP / FIO / VOI ) : APP	File Location ( Calc / Hold ) : Calc	Operations Acceptance Required ( Yes / No ) : No
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Evaluation Number(s): SE 98-010 Copy of Applicability Review Attached (Yes / N/R)?N/R	Component ID(s) (As shown in MEL) : NONE
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Key Words : DOSE, GDC 19, CONTROL ROOM HABITABILITY, CRDA

Post-It™ brand fax transmittal memo 7671 # of pages 1

To: D. HOOD	From: P. TRACA
Co.:	Co.:
Dept.:	Phone: 315-349-1322
Fax #: 315-415-2182	Fax #:

	<b>CALCULATION CONTINUATION SHEET</b>	Page 2 (Next _____)
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Nine Mile Point Nuclear Station

Unit:   1  

Disposition:   N/A  

Originator/Date T. M. Kurtz / 4/29/98	Checker/Date A. C. Moisan / 5/1/98	Calculation No. H21C046	Revision 00
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Ref.	<b>TABLE OF CONTENTS</b>
	<u>page #</u>
	1
	2
	3
	3
	3
	4
	5
	8
	8
	8
	9
	10

	<b>CALCULATION CONTINUATION SHEET</b>	Page 3 (Next <u>      </u> )
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Nine Mile Point Nuclear Station

Unit:   1  Disposition:   N/A  

Originator/Date T. M. Kurtz / 4/29/98	Checker/Date A. C. Moisan / 5/1/98	Calculation No. H21C046	Revision 00
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Ref.

### PURPOSE

The purpose of this calculation is to determine if the control room air treatment system is required during a control rod drop accident (CRDA) to maintain control room doses within 10CFR50 Appendix A, GDC 19 acceptance criteria.

### BACKGROUND

In March, 1984 NMP1 submitted to the staff the results of their control room habitability study in response to NUREG 0737 TMI Task Action Item III.D.3.4, Control Habitability Requirements. The design basis accidents identified in the submittal were the main steam line break and the loss of coolant accidents. The staff required that the licensee use Standard Review Plan (SRP) 6.4, Control Room Habitability System, as one of the documents to determine if the control room habitability acceptance criteria was met. SRP 6.4, section II.6, acceptance criteria states that "In accordance with GDC 19 (Ref. 3), these doses (5 rem or equivalent) to an individual in the control room should not be exceeded for any postulated design basis accident." Furthermore, SRP 15.4.9, Radiological Consequences of Control Rod Drop Accident (BWR), requires a specific evaluation of the CRDA for the first application involving a particular standardized design to establish a reference point for comparison of future applications incorporating the design. The Safety Evaluation Report (SER) received from the staff in May, 1984 stated "... The staff's conclusion is that control room operator doses following design basis accidents would be within GDC-19 guidelines." As stated in the UFSAR section III.B.1.5, Shielding and Access Control, "The most limiting accidents are the main steam line (MSL) break accident and the loss-of-coolant accident (LOCA) without core spray..." However, supporting documentation could not be found that supported the assumption that the control rod drop and fuel handling (FHA) accident were also evaluated as part of the study. As stated in the PURPOSE section, this calculation is for CRDA only. The FHA has been evaluated in calculation H21C045.

Although it has not been determined if the CRDA was required to be evaluated as part of the Task Action Item. This calculation has been performed to determine if the control room air treatment system would be required to mitigate the radiological consequences in the control room to ensure an individual in the control room will not received greater that 5 rem whole body dose or equivalent for any postulated design basis accident.

### METHODOLOGY

The resultant halogen and noble gas radioactivity described in the UFSAR assumed to be released to the coolant as a result of a CRDA is compared to a calculated reactor coolant source term based on the number of fuel rods that are assumed to fail as a result of a CRDA. The higher of the two coolant source terms is input to the Stone and Webster computer code DRAGON for conservatism. The method used for release assumptions from the coolant to the environment are consistent with the Standard Review Plan 15.4.9 (REF 4) with the exception that an elevated release is assumed to be consistent with the UFSAR (REF 1.d).

