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 prèsumed 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 gap 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 ft3 (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 ft3. (REF 8)

- The control room normal ventilation intake flow rate is $2250 \text{ cfm} \pm 10\%$. 2250 cfm + 10% = 2475 cfm, use 2500 cfm. (REF 14). An additional 30 cfm in-leakage is to account for 10 cfm inleakage to the control room assumed in accordance with SRP 6.4, section III.3.d.(2).(ii) and 20 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 sec/m3 and the 2-720 hour X/Q is 1.22E-08 sec/m3.
- U1 turbine building blowout panel 0-2 hour X/Q is 1.93E-03 sec/m3. 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 Ci and Noble Gas = 6.64E+04 Ci.(REF 1.d)
- Breathing rate of 3.47E-04 m3 / 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

		TABLE 2		
ISOTOPE	Ci/MWt	MWt	TOTAL CORE	ACTIVITY IN
	1	Ø	ACTIVITY	COOLANT
			3	4
			Ci	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
			TOTAL	1.70E+06
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-131M XE-133M	2.00E+02		3.77E+05	1.33E+03
		1887		
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
			TOTAL	1.37E+06
	Technical Description	on of a single (

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

 ③ 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 ft3 (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 ft3 intake rate 0 to 720 hours: 2530 cfm filter efficiencies 0 - 720 hours = 0 Breathing rate 0 - 720 hours is 3.47E-04 m3 /sec (DATA/ASSUMPTION #12) 0-2 hour X/Q: 3.12E-04 sec/m3 2 - 24 hour X/Q: 1.22E-08 sec/m3 (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/m3

RESULTS

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

	TABLE 4				
CRDA TO U1 CONTROL ROOM - ELEVATED RELEASE					
	UNIT 1 CR DOSES, GDC 19 LIMIT,				
REM REM					
THYROID	4.50E-01	30			
GAMMA	7.10E-04	5			
BETA	9.21E-03	30			

TABLE 5 CRDA TO U1 CONTROL ROOM - GROUND RELEASE				
	UNIT 1 CR DOSES,	GDC 19 LIMIT,		
	REM	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>	DATE	DESCRIPTION OF RUN
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

١

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

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From:	<kulczyckyt@nimo.com></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

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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 th 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 ft3

- Control Room free volume : 1.36E+05 ft3. The total volume, 1.69E+05 ft3, 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/m3 and the 2-720 hour X/Q is 1.22E-08 sec/m3.
- 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 m3 / 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).

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CALCULATION

UNIT 1 COOLANT ACTIVITY

			TABLE 1		
ISOTOPE	Ci/MWt	MWt	TOTAL	ACTIVITY IN	ACTIVITY
	1	2	CORE	COOLANT FROM	RELEASED TO
			ACTIVITY	RODS	THE CONDENSER
			3	4	6
			Ci	Ci	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	
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① GE BWR 6, Technical Description of a single cycle Boiling Water Reactor Nuclear System, January 1, 1974 (Table F.2.3-9)

@ 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 lodines and 1.0 for Noble Gases

		TABLE 2		
ISOTOPE	Ci/MWt	MWt	TOTAL CORE	ACTIVITY IN
	1	Ø	ACTIVITY	COOLANT
			3	4
			Ci	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
		TOTAL	4.34E+08	
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	

2.0 COOLANT SOURCE TERM ASSUMING 850 FAILED RODS

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 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.

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 @ is multiplied by 0.1 for iodines and 1.0 for noble gases to

determine activity in the condenser.

