CP&L
A Progress Energy Company

James Scarola
Vice President
Harris Nuclear Plant

SERIAL: HNP-01-066
10CFR50.4

MAY 1 4 2001

United States Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
STEAM GENERATOR REPLACEMENT AND POWER UPRATE

Dear Sir or Madam:

By letters dated October 4, 2000 and December 14, 2000, Carolina Power & Light Company (CP&L) submitted license amendment requests to revise the Harris Nuclear Plant (HNP) Facility Operating License and Technical Specifications to support steam generator replacement and to allow operation at an uprated reactor core power level of 2900 megawatts thermal (Mwt). NRC letter dated April 10, 2001 requested additional information to support staff review of the proposed license amendment requests. The requested information is provided by the Enclosure to this letter.

The enclosed information is provided as a supplement to our October 4, 2000 and December 14, 2000 submittals and does not change the purpose or scope of the submittals, nor does it change our initial determinations that the proposed license amendments represent a no significant hazards consideration.

Please refer any questions regarding the enclosed information to Mr. Eric McCartney at (919) 362-2661.

P.O. Box 165
New Hill, NC 27562

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A001

Sincerely,

James Scarola

James Scarola
Vice President
Harris Nuclear Plant

James Scarola, 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 employees, contractors, and agents of Carolina Power & Light Company.

DARLENE S. YARBOROUGH
NOTARY
***
PUBLIC
HARNETT COUNTY, NC

Darlene S. Yarborough
Notary (Seal)

My commission Expires: 2-21-2005

KWS/kws

Enclosures

c: Mr. J. B. Brady, NRC Senior Resident Inspector
Mr. Mel Fry, NCDENR
Mr. R. J. Laufer, NRC Project Manager
Mr. L. A. Reyes, NRC Regional Administrator

bc:

Ms. D. B. Alexander
Mr. G. E. Attarian
Mr. R. H. Bazemore
Mr. L. R. Beller (BNP)
Mr. C. L. Burton
Mr. J. R. Caves
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Mr. J. W. Donahue
Mr. R. J. Duncan II
Mr. R. J. Field
Mr. W. J. Flanagan
Mr. K. N. Harris
Ms. L. N. Hartz
Mr. W J. Hindman
Mr. C. S. Hinnant
Mr. J. W. Holt
Mr. M. T. Janus
Mr. W. D. Johnson
Ms. T. A. Hardy (PE&RAS File)
Mr. R. D. Martin
Mr. T. C. Morton
Mr. W. M. Peavyhouse
Mr. J. M. Taylor
Nuclear Records
Harris Licensing File (s) (2 copies)

NRC Question 1 - Large-Break Loss-Of-Coolant Accident (LOCA)

In Table 2.22-10 of Appendix A to Reference 1, you have provided the major assumptions and parameters used for the radiological consequence analysis for the large-break LOCA. Provide the following additional information:

- Termination time of containment spray injection.

CP&L Response:

The Containment Spray System at the Harris Nuclear Plant (HNP) is designed for automatic switchover between the injection mode and containment sump recirculation mode. The containment spray system is normally lined up in the injection mode to take suction from the Refueling Water Storage Tank (RWST). The earliest start of switchover to the sump has been calculated as twenty minutes. When the RWST level reaches the Lo-Lo Level setpoint (approximately 23% of useable volume), the operating mode of the Containment Spray System is automatically changed from injection mode to recirculation mode by transferring the suction of the pumps to the containment sump, opening the valves in the sump recirculation lines, and closing the valves from the RWST. Thus, there is no termination time of containment spray injection in the sense that spray flow does not cease during the transfer from the injection to the recirculation modes of operation.

- Beginning time of containment spray recirculation phase.

CP&L Response:

Please see the above discussion regarding Containment Spray System operation. Recirculation actually begins with the automatic switchover to the containment sump source of suction water, and spray flow continues during switchover from the injection to the recirculation modes of operation. The earliest start of switchover to the sump has been calculated as twenty minutes.

- Termination time of containment spray recirculation phase.

CP&L Response:

HNP plant procedures provide for continuation of containment spray if the operators determine that sprays are necessary for radiological control (i.e., if containment radiological activities are high), or for containment pressure control (i.e., if containment pressures are high). There is no absolute set time for containment spray termination.

- Time to reach elemental iodine decontamination factor of 100.

CP&L Response:

A calculation was performed to develop the time steps and evaluate the decontamination factors achieved at these various times, as well as to evaluate the elemental and particulate decontamination factors as a function of time. The Maximum Allowable Elemental Decontamination Factor was determined from the equation in the Standard Review Plan Section 6.5.2 (value determined was 84.5), and the time to reach that Maximum Decontamination Factor was determined by interpolation. This time value was determined to be approximately 6132 seconds (approximately 1.7 hours).

- Time to reach particulate iodine decontamination factor of 50.

CP&L Response:

Similar to the Elemental Iodine Decontamination Factor (DF), a calculation was performed determine the particulate DF as a function of time, and the time to reach DF=50 was estimated by interpolation. The time to reach DF=50 was estimated to be approximately 7134 seconds (approximately 1.98 hours).

- Explain how you obtained a containment atmosphere mixing rate of 1.60 between the sprayed and unsprayed regions.

CP&L Response:

Mixing was determined from the containment volume and the forced air recirculation capacity of the containment air handling units. Shown below is the equation used to determine the mixing rate:

$$\frac{2\,\text{Fans} \;*\; 31{,}250\ \text{cfm/fan} \;*\; 60\ \text{min/hr}}{2{,}340{,}000\ \text{cu.ft.}} = 1.60 / hr$$

- Provide technical basis for assuming flashing fraction of 2 percent for the emergency core coolant system (ECCS) leakage. A constant enthalpy method should be used to determine the flashing fractions.