	<h2 style="margin: 0;">CALCULATION CONTINUATION SHEET</h2>	Page 4 (Next _____)
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Nine Mile Point Nuclear Station

Unit:   1  

Disposition:   N/A  

Originator/Date T. M. Kurtz / 4/29/98	Checker/Date A. C. Moisan / 5/1/98	Calculation No. H21C046	Revision 00
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Ref.	DATA/ASSUMPTIONS
1.a 6 2,3 4	1. Nominal Reactor Power is 1850 MWt. A 2 % uncertainty is added. Therefore the Reactor Power used in this calculation is 1887. 2. 850 fuel rods reach 170 cal/gm, which is the enthalpy limit for eventual cladding perforation. Since U1's peak enthalpy will be less than 280 cal/gm, melting does not occur.
1.b 1.c 7	3. Fuel assumptions. <ul style="list-style-type: none"> <li>a. 532 fuel assemblies.</li> <li>b. 62 rods/fuel assembly.</li> <li>c. 32,632 total rods. This number was taken from radiological calculation 1H-009, Control Rod Drop Accident. The total rods were based on the actual fuel type in the Unit 1 core in 1991. Using only 8x8 fuel the total number of rods would be 62 by 532 = 32,984. This would result in an approximate 1 % decrease in source term (850/32,984 versus 850/32,632). Therefore, the dose contribution using the 1991 total number of rods are conservative and also negligible.</li> </ul>
1.d 4	4. Elevated release (see METHODOLOGY section). 5. Release fractions <ul style="list-style-type: none"> <li>a. Amount of fuel gap activity is 10% of total activity in the rod</li> <li>b. Fuel rods presumed failed are assumed to have operated at 1.5 times that of the average power level in the core.</li> <li>c. 10 % of the gap activity for noble gas and iodines instantaneously and uniformly mix in reactor coolant in the pressure vessel at the time of the accident. Although Regulatory Guide 1.25 was used to define the gap activity released to the coolant, only 10% of the krypton 85 activity was assumed to be released instead of the 30% stated in RG1.25. Since Kr-85 had a negligible impact on dose. The computer run was not repeated.</li> <li>d. 10% of the iodines and 100% of the noble gases are released from the pressure vessel to the turbine.</li> <li>e. All noble gases in the turbine are available for release.</li> <li>f. 90% of the iodines are assumed to be removed by plateout and partitioning in the turbine and the condenser leaving only 10 % airborne and available for leakage.</li> <li>g. The turbine and condensers leak to the atmosphere at 1%/day for a period of 24 hours, at which time the leak terminates.</li> </ul>
7 8	6. Main Condenser volume: 5.00E+04 ft <sup>3</sup> 7. Control Room free volume : 1.36E+05 ft <sup>3</sup> . The total volume, 1.69E+05 ft <sup>3</sup> , is multiplied by 0.8 to account for equipment located in the control room envelope (main control room, auxiliary control room, instrument shop, and computer room).
1.e 11	8. Control Room normal ventilation intake flow rate: 3550 cfm + 30 cfm in-leakage. This is calculated by taking the maximum flow rate of 16,300 cfm minus the minimum recirculation flow rate of 12,750 cfm (REF 1.e). The additional 30 cfm is to account for 10 cfm inleakage to the control room assumed in accordance with SRP 6.4, section III.3.d.(2).(ii). An additional 20 cfm is assumed to account for an unfiltered inleakage.



**CALCULATION CONTINUATION SHEET**

Nine Mile Point Nuclear Station

Unit: 1

Disposition: N/A

Originator/Date T. M. Kurtz / 4/29/98	Checker/Date A. C. Moisan / 5/1/98	Calculation No. H21C046	Revision 00
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Ref.  
9  
1-a  
10

- 9. Credit is not taken for Control Room Air Treatment System initiation.
- 10. U1 stack 0-2 hour X/Q is 3.12E-04 sec/m<sup>3</sup> and the 2-720 hour X/Q is 1.22E-08 sec/m<sup>3</sup>.
- 11. Radioactivity assumed in the coolant as a result of a postulated CRDA is  
Halogens = 5.62E+04 Ci and Noble Gas = 6.64E+04 Ci.
- 12. Breathing rate of 3.47E-04 m<sup>3</sup> / sec for the duration (0-720 hours) of the accident.
- 13. 10CFR50 Appendix A, GDC 19 dose limit of 5 rem whole body or equivalent. This equates to 30 rem thyroid and 30 rem beta (skin).