ISOTOPE	TABLE 3 ACTIVITY TO CONDENSER
ISUIDE	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 ft3 (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 ft3 intake rate 0 to 720 hours: 3580 cfm filter efficiencies 0 - 720 hours = 0 Breathing rate 0 - 720 hours is 3.47E-04 m3 /sec (DATA/ASSUMPTION #12) 0-2 hour X/Q: 3.12E-04 sec/m3 2 - 24 hour X/Q: 1.22E-08 sec/m3 (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,	GDC 19 LIMIT,
	REM	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

JOB #DATE53414/29/98Room-no filtersCard image is given in Appendix A

DESCRIPTION OF RUN DRAGON (REF 10) CRDA to U1 Control

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

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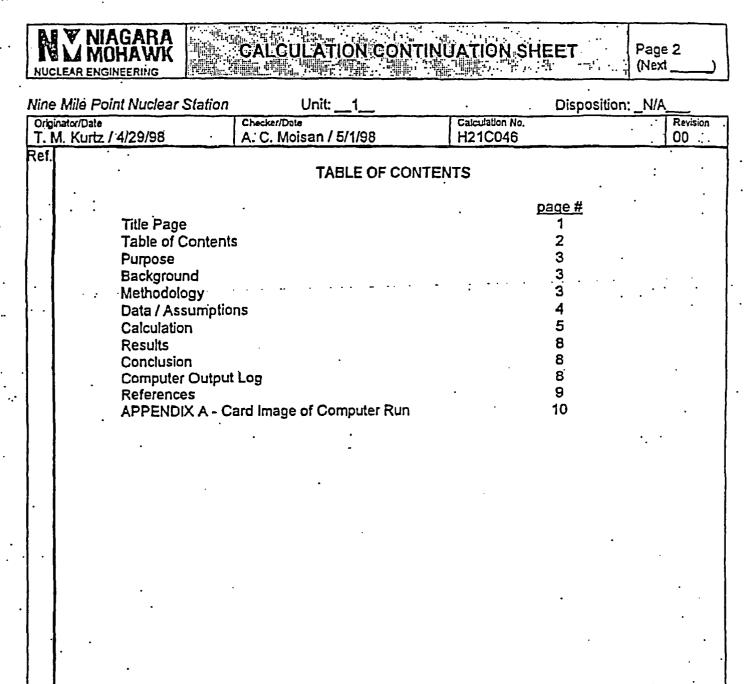
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NUCLE	AR ENGINEERING	CALL(NGO	VERS	HEE	Total	11 11		
	ILE POINT NUCLEAR	STATION	Unit (1, :		: 1 Discip	line: Al	NALYSIS .		•	
CONTRO	OL ROD DROP ACCIDEN	IT (CRDA) IMP	ACT ON CONTROL	Calculat H21C			•			
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Remark Calculat calculat GDC 19	tion to detrmine Unit	1 Control Ro no Unit 1 Cor	om doses from : strol Room Air Tr	a design ba reatment Sy	sis Unit 1 C rstem Initiat	ontrol Roi ion is req	d Drop Acci uired to red	dent (CRD. uce dases	A). This below	
	ation Required (Yes	/ No) : No	Final Issue Stat (APP / FIO / VC		File Locat (Cale / Ho			ons Accept d (Yes / N		
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Originator/Date T. M. Kurtz / 4/29/98	Checker/Date A. C. Moisan / 5/1/98	Calculation No. H21C046	Revision
ef. PURPOSE	••••••••••••••••••••••••••••••••••••••		
. required du	se of this calculation is to determine if uring a control rod drop accident (CRI Appendix A, GDC 19 acceptance crite	DA) to maintain control roor	nent system is m doses within
BACKGRO	UND	· · · · ·	• • •
in response Requireme line break a Review Pla determine i acceptance equivalent) design bas Drop Accid involving a future appli from the sta doses follow UFSAR sec main steam spray" Ho assumption part of the sta	984 NMP1 submitted to the staff the is to NUREG 0737 TMI Task Action Itents. The design basis accidents identiand the loss of coolant accidents. The n (SRP) 8.4, Control Room Habitability f the control room habitability acceptate criteria states that "In accordance we to an individual in the control room shist accident." Furthermore, SRP 15.4.9 ent (BWR), requires a specific evalual particular standardized design to estat cations incorporating the design. The aff in May, 1984 stated " The staff's wing design basis accidents would be tion III.B.1.5, Shielding and Access C line (MSL) break accident and the los wever, supporting documentation cou- that the control rod drop and fuel har study. As stated in the PURPOSE sec- ten evaluated in calculation H21C045	em III.D.3.4, Control Habital ified in the submittal were to staff required that the lice ty System, as one of the d ince criteria was met. SRP ith GDC 19 (Ref. 3), these hould not be exceeded for b, Radiological Consequence tion of the CRDA for the fir ablish a reference point for Safety Evaluation Report (conclusion is that control re within GDC-19 guidelines. control, "The most limiting a ss-of-coolant accident (LOC uld not be found that support adding (FHA) accident were ation, this calculation is for (bility the main steam nsee use Standar ocuments to 6.4, section II.6, doses (5 rem or any postulated ces of Control Roc st application comparison of (SER) received oom operator "As stated in the ccidents are the CA) without core ofted the also evaluated as
Task Action treatment s room to ens	nas not been determined if the CRDA Item. This calculation has been perfort stem would be required to mitigate the ure an individual in the control room v or equivalent for any postulated design	ormed to determine if the co ne radiological consequenc will not received greater that	ontrol room air es in the control