CP&L Response:

The 2% flashing fraction described in the current licensing basis (FSAR Section 6.5.2) is based on a constant enthalpy method and remains applicable for SGR/PUR conditions. Using a maximum sump temperature of 230 °F from the analysis of record and the steam tables for enthalpies between 230 °F and 212 °F, a flashing fraction of about 1.87% was calculated (and rounded to 2%). Since the date of that calculation of record, CP&L has evaluated this licensing basis flashing fraction for continued applicability using a 240 °F maximum (transient) sump temperature. The SGR/PUR evaluation using a maximum (transient) sump temperature of 244 °F was likewise evaluated. The disposition is summarized as follows:

a) The increased sump temperature (above the 230 °F "licensing basis" for 2% flashing fraction) occurs only for a short duration, relative to the duration of the event.

b) The sump temperature will drop below 212 °F after approximately 4 hours, and while no flashing would be expected after this time, the 2% flashing fraction assumption is continued for the duration of the event.

c) As discussed in the HNP FSAR, the release fraction based only on the constant enthalpy model is significantly conservative (by about a factor of 10) due to partition coefficients between the liquid and vapor fractions (see NUREG/CR-2493, April 1982).

NRC Question 2 - Steamline Break Accident

- In Table 2.22-1 of Attachment A Reference 1, you have provided an iodine protection factor of 51.1 using 45 cfm unfiltered air inleakage rate into the control room while reference 3 showed the iodine protection factor of 81.8. State the unfiltered air inleakage rate used in Reference 3 and explain the discrepancy.

CP&L Response:

The calculation text (page 8 of reference 3) and the Table 2.22-1 of Attachment A to Reference 1 show an iodine protection factor of 51.1, while the detailed calculation attached to reference 3 used a value of 81.8 for the control room iodine protection factor. It has been determined that this discrepancy can be attributed to a transcription error in transferring the input from the detailed calculation attached to reference 3 to the summary tables in the text of the calculation, and then into the Licensing Report (Reference 1). The iodine protection factors for control room, technical support center (TSC) and the emergency operations facility (EOF) reported in the reference 1 license amendment request are all affected by this transcription error. The calculation correctly uses the unfiltered inleakage value of 45 cfm for the control room and the EOF, and 13 cfm is used for the TSC. The HVAC flow rates and filter efficiencies were used to develop the iodine protection factors. The correct iodine protection factors are 81.8 for the control room, 96.0 for the TSC, and 14.7 for the EOF.

- Provide delay times (if applicable) in control room isolation after an isolation signal is generated. Explain how you obtained the control room operator thyroid doses in reference 3 for (1) initial blowdown of secondary coolant, (2) pre-existing iodine spike case, and (3) accident initiated iodine spike case. Did you use the ratios of the atmospheric dispersion factors between exclusion area boundary (EAB) and control room adjusting with the EAB dose and the iodine protection factor?

CP&L Response:

No delay time was modeled for control room isolation. The only delay times involved with control room isolation would be for the time required to close the associated dampers. Since the isolation signal would be generated (at the latest) by the radiation detectors at the intakes, the associated radioactive gas/particles would have to travel from the intakes past the closing dampers, which are located well inboard, to have an effect on the analysis. Thus, the delay time was considered insignificant, and consequently was not modeled.

CP&L Response (cont.):

Operator thyroid doses were determined by summing the dose contribution from the initial blowdown of the secondary coolant (assumed to be at the maximum Technical Specification allowable DEI-131 0.1 uCi/gm level), plus the dose contribution from the primary RCS leakage through the affected steam generator during the cooldown (assuming a total leakage of 1.0 gpm, comprised of 0.3 gpm through the affected steam generator, and 0.7 gpm through the unaffected steam generators).

For the pre-existing iodine spike case, the RCS coolant leaking into the secondary was assumed to be at a concentration of 60 uCi/gm in accordance with the HNP Technical Specifications. For the accident-initiated iodine spike case, a time varying concentration of RCS coolant DEI-131 was used consistent with an appearance rate of 500 times the equilibrium rate (determined from 1.0 uCi/gm steady state concentration and maximum credible cleanup rates).

Doses were calculated by direct multiplication of curie releases by the applicable X/Q, then multiplied by breathing rates, then multiplied by dose conversion factors, and then divided by iodine protection factors for the control room. No ratios between exclusion area boundary and control room were used.

- Include a copy of reference 3 in your response so it will be included as a docketed submittal for this review.

CP&L Response:

As requested, a copy of the pages from reference 3, previously faxed to the NRC to support a conference call, is attached to this RAI response (8 pages).

NRC Question 3 - Steam Generator Tube Rupture Accident

In Section 6.3.2 Reference 2, you have provided the major assumptions and parameters used for the radiological consequence analysis for the steam generator tube rupture accident. Provide the following additional information.

For pre-existing iodine spike:

- Iodine activity released (Ci) to the environment through flashed break flow from ruptured steam generator with iodine partition factor of 1.0.

- Iodine activity released (Ci) to the atmosphere from ruptured steam generator and intact steam generators over 0 to 2 hours and 2 to 8 hours with iodine partition factor of 100.

CP&L Response:

A conference call was held on March 29, 2001 between the NRC staff, Westinghouse, and CP&L to discuss this question. The following information was discussed with the NRC staff and is being provided to assist with the NRC staff review of the referenced license amendment requests. The total amount of activity released for each isotope is given in the TITAN5 output and is available, however the amount of activity released via specific pathways (i.e., flashed break flow, ruptured and intact steam generators) is not.