**CALCULATION**

**1.0 UNIT 1 COOLANT ACTIVITY**

ISOTOPE	Ci/MWt ①	MWt ②	TABLE 1		
			TOTAL CORE ACTIVITY ③ Ci	ACTIVITY IN COOLANT FROM RODS ④ Ci	ACTIVITY RELEASED TO THE CONDENSER ⑤ Ci
I-131	2.90E+04	1887	5.47E+07	7.09E+03	7.09E+02
I-132	4.20E+04	1887	7.93E+07	1.03E+04	1.03E+03
I-133	4.80E+04	1887	9.06E+07	1.17E+04	1.17E+03
I-134	6.20E+04	1887	1.17E+08	1.51E+04	1.51E+03
I-135	4.90E+04	1887	9.25E+07	1.20E+04	1.20E+03
		TOTAL	4.34E+08	5.62E+04	5.62E+03
KR-83M	3.00E+03	1887	5.66E+06	1.07E+03	1.07E+03
KR-85M	6.50E+03	1887	1.23E+07	2.33E+03	2.33E+03
KR-85	3.00E+02	1887	5.66E+05	1.07E+02	1.07E+02
KR-87	1.20E+04	1887	2.26E+07	4.30E+03	4.30E+03
KR-88	1.70E+04	1887	3.21E+07	6.09E+03	6.09E+03
KR-89	2.00E+04	1887	3.77E+07	7.16E+03	7.16E+03
XE-131M	1.80E+02	1887	3.40E+05	6.45E+01	6.45E+01
XE-133M	2.00E+02	1887	3.77E+05	7.16E+01	7.16E+01
XE-133	5.60E+04	1887	1.06E+08	2.01E+04	2.01E+04
XE-135M	1.70E+04	1887	3.21E+07	6.09E+03	6.09E+03
XE-135	9.80E+03	1887	1.85E+07	3.51E+03	3.51E+03
XE-138	4.40E+04	1887	8.30E+07	1.58E+04	1.58E+04
		TOTALS	3.51E+08	6.66E+04	

① GE BWR 6, Technical Description of a single cycle Boiling Water Reactor Nuclear System, January 1, 1974 (Table F.2.3-9)

② 1850 \* 1.02 (DATA/ASSUMPTION # 11)

③ Column ① \* Column ②

④ Activity in coolant from rods = (Column ③ / Total Core Activity) \* Total Coolant Activity