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).

		CALCULATION CON	ITINUATION SHEET	Page 4 (Next
	lile Point Nuclear Stat	lion Unit:1	Disp	osition: _N/A
	or/Dzte Kurtz / 4/29/98	Checker/Date	Calculation No.	Revisi
Ref.	<u>NUILZ 1 4/29/90</u>	A. C. Moisan / 5/1/98	· H21C046	00
	DATAASSUN	IPTIONS		
1.4	1. Nominal R	leactor Power is 1850 MWI. A 2	% uncertainty is added. Th	nerefore the React
6	Power use	ed in this calculation is 1887.		•
2,3	2. 850 fuel ro	ods reach 170 cal/gm, which is I	h enthalpy limit for eventua	I cladding
4	perioration	n. Since U1's peak enthalpy will	be less than 280 cal/gm, m	elting does not
1.5	occur. 3. Fuel assur	mintione		. •
1.c	•	2 fuel assemblies.		
7	•	rods/fuel assembly.	· · · ·	•
	c. 32,	632 total rods. This number wa	s taken from radiological ca	Iculation 1H-009,
	· Ca	ntrol Rod Drop Accident. The to	tal rods were based on the	actual fuel type in
		Unit 1 core in 1991. Using only		
		532 = 32,984. This would result		
		0/32,984 versus 850/32,632). T		tion using the 199
1.d		al number of rods are conservat elease (see METHODOLOGY s		
4	-5. Release fr		ecaony.	
		ount of fuel gap activity is 10%	of total activity in the rod	
	b. Fue	al rods presumed failed are assu	umed to have operated at 1	.5 times that of the
	•	rage power level in the core.	• . •	· · ·
-		% of the gap activity for noble g		
	mix · · · · · · · · · · · · · · · · · · ·	in reactor coolant in the pressu	re vessel at the time of the	accident. Although
		gulatory Guide 1.25 was used to y 10% of the krypton 85 activity		
		6 stated in RG1.25. Since Kr-85		
		nputer run was not repeated.		
		6 of the iodines and 100% of the	e noble gases are released	from the pressure
		sel to the turbine.	-	•
.		noble gases in the turbine are a		
:] .		6 of the iodines are assumed to		
		ine and the condenser leaving (turbine and condensers leak to		
		rs, at which time the leak termin		for a period of 24
7	-	enser volume: 5.00E+04 ft ³	· · ·	•
7		om free volume : 1.36E+05 fi ³ .	The total volume, 1.69Ė+05	ft^3 , is multiplied b
		unt for equipment located in the		
·	auxiliary co	ntrol room, instrument shop, an	d computer room).	
he		om normal ventilation intake flow		
		by taking the maximum flow rate		
· .		n flow rate of 12,750 cfm (REF		
1	cim inieaka III.3.d.(2).(ii	ge to the control room assumed	i in accordance with SKP 6	.4, section