The total amount of iodine activity released for the pre-accident iodine spike:

I-131: 1.529E+02 Ci
I-132: 2.000E+02 Ci
I-133: 6.352E+02 Ci
I-134: 3.686E+01 Ci
I-135: 1.583E+02 Ci

The volume transfers are modeled as average flow rates over time periods. The time periods chosen are based on the thermal and hydraulic results and were divided into 6 time periods. The time periods are as follows:

1. from start until reactor trip and loss of offsite power (114 sec)
2. from reactor trip until the ruptured SG PORV fails open (722 sec)
3. from SG PORV failure until the ruptured SG PORV is closed (1922 sec)
4. from PORV closure until the break flow stops flashing (2500 sec)
5. from end of break flow flashing until break flow termination (4652 sec)
6. from break flow termination until 2 hours from start of event (2 hours)
7. from 2 hours until 8 hours from the start of the event (8 hours)

CP&L Response (cont.):

The flashed break flow is assumed to be released directly to the atmosphere and so it is subtracted from the break flow. The flashed break flow is also subtracted from the ruptured steam generator steam release to the atmosphere. The average mass flow rates of break flow (with flashed break flow deducted), flashed break flow, ruptured steam generator steam release to the atmosphere (with flashed break flow deducted) and intact steam generator steam release to the atmosphere are then used, together with the assumed steam generator leakage and the intact and ruptured steam generator steam releases to the condenser, and ruptured steam generator steam flow to the turbine driven auxiliary feedwater pump.

The data in the following table is taken from the SGTR thermal & hydraulic analysis.

Start of Period (sec)	End of Period (sec)	Total Break Flow During Period (lb_m)	Total Flashed Break Flow For Period (lb_m)	Total Ruptured SG Steam Flow To Atmosphere For Period (lb_m)	Total Intact SG Steam Flow To Atmosphere For Period (lb_m)
0	114	4874.0	837.3*	128300.0*	254100.0*
114	722	23736.0	1614.8	18460.0	29677.0
722	1922	58719.0	6920.5	119340.0	2697.0
1922	2500	29017.0	1470.6	0.0	58489.0
2500	4652	51487.0	0.0	500.0	85924.0
4652	7200	0.0	0.0	0.0	183300.0
7200	28800	0.0	0.0	35100.0	862800.0

*Flashed break flow and steam releases are released to the condenser for this time interval. An additional partition factor of 0.01 is applied to the condenser.

For accident-initiated iodine spike:

- Iodine activity released (Ci) to the environment through flashed break flow from ruptured steam generator with iodine partition factor of 1.0.

- Iodine activity released (Ci) to the atmosphere from ruptured steam generator and intact steam generators over 0 to 2 hours and 2 to 8 hours with iodine partition factor of 100.

- Post-trip average primary coolant iodine concentrations for 0 to 2 hour and 2 to 8 hour durations.

CP&L Response:

As stated in the response for the pre-accident iodine spike information, the total amount of activity released for each isotope is given in the TITAN5 output and is available; however, the amount of activity released via specific pathways (i.e., flashed break flow, ruptured and intact steam generators) is not available.

The total amount of iodine activity released for the accident-initiated iodine spike case:

I-131: 4.244E+01 Ci
I-132: 1.185E+02 Ci
I-133: 1.965E+02 Ci
I-134: 4.555E+01 Ci
I-135: 6.136E+01 Ci

The volume transfers are the same as discussed in the response for the pre-accident iodine spike except there is a constant transfer from the fuel volume to the RCS volume for the spike duration at spike rates of the following:

I-131 = 67 Ci/min
I-132 = 220.5 Ci/min
I-133 = 318.0 Ci/min
I-134 = 97.0 Ci/min
I-135 = 103.5 Ci/min

The iodine spike appearance rates for the accident initiated spike case have already been discussed with the NRC staff and background material was provided at the February 22, 2001 meeting between CP&L and the NRC staff.

NRC Question 4 - Control Rod Ejection Accident

In Table 2.22-7 of Appendix A to Reference 1, you have provided the major assumptions and parameters used for the radiological consequence analyses for the control rod ejection accident. In the table, you stated that you assumed a credit for fission product removal by the containment spray. State what initiated the containment spray and describe its operation.

CP&L Response:

The Rod Ejection accident was evaluated for either of two possible outcomes, in accordance with the Standard Review Plan 15.4.8 and Reg. Guide 1.77. For the event where the rod ejection does not breach the primary RCS (i.e., the force of the event does not damage the rod drive housing and cause a primary RCS breach or LOCA), the consequences are those resulting from a primary to secondary leakage event with fuel failures. This event was described in the submittal as being bounded by the single RCCA withdrawal and fuel misload event consequences.

For the second case, the force of the rod ejection is assumed to have breached the primary RCS in the area of the RCCA housing/rod drive, and thus initiated a LOCA. The event analysis used the Maximum Credible Accident as a basis, and applied correction factors for the amount of activity released into the RCS and then subsequently into the containment.

In a Maximum Credible Accident (considered to be a LBLOCA), containment sprays are initiated by the containment high pressure signal. The amount of steam/energy released into containment raises the containment pressure, and the sprays are needed, in part, to control the containment pressure as well as to provide the benefit of activity removal from the containment atmosphere. CP&L understands this NRC question to be a request for the basis of why this credit for fission product removal by the containment spray, which is provided for a LBLOCA initiation of containment spray, is also acceptable for application to a rod ejection accident, effectively a SBLOCA event.