⑤ Column ④ \* 0.1 for Iodines and 1.0 for Noble Gases

Nine Mile Point Nuclear Station

Unit:   1  

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Originator/Date T. M. Kurtz / 4/29/98	Checker/Date A. C. Moisan / 5/1/98	Calculation No. H21C046	Revision 00
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Ref.	<p style="text-align: center;"><b>2.0 COOLANT SOURCE TERM ASSUMING 850 FAILED RODS</b></p> <p style="text-align: center;"><b>TABLE 2</b></p> <table style="width: 100%; border-collapse: collapse;"> <thead> <tr> <th style="text-align: left;">ISOTOPE</th> <th style="text-align: center;">Ci/MWt ①</th> <th style="text-align: center;">MWt ②</th> <th style="text-align: center;">TOTAL CORE ACTIVITY ③ Ci</th> <th style="text-align: center;">ACTIVITY IN COOLANT ④ Ci</th> </tr> </thead> <tbody> <tr><td>I-131</td><td style="text-align: right;">2.90E+04</td><td style="text-align: center;">1887</td><td style="text-align: right;">5.47E+07</td><td style="text-align: right;">2.14E+05</td></tr> <tr><td>I-132</td><td style="text-align: right;">4.20E+04</td><td style="text-align: center;">1887</td><td style="text-align: right;">7.93E+07</td><td style="text-align: right;">3.10E+05</td></tr> <tr><td>I-133</td><td style="text-align: right;">4.80E+04</td><td style="text-align: center;">1887</td><td style="text-align: right;">9.06E+07</td><td style="text-align: right;">3.54E+05</td></tr> <tr><td>I-134</td><td style="text-align: right;">6.20E+04</td><td style="text-align: center;">1887</td><td style="text-align: right;">1.17E+08</td><td style="text-align: right;">4.57E+05</td></tr> <tr><td>I-135</td><td style="text-align: right;">4.90E+04</td><td style="text-align: center;">1887</td><td style="text-align: right;">9.25E+07</td><td style="text-align: right;">3.61E+05</td></tr> <tr><td colspan="3" style="text-align: right;"><b>TOTAL</b></td><td style="text-align: right;"><b>4.34E+08</b></td><td></td></tr> <tr><td colspan="5"> </td></tr> <tr><td>KR-83M</td><td style="text-align: right;">3.00E+03</td><td style="text-align: center;">1887</td><td style="text-align: right;">5.66E+06</td><td style="text-align: right;">2.21E+04</td></tr> <tr><td>KR-85M</td><td style="text-align: right;">6.50E+03</td><td style="text-align: center;">1887</td><td style="text-align: right;">1.23E+07</td><td style="text-align: right;">4.79E+04</td></tr> <tr><td>KR-85</td><td style="text-align: right;">3.00E+02</td><td style="text-align: center;">1887</td><td style="text-align: right;">5.66E+05</td><td style="text-align: right;">2.21E+03</td></tr> <tr><td>KR-87</td><td style="text-align: right;">1.20E+04</td><td style="text-align: center;">1887</td><td style="text-align: right;">2.26E+07</td><td style="text-align: right;">8.85E+04</td></tr> <tr><td>KR-88</td><td style="text-align: right;">1.70E+04</td><td style="text-align: center;">1887</td><td style="text-align: right;">3.21E+07</td><td style="text-align: right;">1.25E+05</td></tr> <tr><td>KR-89</td><td style="text-align: right;">2.00E+04</td><td style="text-align: center;">1887</td><td style="text-align: right;">3.77E+07</td><td style="text-align: right;">1.47E+05</td></tr> <tr><td colspan="5"> </td></tr> <tr><td>XE-131M</td><td style="text-align: right;">1.80E+02</td><td style="text-align: center;">1887</td><td style="text-align: right;">3.40E+05</td><td style="text-align: right;">1.33E+03</td></tr> <tr><td>XE-133M</td><td style="text-align: right;">2.00E+02</td><td style="text-align: center;">1887</td><td style="text-align: right;">3.77E+05</td><td style="text-align: right;">1.47E+03</td></tr> <tr><td>XE-133</td><td style="text-align: right;">5.60E+04</td><td style="text-align: center;">1887</td><td style="text-align: right;">1.06E+08</td><td style="text-align: right;">4.13E+05</td></tr> <tr><td>XE-135M</td><td style="text-align: right;">1.70E+04</td><td style="text-align: center;">1887</td><td style="text-align: right;">3.21E+07</td><td style="text-align: right;">1.25E+05</td></tr> <tr><td>XE-135</td><td style="text-align: right;">9.80E+03</td><td style="text-align: center;">1887</td><td style="text-align: right;">1.85E+07</td><td style="text-align: right;">7.23E+04</td></tr> <tr><td>XE-138</td><td style="text-align: right;">4.40E+04</td><td style="text-align: center;">1887</td><td style="text-align: right;">8.30E+07</td><td style="text-align: right;">3.24E+05</td></tr> <tr><td colspan="3"></td><td style="text-align: right;">3.51E+08</td><td></td></tr> </tbody> </table> <p>① GE BWR 6, Technical Description of a single cycle Boiling Water Reactor Nuclear System, January 1, 1974 (Table F.2.3-9)</p> <p>② 1850 MWt * 1.02 (DATA/ASSUMPTION # 1)</p> <p>③ Column ① * Column ②</p> <p>④ Column ③ * (850 failed rods / 32,632 total rods) * 1.5 peaking factor * 0.1</p> <p>As can be seen by comparing Table 1 Column 4 to Table 1.Column 4, the activity assumed in the coolant as a result of 850 failed fuel rods is greater that the activity in the coolant using the UFSAR coolant source term. Therefore the activity from 850 failed fuel rods will be for this calculation.</p>	ISOTOPE	Ci/MWt ①	MWt ②	TOTAL CORE ACTIVITY ③ Ci	ACTIVITY IN COOLANT ④ Ci	I-131	2.90E+04	1887	5.47E+07	2.14E+05	I-132	4.20E+04	1887	7.93E+07	3.10E+05	I-133	4.80E+04	1887	9.06E+07	3.54E+05	I-134	6.20E+04	1887	1.17E+08	4.57E+05	I-135	4.90E+04	1887	9.25E+07	3.61E+05	<b>TOTAL</b>			<b>4.34E+08</b>							KR-83M	3.00E+03	1887	5.66E+06	2.21E+04	KR-85M	6.50E+03	1887	1.23E+07	4.79E+04	KR-85	3.00E+02	1887	5.66E+05	2.21E+03	KR-87	1.20E+04	1887	2.26E+07	8.85E+04	KR-88	1.70E+04	1887	3.21E+07	1.25E+05	KR-89	2.00E+04	1887	3.77E+07	1.47E+05						XE-131M	1.80E+02	1887	3.40E+05	1.33E+03	XE-133M	2.00E+02	1887	3.77E+05	1.47E+03	XE-133	5.60E+04	1887	1.06E+08	4.13E+05	XE-135M	1.70E+04	1887	3.21E+07	1.25E+05	XE-135	9.80E+03	1887	1.85E+07	7.23E+04	XE-138	4.40E+04	1887	8.30E+07	3.24E+05				3.51E+08	
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	<b>CALCULATION CONTINUATION SHEET</b>	Page 7 (Next _____)
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Nine Mile Point Nuclear Station