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•	le Point Nuclear	Station	Unit: _	_1	Ď	isposition: _N/A
nginator	/Date (urtz / 4/29/98		er/Date		Calculation No.	Revisici
f.	unz / 4/29/96	<u> </u>	. Moisan / 5	5/1/98	H21C046	00
1	9 Credi	t is not taken f	or Control I		tment System initiatio	
·	10. U1 st	ack 0-2 hour X		$-0.4 \text{ soc}/\text{m}^3 \text{ ar}$	nd the 2-720 hour X/C	
<u>,</u>	11. Radio	activity assum	ned in the c	oolant as a re-	sult of a postulated C	
7	Halog	ens = 5.62E+1	04 Ci and N	loble Gas = 6.	64E+04 Ci	
1	12. Breat	hing rate of 3.4	47E-04 m ³ /	/ sec for the d	uration (0-720 hours)	of the accident
1 .	13. 10CF	R50 Appendix	: A, GDC 19	dose limit of	5 rem whole body or	equivalent. This
	equat	es to 30 rem t	hyroid and	30 rem beta (s	skin). 👘 👘 👘 👘	· · · · · ·
ŀ.		ATION		•		·
1				•		
	1.0 UNIT	1 COOLANT /	ACTIVITY			
{	ISOTOPE	Ci/MWt-	MWt		-	
1	ISCIOFE		2 2	TOTAL CORE		ACTIVITY
1		Ŭ	•	ACTIVITY	COOLANT FROM RODS	RELEASED TO THE CONDENSER
				. 3	(A)	S
				Ċi	Ci	Ci
	1-131	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
	l-133	4.80E+04	1887	9.06E+07	1.17E+04	1.17E+03
	1-134	6.20E+04	1887	1.17E+08	1.51E+04	1.51E+03
1.	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	FEELOE		4.075.00
1	KR-85M	6.50E+03	1887	5.66E+06 1.23E+07	· 1.07E+03 2.33E+03	1.07E+03 2.33E+03
	KR-85	3.00E+D2	1887	5.66E+05	1.07E+02	1.07E+02
ľ	. KR-87	1.20E+04	1887	2.26E+07	4.30E+03	4.30E+03
l ·	KR-88	1.70E+04	1887	3.21E+07	6.09E+03	.6.09E+D3 .
1	KR-89	2.00E+04	1887	3.77E+07	7.16E+03	7.16E+03
		•				
ļ	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
•		R 6 .Technical	TOTALS	3.51E+08	6.66E+04	enter Nuelser Sudam
	January 1	1974 (Table F	: 2 3.91	i or a single Cy	vie bolling water Re	actor Nuclear System,
		.02 (DATAJAS	•	J 出 11)		•
		• Column		νπ (1)		• • • •
				lumn @ / Tota	I Core Activity) * Tota	I Coolant Activity
	© Column		(0"			