For the SBLOCA Chapter 15 event analysis, Siemens Power Corporation (now Framatome-ANP) performed an evaluation of a spectrum of primary RCS breaks ranging from 2 to 4 inches in size. Also, a specific analysis was performed for the main steam line break (MSLB) inside containment to determine the containment response to a MSLB energy release. The mass and energy releases from the SBLOCA cases were compared to the mass and energy releases from the MSLB analysis to determine the containment and spray system response to a SBLOCA. It was determined that in a very short time (less than 7 minutes, ranging down to just under a minute) the containment spray setpoint would be reached.

CP&L Response (cont.):

Since the rod ejection event, which is postulated to breach the RCS through the RCS housing/drive mechanism, is effectively a SBLOCA, there will be enough energy released to containment to actuate the containment sprays. A detailed description of containment spray operation is described in FSAR Sections 6.2.2.2.2, 6.5.2.3.2, and 7.3.1.3.1.1.

NRC Question 5 - Meteorology

As a result of the February 22, 2001 meeting on the Harris alternate source term analysis, it is the staff's understanding that either 1) one set of relative concentration (X/Q) values for control room dose calculations is considered to be the design basis input for all of the Harris design basis accident dose assessments or 2) because this set is associated with the design basis accident currently resulting in the highest estimated dose, other X/Q values do not need to be considered.

- If this is correct, confirm and discuss why this single set of X/Q values is adequate (e.g., they bound the X/Q values for all of the other accidents):

CP&L Response:

The NRC staff understanding in item 1) above is correct. One set of X/Q values for control room dose calculations is the current design and licensing basis for HNP design basis accident dose assessments. In addition to being the current design and licensing basis, this single set of X/Q values is also bounding for other credible release paths.

CP&L has determined the single reported X/Q value for the control room is applicable and bounding for the complete spectrum of accidents considered by the FSAR Chapter 15 radiological consequences evaluations. The control room has two emergency intake sources, which are selectable by the operator to provide the lowest dose source of makeup air for the control room HVAC system. Releases from the containment are 160 feet from the north intake and approximately 150 feet from the south intake at the closest points.

Releases from the reactor auxiliary building (RAB) following an accident are filtered, released through the RAB Emergency Exhaust System (RABEES), and then exit through the plant vent stack #1. The distance from the centerline of this vent stack is 175 feet to the north control room emergency intake and 360 feet to the south control room emergency intake. Leakage from the RABEES would have to diffuse though a tortuous path of multiple doors and stairways to reach the control room area. There is no direct path for that leakage to reach the control room or the emergency intakes.

CP&L Response (cont.):

The Fuel Handling Building Emergency Exhaust System also exits through the Plant Vent Stack #1, which is the same release point discussed above. Therefore, this release path is also bounded by the containment X/Q.

- If the set of X/Q values is not bounding for all accidents, then for all sets not bounded, provide the X/Q values, and the methodology, inputs, and assumptions used to calculate the other X/Q values.

CP&L Response:

The one set of X/Q values is bounding for all accidents.

- In addition, list the accidents for which each set applies, as well as the postulated release location/receptor pairs. A figure would be helpful in understanding the physical relationship of the release locations and receptor pairs with respect to heights, distances, directions, and plant structures.

CP&L Response:

As stated above, only one set of X/Q values apply to all events. The simplified plant diagram provided by the staff in the February 22, 2001 meeting has been used to show the air intake and release locations. A copy of this diagram is attached to this RAI response (1 page).

- At the February 22, 2001, meeting, a handout related to Iodine activity was provided. Please submit a copy of the handout with this response so it will be included as a docketed submittal for this review.

CP&L Response:

As requested, the information related to iodine activity previously provided at the February 22, 2001 meeting and subsequently included as Enclosure 4 to the March 5, 2001 NRC letter summarizing the meeting is attached to this RAI response (2 pages).

During the meeting, two relatively recent license amendments were discussed. Both Amendment 88 and 97 relate to fuel-handling dose assessments. In both safety evaluations, the staff agreed with the licensee's finding that the fuel-handling dose to control room personnel was bounded by the dose for the LOCA. In the case of Amendment 97, staff concluded that the acceptability applied for Outage 9 and Operating Cycle 10. Other design-basis accidents were not considered and acceptability was discussed in terms of dose. X/Q values are a function of release location/receptor pairing that may be a function of accident. The highest X/Q values are not necessarily associated with the highest dose since dose is also a function of other inputs that may vary as a function of accident and, for a given accident, change over time with changes in assumptions related to plant design and/or operation. While Amendments 88 and 97 may provide some useful information, additional justification is needed for X/Q values for other release location/receptor pairings and other accidents.

CP&L Response:

The above paragraph discusses NRC concerns communicated at the time the above RAI questions were being developed. The concerns are about the way in which CP&L may have proposed to justify X/Q inputs on the basis of licensing history or prior NRC review of selected submittals. These concerns are more specifically communicated in NRC RAI question number five above. CP&L responses to each of the specific NRC questions above have considered the background material provided by the NRC in the above paragraph. In particular, the justification for a single set of X/Q values is not based solely on licensing history or prior NRC review of specific submittals. Rather, the analyses of radiological dose consequences demonstrate that a single set of X/Q values provides a demonstrably bounding input for the Harris plant configuration. Therefore, the concluding NRC statement above regarding additional justification being needed for X/Q values associated with other release location/receptor pairs and other accidents requires no additional information in the form of a CP&L response.

References:

(1) Carolina Power and Light Company letter to the NRC dated December 14, 2000, titled "Power Uprate."

(2) Carolina Power and Light Company letter to the NRC dated October 4, 2000, titled "Steam Generator Replacement."