Unit:   1  

Disposition:   N/A  

Originator/Date T. M. Kurtz / 4/29/98	Checker/Date A. C. Moisan / 5/1/98	Calculation No. H21C046	Revision 00
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Ref.	<p><b>3.0 ACTIVITY AVAILABLE FOR RELEASE</b></p> <p>As stated in DATA/ASSUMPTIONS 5.d all the noble gases and 10% of the iodines are released to the turbine and condenser. Therefore, Table 2 Column ④ is multiplied by 0.1 for iodines and 1.0 for noble gases to determine activity in the condenser.</p> <p style="text-align: center;"><b>TABLE 3 ACTIVITY TO CONDENSER</b></p> <table style="margin-left: auto; margin-right: auto;"> <thead> <tr> <th style="text-align: left;">ISOTOPE</th> <th style="text-align: right;">ACTIVITY TO CONDENSER</th> </tr> </thead> <tbody> <tr><td style="text-align: left;">I-131</td><td style="text-align: right;">2.14E+04</td></tr> <tr><td style="text-align: left;">I-132</td><td style="text-align: right;">3.10E+04</td></tr> <tr><td style="text-align: left;">I-133</td><td style="text-align: right;">3.54E+04</td></tr> <tr><td style="text-align: left;">I-134</td><td style="text-align: right;">4.57E+04</td></tr> <tr><td style="text-align: left;">I-135</td><td style="text-align: right;">3.61E+04</td></tr> <tr><td colspan="2"> </td></tr> <tr><td style="text-align: left;">KR-83M</td><td style="text-align: right;">2.21E+04</td></tr> <tr><td style="text-align: left;">KR-85M</td><td style="text-align: right;">4.79E+04</td></tr> <tr><td style="text-align: left;">KR-85</td><td style="text-align: right;">2.21E+03</td></tr> <tr><td style="text-align: left;">KR-87</td><td style="text-align: right;">8.85E+04</td></tr> <tr><td style="text-align: left;">KR-88</td><td style="text-align: right;">1.25E+05</td></tr> <tr><td style="text-align: left;">KR-89</td><td style="text-align: right;">1.47E+05</td></tr> <tr><td colspan="2"> </td></tr> <tr><td style="text-align: left;">XE-131M</td><td style="text-align: right;">1.33E+03</td></tr> <tr><td style="text-align: left;">XE-133M</td><td style="text-align: right;">1.47E+03</td></tr> <tr><td style="text-align: left;">XE-133</td><td style="text-align: right;">4.13E+05</td></tr> <tr><td style="text-align: left;">XE-135M</td><td style="text-align: right;">1.25E+05</td></tr> <tr><td style="text-align: left;">XE-135</td><td style="text-align: right;">1.25E+05</td></tr> <tr><td style="text-align: left;">XE-138</td><td style="text-align: right;">3.24E+05</td></tr> </tbody> </table> <p>The above condenser activity is input to DRAGON run # 5341 dated 4/29/98. The card input to this run is included in Appendix A.</p> <p>DRAGON inputs:</p> <p>Main Condenser volume: 5.00E+04 ft<sup>3</sup> (DATA/ASSUMPTION # 6)                  Main Condenser release rate: 0.01 fractions per day (DATA/ASSUMPTION #5.g)                  Unit 1 Control Room (DATA/ASSUMPTIONS # 7 - 9)</p> <ul style="list-style-type: none"> <li>• volume: 1.36 E+05 ft<sup>3</sup></li> <li>• intake rate 0 to 720 hours: 3580 cfm</li> <li>• filter efficiencies 0 - 720 hours = 0</li> </ul> <p>Breathing rate 0 - 720 hours is 3.47E-04 m<sup>3</sup> /sec (DATA/ASSUMPTION #12)                  0-2 hour X/Q: 3.12E-04 sec/m<sup>3</sup> 2 - 24 hour X/Q: 1.22E-08 sec/m<sup>3</sup> (DATA/ASSUMPTION #10)</p>	ISOTOPE	ACTIVITY TO CONDENSER	I-131	2.14E+04	I-132	3.10E+04	I-133	3.54E+04	I-134	4.57E+04	I-135	3.61E+04			KR-83M	2.21E+04	KR-85M	4.79E+04	KR-85	2.21E+03	KR-87	8.85E+04	KR-88	1.25E+05	KR-89	1.47E+05			XE-131M	1.33E+03	XE-133M	1.47E+03	XE-133	4.13E+05	XE-135M	1.25E+05	XE-135	1.25E+05	XE-138	3.24E+05
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Nine Mile Point Nuclear Station Unit:   1   Disposition:   N/A  

Originator/Date T. M. Kurtz / 4/29/98	Checker/Date A. C. Moisan / 5/1/98	Calculation No. H21C046	Revision 00
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Ref.

**RESULTS**

The 0 - 720 hour Unit 1 Control Room Doses are as follows

<b>TABLE 4</b>		
<b>CRDA TO U1 CONTROL ROOM</b>		
<b>UNIT 1 CR DOSES,</b>	<b>REM</b>	<b>GDC 19 LIMIT,</b>
	<b>REM</b>	<b>REM</b>
THYROID	4.52E-01	30
GAMMA	7.65E-04	5
BETA	9.68E-03	30

**CONCLUSIONS**

The control room doses are within the dose guidelines of 10CFR50 Appendix A GDC 19 assuming no control room air treatment system initiation. Therefore, the control room air treatment system is not required to meet the guideline doses.