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M. Kurtz / 4/29/98 A. C. Moisan / 5/1/98 H21C046 00 2.0 COOLANT SOURCE TERM ASSUMING 850 FAILED RODS TABLE 2 ISOTOPE CI/MWH MWt TOTAL CORE ACTIVITY COOLANT 0 0 0 ACTIVITY COOLANT 0 0 Ci	e Mile Point Nuclear Station		_1		Disposition: _N/	A	
100 2.0 COOLANT SOURCE TERM ASSUMING 850 FAILED RODS TABLE 2 ISOTOPE CI/MWH OU OL 0 OL 0 OL ISOTOPE CI/MWH OU OL 0 OL CI CI ISOTOPE CI/MWH OU OL OL CI CI CI ISOTOPE CI/MWH MWH TOTAL CORE ACTIVITY IN OL OL ISOTOPE CI/MWH MCTIVITY COOLANT ISOTOPE CI/MWH MCTIVITY COOLANT ISOTOPE CI/MWH MCTIVITY COOLANT ISOTOPE ISOTOPE ISOTAL CORE ACTIVITY <th col<="" th=""><th>ginator/Dze M. Kurtz / 4/29/98</th><th>Checker/Date A. C. Moisan / S</th><th>5/1/98</th><th>Calculation No. H21C046</th><th>•</th><th>Revisio</th></th>	<th>ginator/Dze M. Kurtz / 4/29/98</th> <th>Checker/Date A. C. Moisan / S</th> <th>5/1/98</th> <th>Calculation No. H21C046</th> <th>•</th> <th>Revisio</th>	ginator/Dze M. Kurtz / 4/29/98	Checker/Date A. C. Moisan / S	5/1/98	Calculation No. H21C046	•	Revisio
TABLE 2 ISOTOPE CI/MWt © MWt © TOTAL CORE © ACTIVITY IN COOLANT © Ci <						100	
TABLE 2 ISOTOPE CI/MWt © MWt © TOTAL CORE © ACTIVITY IN COOLANT © Ci <						•	
ISOTOPE Ci/MWt 0 MWt 0 TOTAL CORE ACTIVITY ACTIVITY COOLANT 0 0 0 0 0 0 0 1 2.90E+04 1887 5.47E+07 2.14E+05 1 1 0	2.0 COOLAI	NT SOURCE TERI	M ASSUMING	850 FAILED ROD	S .	•	
ISOTOPE Ci/MWt 0 MWt 0 TOTAL CORE ACTIVITY ACTIVITY COOLANT 0 0 0 0 0 0 0 1 2.90E+04 1887 5.47E+07 2.14E+05 0 1-131 2.90E+04 1887 7.93E+07 3.10E+05 0 1-132 4.20E+04 1887 9.06E+07 3.54E+05 0 1-133 4.80E+04 1887 9.05E+07 3.61E+05 0 1-134 6.20E+04 1887 9.25E+07 3.61E+05 0 1-135 4.90E+04 1887 9.25E+07 3.61E+05 0 1-135 4.90E+04 1887 9.25E+07 3.61E+05 0 KR-83M 3.00E+03 1887 1.23E+07 4.79E+04 KR-85 KR-85 3.00E+02 1887 5.66E+05 2.21E+04 KR-85 KR-85 1.70E+04 1887 3.21E+07 1.25E+05 KR-85 KR-85 1.70E+04 1887 3.40E+05	•		TABL	.E 2			
① ② ACTIVITY COOLANT ○	ISOTOPE	Ci/MWt			ACTIVITY IN		
		0	Q	ACTIVITY			
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 9.25E+07 3.61E+05 I-135 4.90E+04 1887 9.25E+07 3.61E+05 I-135 4.90E+03 1887 5.66E+06 2.21E+04 KR-83M 3.00E+03 1887 5.66E+05 2.21E+04 KR-85 3.00E+02 1887 5.66E+05 2.21E+04 KR-85 3.00E+02 1887 5.66E+05 2.21E+04 KR-85 3.00E+02 1887 3.21E+07 4.79E+04 KR-85 3.00E+02 1887 3.21E+07 1.25E+05 KR-88 1.70E+04 1887 3.40E+05 1.33E+03 XE-131M 1.80E+02 1887 3.40E+05 1.33E+03 XE-133M 2.00E+04 1887 3.21E+07 1.25E+05 XE-133M 1.00E+02 1887 3.21E+07 1.25E+05 XE-133 5.60E	· · ·		••• ••		• •		