(3) Carolina Power and Light Company, Harris Nuclear Plant Calculation Sheet, HNP-F/NFSA-0072, titled "SGRP/POWER UPRATE PROJECT," dated August 28, 2000. (Faxed to NRC to support a conference call)

7. BODY OF CALCULATION

7.1. Main Steam Line Break (MSLB) Outside Containment [FSAR Section 15.1.5]

Description: For dose calculations the following three cases are analyzed:

1. Pre-existing iodine spike case (An SRP 15.1.5, Appendix A, Paragraph III.4.(a) Event): A reactor transient has occurred prior to the postulated MSLB and has raised the primary coolant iodine concentration to the Technical Specification limit of 60 µCi/gm I-131 eq. limit identified in Ref. 11. The secondary coolant activity is assumed to be at the Technical Specification limit of 0.1 µCi/gm I-131 eq. identified in Ref. 12. No fuel failure is assumed.
2. Accident Generated iodine spike case (An SRP 15.1.5, Appendix A, Paragraph III.4.(b) Event: The reactor trip and/or primary system depressurization associated with the MSLB creates an iodine spike in the primary system. The spiking model is based on an increase in the iodine release rate from the fuel rods to the primary coolant to a value that is 500 times greater that the values that yields an equilibrium reactor coolant iodine concentration of 1 uCi/gm. The resulting releases to coolant used in this assessment are based on appearance rates provided in Ref. 7
3. Postulated Fuel Failure Case (An SRP 15.1.5, Appendix A, Paragraph III.5.Event: A MSLB outside containment with a bounding fuel failure assumption of 1% fuel cladding failure, and 0.7% centerline melt. This activity is released instantly to reactor coolant.

Other parameters applicable to both cases include:

- Steam Releases and Feedwater Flows are from Ref. 14, used as shown in Attachment U.
- A primary to secondary leak rate of 1 gpm (Ref. 10), with 0.30 gpm to the affected steam generator, and 0.70 gpm to the unaffected steam generators. The definition of the affected steam generator for this purpose is that one that is nearest the break, and more importantly, that the event single failure is a failure to close the associated MSIV. This results in primary to secondary leakage in the affected steam generator to be through the break, rather than through the atmospheric dump valves. No partition factor credit is taken for the affected steam generator. A factor of 100 is taken for the unaffected steam generators. This is consistent with Ref. 45.guidance. Note that HNP T.S. 3.4..2 limit leakage through any 1 steam generator to 150 gpd which is 0.104 gpm. Therefore, the above values are a significant conservatism.
- Releases during the 8-hour cooldown period are also modeled in the same manner as in Ref. 13. At 8 hours, the reactor cooling is assumed to be by the RHR system, and cooling using the steam generators and the atmospheric dump valve is assumed to have ceased. Thus, releases from the

not-affected steam generators have ceased. Some release from the affected steam generator could continue if the isolation valve is not closed. However, given the above conservatism in primary to secondary leakage rate treatment it is considered acceptable to cease the radiological evaluation at 8 hours for this pathway as well. The assumed leakage is almost three times the expected value for the first 8 hours. Improved X/Qs would also apply during the 8-24 hour period. Therefore, the existing 8 hour analysis is bounding.

Radiological Criteria: EAB and LPZ doses less than 10% of 10CFR100 limits of 25 rem whole body (EDE) and 300 rem thyroid, for cases 1 and 2, and 100% of the limits for case 3.

Attachment A contains the spreadsheet which is used to determine EAB, LPZ, CR, TSC, and EOF doses.

The resulting doses are:

CASE 1: Pre-Existing Iodine Spike, no Fuel Damage

SRP 15.1.5, Appendix A, Section III.4.(a) based Event

EAB	LPZ	CR	TSC	EOF	
3.67E+00	2.17E+00	7.72E-01	2.92E-01	1.16E-01	Total Thyroid Dose (rem)
1.01E-03	9.14E-04	1.03E-03	4.73E-04	2.03E-05	Total Whole Body Dose (rem)
1.28E-03	1.17E-03	3.40E-02	1.51E-02	9.16E-04	Total β-Skin Dose (rem)

CASE 2: Accident Generated Iodine Spike, no Fuel Damage

SRP 15.1.5, Appendix A, Section III.4.(b) based Event

EAB	LPZ	CR	TSC	EOF	
5.38E+00	1.31E+01	4.66E+00	1.76E+00	7.01E-01	Total Thyroid Dose (rem)
1.01E-03	9.14E-04	1.03E-03	4.73E-04	2.03E-05	Total Whole Body Dose (rem)
1.28E-03	1.17E-03	3.40E-02	1.51E-02	9.16E-04	Total β-Skin Dose (rem)

CASE 3: Bounding Fuel Damage

SRP 15.1.5, Appendix A, Section III.5.based Event

EAB	LPZ	CR	TSC	EOF	
8.45E+01	7.57E+01	2.70E+01	1.02E+01	4.06E+00	Total Thyroid Doses (rem)
7.03E-01	6.38E-01	7.20E-01	3.30E-01	1.42E-02	Total Whole Body Doses (rem)
7.63E-01	6.92E-01	2.02E+01	8.95E+00	5.44E-01	Total β-Skin Dose (rem)

These doses are within the identified acceptance criteria.