**COMPUTER RUN LOG**

<u>JOB #</u>	<u>DATE</u>	<u>DESCRIPTION OF RUN</u>
5341	4/29/98	DRAGON (REF 10) CRDA to U1 Control Room-no filters

Card image is given in Appendix A

	<b>CALCULATION CONTINUATION SHEET</b>	Page 9 (Next <u>    </u> )
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Nine Mile Point Nuclear Station

Unit:   1  Disposition:   N/A  

Originator/Date T. M. Kurtz / 4/29/98	Checker/Date A. C. Moisan / 5/1/98	Calculation No. H21C046	Revision 00
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Ref.
<p style="text-align: center;"><b><u>REFERENCES</u></b></p> <ol style="list-style-type: none"> <li>1. Nine Mile Point 1 Final Safety Analysis Report Revision 14       <ol style="list-style-type: none"> <li>a. Table XV-9</li> <li>b. I.B.4.0</li> <li>c. XV.C.3.2</li> <li>d. XV.C.4.5.1</li> <li>e. III.B.2.2</li> </ol> </li> <li>2. General Electric Standard Application for Reactor Fuel, Licensing Topical Report, NEDE-24011-P-A-13 Class III, August 1996.</li> <li>3. Engineering Report for Application of GE11 to Nine Mile Point Nuclear Station Unit 1 Reload 12, GENE-770-31-1292, revision 2, April, 1993</li> <li>4. NUREG 0800, Standard Review Plan 15.4.9, Radiological Consequences of Control Rod Drop Accident (BWR).</li> <li>5. Regulatory Guide 1.25, Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility For A Boiling and Pressurized Water Reactor.</li> <li>6. Technical Specifications section 1.14, Amendment 142, page 5</li> <li>7. Calculation 1H-009, Control Rod Drop Accident, revision 00</li> <li>8. S10-210-HV12, Control Room &amp; Auxiliary Control Room, revision 00</li> <li>9. Letter dated March 19, 1984, from T. E. Lempges (NMPC) to D. B. Vassallo (NRC)</li> <li>10. DRAGON computer code, SWEC Number NU-115, Version 5, Level 0</li> <li>11. NMPC Calculation S10-CR277.A-U1.210.</li> </ol>

<b>NIAGARA MOHAWK</b> NUCLEAR ENGINEERING	<b>CALCULATION CONTINUATION SHEET</b>	Page 10 (Next _____)
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Nine Mile Point Nuclear Station

Unit: 1

Disposition: N/A

Originator/Date T. M. Kurtz / 4/29/98	Checker/Date A. C. Moisan / 5/1/98	Calculation No. H21C046	Revision 00
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Ref.	APPENDIX A ( 1 ATTACHED) CARD IMAGE OF COMPUTER RUNS
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\*\*\*\*\* CARD IMAGE OF INPUT SUBMITTED TO DRAGON \*\*\*\*\*

CARD COLUMNS

CARD NO.

1  
2  
3  
4  
5  
6  
7  
8  
9  
10

	1	2	3	4	5	6	7	8
1234567890123456789012345678901234567890123456789012345678901234567890								
**NMPI CRDA RECREATED USING SRP MODEL AND 750 OF 32632 PINS PERF								
3 1011 1 1011	0	1	0	0	0.10	0	1	
MAIN CONDENSER		5.0014	0	7	0	0	0.01	0 0 0
CONTROL ROOM		1.3615	0	0	0	0	1.0	0 0 1
	0.0010	2.1414	3.1014	1.7414	4.5714	3.6114	0.0010	0.0010
	0.0010	0.0010	0.0010	2.2114	4.7914	2.2113	8.9514	1.2515
	1.4715	1.3313	1.4713	4.1315	1.2515	7.2314	0.0010	3.2415
1	1	0	1	1	0	3500.	3500.	3.12-4 1 1 2.0
2	1	0	0	0	0	3500.	3500.	1.22-9 1 1 24.0
3	0	0	0	0	0	3590.	3500.	1.22-8 1 1 720.0
1234567890123456789012345678901234567890123456789012345678901234567890								

CARD COLUMNS

DRAGON RUN # 5341 4/23/58

CALC H21C046 P 11  
Rev 00