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 I-135 4.90E+04 1887 9.25E+07 3.61E+05 I-135 4.90E+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+04 KR-85 3.00E+02 1887 5.66E+05 2.21E+04 KR-85 3.00E+02 1887 3.21E+07 1.25E+04 KR-88 1.70E+04 1887 3.21E+07 1.25E+05 KR-89 2.00E+02 1887 3.40E+05 1.33E+03 XE-131M 1.80E+02 1887 3.21E+07 1.25E+05 XE-133M 2.00E+04 1887 3.21E+07 1.25E+05 XE-135 9.80E+03 1887 1.85E+07 7.23E+04 XE-135 9.80E+0				Ci	Ci		
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 TOTAL 4.34E+08 70TAL 4.34E+08 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 3.21E+07 8.85E+04 KR-88 1.70E+04 1887 3.77E+07 1.47E+05 KR-89 2.00E+04 1887 3.40E+05 1.33E+03 XE-131M 1.80E+02 1887 3.40E+05 1.33E+03 XE-133M 2.00E+02 1887 3.21E+07 1.47E+05 XE-133M 1.00E+02 1887 3.21E+07 1.25E+05 XE-135 9.80E+03 1887 1.85E+07 7.23E+04 XE-135 9.80E+03 1887 1.85E+07 3.24E+05 XE-138 4.40E+04 18	I-131	2.90E+04	1887	5.47E+07	2.14E+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 TOTAL 4.34E+08 70TAL 4.34E+08 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 8.65E+04 KR-88 1.70E+04 1887 3.21E+07 1.25E+05 KR-89 2.00E+04 1887 3.77E+07 1.47E+03 XE-131M 1.80E+02 1887 3.77E+05 1.47E+03 XE-133M 2.00E+04 1887 3.21E+07 1.25E+05 XE-133M 2.00E+02 1887 3.21E+07 1.25E+05 XE-135M 1.70E+04 1887 3.21E+07 1.25E+05 XE-135M 1.70E+04 1887 3.0E+07 3.24E+05 XE-138 4.40E+04 1887 8.30E+07 3.24E+05 XE-138 4.40E+04 <td< td=""><td>I-132</td><td>4.20E+04</td><td>1887</td><td>7.93E+07</td><td>3.10E+05</td><td></td></td<>	I-132	4.20E+04	1887	7.93E+07	3.10E+05		
I-135 4.90E+04 1887 9.25E+07 3.61E+05 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-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 3.40E+05 1.33E+03 XE-131M 1.80E+02 1887 3.40E+05 1.33E+03 XE-133M 2.00E+04 1887 3.77E+05 1.47E+05 XE-133M 2.00E+02 1887 3.21E+07 1.25E+05 XE-133 5.60E+04 1887 3.21E+07 1.25E+05 XE-135M 1.70E+04 1887 3.21E+07 7.23E+04 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 <t< td=""><td>I-133</td><td>4.80E+04</td><td>1887</td><td>9.06E+07</td><td>3.54E+05</td><td>•</td></t<>	I-133	4.80E+04	1887	9.06E+07	3.54E+05	•	
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 ① GE BWR 6, Technical Description of a single cycle Bolling Water Reactor Nuclear System January 1, 1974 (Table F.2.3-9) ② 1850 MWt * 1.02 (DATA/ASSUMPTION # 1) 	XE-100	4.402104	1001		0.242.00		
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@ 1850 MWt * 1.02 (DATA/ASSUMPTION # 1)			in of a onigie	by the board grader		0,0101	
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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.