		B	C	D	E	F	G		I	J	K	L	M	N
1	**Main Steam Line Break Outside Containment Dose Assessment (FSAR 15.1.5)**													
2														
3	**CASE 1: Pre-Event Iodine Spiking Case**													
4		**SRP 15.1.5, Appendix A, Paragraph III.4.(a) Event**												
5														
6	**Dose from Initial Blowdown of Secondary Coolant**													
7														
8	0.1	Secondary Coolant Activity (uCi/gm) I-131 Equivalent T. S. Limit												
9	7.35E+07	Secondary Coolant Blowdown (gms)				Based on 162,000 lbs from Ref. 14								
10	7.35E+00	Release (Ci) of I-131 equivalent												
11	1.08E+06	Inhalation Dose Conversion Factor (Rem/Ci) for I-131												
12														
13	**EAB**	**LPZ**	**CR**	**TSC**	**EOF**	(Ci/min) appearance rates at 500x normal, corresponding to a normal 132 gpm cleanup and 1.0 uCi/gm								
14	6.17E-04	1.40E-04	4.08E-03	1.81E-03	1.10E-04	X/Q (sec/cu.m.)								
15	3.47E-04	3.47E-04	3.47E-04	3.47E-04	3.47E-04	Breathing Rate (cu.m./sec)								
16	1	1	81.8	96.0	14.7	Iodine Protection Factor								
17	**1.70E+00**	**3.86E-01**	**1.37E-01**	**5.19E-02**	**2.07E-02**	**Thyroid Dose from Secondary Coolant Blowdown (rem)**								
18	*Noble Gas Release considered negligible and no Whole Body or Skin Dose Calculated*													
19														
20	**Dose from Reactor Coolant Leakage and Secondary Releases During Cooldown Period**													
21	**(pre-existing iodine spike case)**													
22														
23	**Parameters for Release Assessment**													
24	60	Reactor Coolant Activity (uCi/gm) I-131 Equivalent T.S. Limit												
25	0.1	Secondary Coolant Activity (uCi/gm) I-131 Equivalent T. S. Limit												
26	1	Primary to Secondary Leak Rate (gpm)												
27	0.3	gpm through affected SG			0.7	gpm through intact SGs								
28	0-2 hr	0-2 hr	2-8 hr	2-8 hr	UPRATE STEAM AND FEEDWATE FLOW ADJUSTMENTS									
29	Steaming	Feedwater	Steaming	Feedwater										
30	386,000	482,000	892,000	967,000	From Ref. 14									
31														
32	100	Iodine Partition Factor for Leakage through intact steam generators												
33	1	Iodine Partition Factor for Leakage through affected steam generators												
34	8.50E+00	I-131 Eq. Release over 0-2 hour period (Ci)												
35	2.54E+01	I-131 Eq. Release over 2-8 hour period (Ci)												
36	2.83E+02	Xe-133 WB Eq. Release over 0-2 hour period (Ci)												
37	8.49E+02	Xe-133 WB Eq. Release over 2-8 hour period (Ci)												
38	2.15E+02	Xe-133 β-Skin Eq. Release over 0-2 hour period (Ci)												
39	6.44E+02	Xe-133 β-Skin Eq. Release over 2-8 hour period (Ci)												
40														
41	1.08E+06	Inhalation Dose Conversion Factor (Rem/Ci) for I-131												
42	5.77E-03	Air Submersion WB Dose Conversion Factor (Rem-M^3 / Ci-Sec) for Xe-133												
43	9.70E-03	Air Submersion β-Skin Dose Conversion Factor (Rem-M^3 / Ci-Sec) for Xe-133												
44														
45														
46														

		B	C	D	E	F	G		I	J	K	L	M	N
47	**CASE 1 DOSE ASSESSMENT RESULTS**													
48	**EAB**	**LPZ**	**CR**	**TSC**	**EOF**									
49	6.17E-04	1.40E-04	4.08E-03	1.81E-03	1.10E-04	X/Q (sec/cu.m.)								
50	3.47E-04	3.47E-04	3.47E-04	3.47E-04	3.47E-04	Breathing Rate (cu.m./sec)								
51	1	1	81.8	96.0	14.7	Iodine Protection Factor								
52	**1.97E+00**	**1.78E+00**	**6.35E-01**	**2.40E-01**	**9.55E-02**	**Cooldown Period Thyroid Dose for Case 1**								
53	1	1	25.8	25.0	35.3	Geometry Factors								
54	**1.01E-03**	**9.14E-04**	**1.03E-03**	**4.73E-04**	**2.03E-05**	**Cooldown Period WB Dose (rem) for Case 1**								
55	**1.28E-03**	**1.17E-03**	**3.40E-02**	**1.51E-02**	**9.16E-04**	**Cooldown Period β-Skin Dose (rem) for Case 1**								
56														
57	**3.67E+00**	**2.17E+00**	**7.72E-01**	**2.92E-01**	**1.16E-01**	**Total Thyroid Dose (rem) Initial Blowdown + Cooldown**								
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95	**CASE 2: Accident Generated Iodine Spiking Case**													
96		**SRP 15.1.5, Appendix A, Paragraph III.4.(b) Event**												
97														
98	**Dose from Initial Blowdown of Secondary Coolant**													
99														
100	0.1	Secondary Coolant Activity (uCi/gm) I-131 Equivalent T. S. Limit												
101	7.35E+07	Secondary Coolant Blowdown (gms)				Based on 162,000 lbs from Ref. 14								
102	7.35E+00	Release (Ci) of I-131 equivalent												
103	1.08E+06	Inhalation Dose Conversion Factor (Rem/Ci) for I-131												
104														
105	**EAB**	**LPZ**	**CR**	**TSC**	**EOF**									
106	6.17E-04	1.40E-04	4.08E-03	1.81E-03	1.10E-04	X/Q (sec/cu.m.)								
107	3.47E-04	3.47E-04	3.47E-04	3.47E-04	3.47E-04	Breathing Rate (cu.m./sec)								
108	1	1	81.8	96.0	14.7	Iodine Protection Factor								
109	**1.70E+00**	**3.86E-01**	**1.37E-01**	**5.19E-02**	**2.07E-02**	**Thyroid Dose from Secondary Coolant Blowdown (rem)**								
110	*Noble Gas Release considered negligible and no Whole Body or Skin Dose Calculated*													
111														
112	**Dose from Reactor Coolant Leakage and Secondary Releases During Cooldown Period**													
113	**(Accident-generated iodine spike case)**													
114														
115	**Parameters for Release Assessment**													
116	113.4	Reactor Coolant Activity (uCi/gm) I-131 Equivalent During the 0-2 hour post-MSLB period												
117	540.3	Reactor Coolant Activity (uCi/gm) I-131 Equivalent During the 2-8 hour post-MSLB period												
118	0.1	Secondary Coolant Activity (uCi/gm) I-131 Equivalent T. S. Limit												
119	1	Primary to Secondary Leak Rate (gpm)												
120	0.3	gpm through affected SG			0.7	gpm through intact SGs								
121	0-2 hr	0-2 hr	2-8 hr	2-8 hr	UPRATE STEAM AND FEEDWATE FLOW ADJUSTMENTS									
122	Steaming	Feedwater	Steaming	Feedwater										
123	386,000	482,000	892,000	967,000	From Ref. 14									
124														
125	100	Iodine Partition Factor for Leakage through intact steam generators												
126	1	Iodine Partition Factor for Leakage through affected steam generators												
127	1.59E+01	I-131 Eq. Release over 0-2 hour period (Ci)												
128	2.26E+02	I-131 Eq. Release over 2-8 hour period (Ci)												
129	2.83E+02	Xe-133 WB Eq. Release over 0-2 hour period (Ci)												
130	8.49E+02	Xe-133 WB Eq. Release over 2-8 hour period (Ci)												
131	2.15E+02	Xe-133 β-Skin Eq. Release over 0-2 hour period (Ci)												
132	6.44E+02	Xe-133 β-Skin Eq. Release over 2-8 hour period (Ci)												
133														
134	1.08E+06	Inhalation Dose Conversion Factor (Rem/Ci) for I-131												
135	5.77E-03	Air Submersion WB Dose Conversion Factor (Rem-M^3 / Ci-Sec) for Xe-133												
136	9.70E-03	Air Submersion β-Skin Dose Conversion Factor (Rem-M^3 / Ci-Sec) for Xe-133												
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138														
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140														