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FORMAT # NEP-DES-08, Rev. 02 (F02)

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P. 07

e Mile P	oint Nuclear Station	0 Unit1	Di	sposition: _N/A <u>·</u>
ginator/Dele	: / 4/29/98 ·	Checker/Date A. C. Moisan / 5/1/98	Calculation No. H21C046	Revision 00
	.1 4/23/30	A. C. Moisail 0/ 1/30	11210040	<u></u>
1	3.0 ACTIVIT	Y AVAILABLE FOR REL	EASE	· · .
· .				
			5.d all the noble gases and nser. Therefore, Table 2 Colu	
			ases to determine activity in t	
		· · · · · · · · · · · · · · · · · · ·		
Į.	· · · ·		TABLE 3	• • • • • • • • • •
1		ISOTOPE	ACTIVITY TO CONDENSE	:R ·
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		1-133	3.54E+04	
		I-134	4.57E+04 3.61E+04	
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i .		KR-87	8.85E+04	
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·		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 XE-138	1.25E+05 3.24E+05	
			•	•
· .		enser activity is input to D Ided in Appendix A.	RAGON run # 5341 dated 4/	29/98. The card input
	DRAGON inputs:	•		
	Main Condenser	volume: 5.00E+04 ft ³ (D/	TA/ASSUMPTION # 6)	
-	Main Condenser	release rate: 0.01 fractio	ns per day (DATA/ASSUMPT	ION #5.g)
	Unit 1 Control Ro	om (DATA/ASSUMPTIO	NS # 7 - 9)	•
	• volum	e: 1.36 E+05 ft ³	•	
		rate 0 to 720 hours: 358		
	filter e	fficiencies 0 - 720 hours		
	Breathing rate 0	- 720 hours is 3.47E-04 n	n ³ /sec (DATA/ASSUMPTION Ir X/Q: 1.22E-08 sec/m ³ (DAT	. #12) 14/4991 IMPTION #40
	0-2 nour X/Q: 3.1	22-04 Sec/m 2 - 24 hol	IT NUT TIZZE-VO SEC/M (UA I	

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CALCULATION	CONTINU	ATION S	HEET=	1 <u></u>	Page 8

CA HAWK (Next NUCLEAR ENGINEERING Nine Mile Point Nuclear Station Disposition: N/A Unit: Originator/Date Checker/Date Calculation No. T. M. Kurtz / 4/29/98 A. C. Moisan / 5/1/98 H21C046 RESULTS The 0 - 720 hour Unit 1 Control Room Doses are as follows TABLE 4 **CRDA TO U1 CONTROL ROOM** UNIT 1 CR DOSES, GDC 19 LIMIT, REM REM 4.52E-01 . THYROID 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 JOB # DATE DESCRIPTION OF RUN 5341 4/29/98 DRAGON (REF 10) CRDA to U1 Control Room-no filters Card image is given in Appendix A FORMAT # NEP-DES-08, Rev. 02 (F02)

	Unit:1			Disposit	ion: _N/A	<u> </u>
DrigInztor/Date . M. Kurtz / 4/29/98	Checker/Date A. C. Moisan / 5/1/98	•	Calculation No. H21C046	• •		Revision 00
ef. <u>REFERENCES</u> 1. Nine Mile Poir a. Table	nt 1 Final Safety Analysis	Report	Revision 14			•.
b. I.B.4.0 c. XV.C.3 d. XV.C.4 e. III.B.2. 2. General Elect 24011-P-A-13 3. Engineering F Reload 12, GI 4. NUREG 0800 Drop Accidem 5. Regulatory Gi Consequence A Boiling and 6. Technical Spe 7. Calculation 11 8. S10-210-HV1 9. Letter dated M 10. DRAGON cor	3.2 4.5.1 2 ric Standard Application fo Class III, August 1996. Report for Application of G ENE-770-31-1292, revisio , Standard Review Plan 1	E11 to n 2, Apr 5.4.9, R sed for l dent in t or. mendr Accident ry Conta Lempg ber NU-	Nine Mile Point ril, 1993 Ladiological Con Evaluating the he Fuel Handli nent 142, page t, revision 00 rol Room, revis les (NMPC) to l	Nuclear S nsequence Potential F ng and Sto 5 5 ion 00 D. B. Vass	station Uni es of Cont Radiologic brage Faci	it 1 rol Rod al ilily For
11. NMPC Calcul	ation S10-CR277.A-U1.21	.				

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			CALCUL			JATION SI	IEET	Page 10 (Next)
<u>۸</u>	line Mile Point	Nuclear Station	i	Init:1		·	Disposition:	N/A
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***** PROGRAM -- PRACON -- NULIS.VCODS.LEVOD-- 4/JO/46 -- *****

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