		B	C	D	E	F	G		I	J	K	L	M	N
141	**CASE 2 DOSE ASSESSMENT RESULTS**													
142	**EAB**	**LPZ**	**CR**	**TSC**	**EOF**									
143	6.17E-04	1.40E-04	4.08E-03	1.81E-03	1.10E-04	X/Q (sec/cu.m.)								
144	3.47E-04	3.47E-04	3.47E-04	3.47E-04	3.47E-04	Breathing Rate (cu.m./sec)								
145	1	1	81.8	96.0	14.7	Iodine.Protection Factor								
146	**3.68E+00**	**1.27E+01**	**4.52E+00**	**1.71E+00**	**6.81E-01**	**Cooldown Period Thyroid Dose for Case 2**								
147	1	1	25.8	25.0	35.3	Geometry Factors								
148	**1.01E-03**	**9.14E-04**	**1.03E-03**	**4.73E-04**	**2.03E-05**	**Cooldown Period WB Dose (rem) for Case 2**								
149	**1.28E-03**	**1.17E-03**	**3.40E-02**	**1.51E-02**	**9.16E-04**	**Cooldown Period β-Skin Dose (rem) for Case 2**								
150														
151	**5.38E+00**	**1.31E+01**	**4.66E+00**	**1.76E+00**	**7.01E-01**	**Total Thyroid Dose (rem) Initial Blowdown + Cooldown**								
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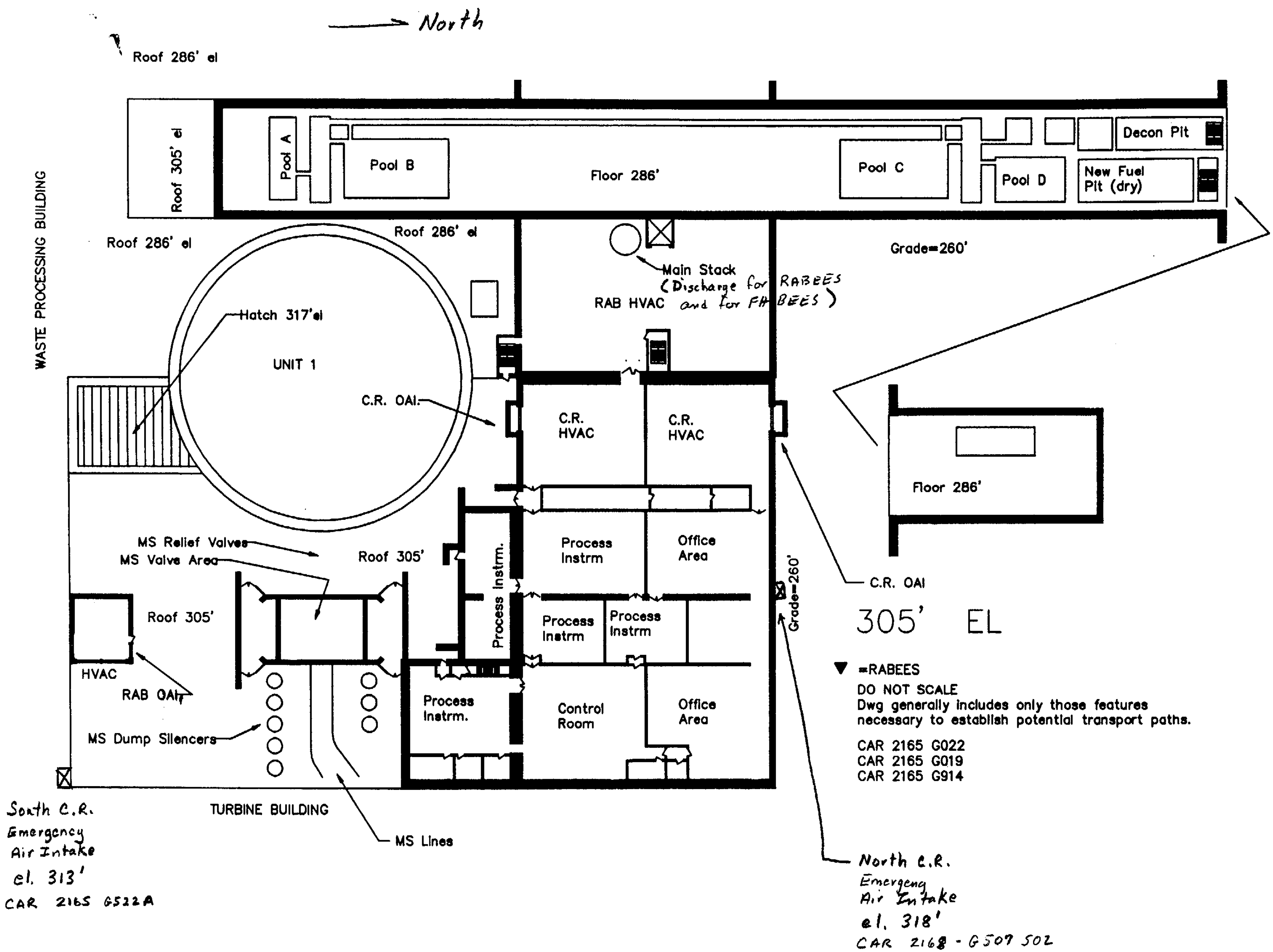
188	**CASE 3: Bounding fuel damage case (highest worth roα stuck out of core)**												
189		**SRP 15.1.5, Appendix A, Paragraph III.5 Event**											
190													
191	9406	Reactor coolant liquid volume (cu.ft. at 593F, 2250 psia)											
192	0.0229	Specific Volume at 593F, 2250 psia (cu.ft./lbm)											
193	4.11E+05	Reactor Coolant mass (lbm)											
194	1.86E+08	Reactor Coolant mass (gms)											
195	1.13E+08	I-131 equivalent in Core (Ci)											
196	1.29E+07	I-131 equivalent in Gap (Ci)											
197	1.00E+08	I-131 equivalent in Fuel, and not Gap (Ci)											
198	1.11E+10	Xe-133 WB equivalent in Core (Ci)											
199	1.11E+09	Xe-133 WB equivalent in Gap (Ci)											
200	9.98E+09	Xe-133 WB equivalent in Fuel, and not Gap (Ci)											
201	7.16E+09	Xe-133 β-Skin equivalent in Core (Ci)											
202	7.17E+08	Xe-133 β-Skin equivalent in Gap (Ci)											
203	6.44E+09	Xe-133 β-Skin equivalent in Fuel, and not Gap (Ci)											
204	0.01	Assumed Cladding Failure											
205	0.007	Assumed Centerline Melt (of which, 50% of iodines and 100% of noble gasses are released to coolant)											
206	4.80E+05	I-131 equivalent activity release to coolant (Ci)											
207	2.58E+03	Initial I-131 equivalent coolant concentration (uCi/gm)											
208	8.10E+07	Xe-133 WB equivalent activity release to coolant (Ci)											
209	4.35E+05	Initial WB equivalent Xe-133 coolant concentration (uCi/gm)											
210	5.23E+07	Xe-133 β-Skin equivalent activity release to coolant (Ci)											
211	2.81E+05	Initial β-Skin equivalent Xe-133 coolant concentration (uCi/gm)											
212													
213	**Dose from Initial Blowdown of Secondary Coolant**												
214													
215	0.1	Secondary Coolant Activity (uCi/gm) I-131 Equivalent T. S. Limit											
216	7.35E+07	Secondary Coolant Blowdown (gms)				Based on 162,000 lbs from Ref. 14							
217	7.35E+00	Release (Ci) of I-131 equivalent											
218	1.08E+06	Inhalation Dose Conversion Factor (Rem/Ci) for I-131											
219													
220	**EAB**	**LPZ**	**CR**	**TSC**	**EOF**								
221	6.17E-04	1.40E-04	4.08E-03	1.81E-03	1.10E-04	X/Q (sec/cu.m.)							
222	3.47E-04	3.47E-04	3.47E-04	3.47E-04	3.47E-04	Breathing Rate (cu.m./sec)							
223	1	1	81.8	96.0	14.7	Iodine Protection Factor							
224	**1.70E+00**	**3.86E-01**	**1.37E-01**	**5.19E-02**	**2.07E-02**	**Thyroid Dose from Secondary Coolant Blowdown (rem)**							
225	*Noble Gas Release considered negligible and no Whole Body or Skin Dose Calculated*												
226													
227													
228	**Parameters for Release Assessment**												
229	2.58E+03	Reactor Coolant Activity (uCi/gm) I-131 Equivalent, with bounding fuel failure											
230	0.1	Secondary Coolant Activity (uCi/gm) I-131 Equivalent T. S. Limit											
231	1	Primary to Secondary Leak Rate (gpm)											
232	0.3	gpm through affected SG			0.7	gpm through intact SGs							
233	0-2 hr	0-2 hr	2-8 hr	2-8 hr	UPRATE STEAM AND FEEDWATE FLOW ADJUSTMENTS								
234	Steaming	Feedwater	Steaming	Feedwater									

		B	C	D	E	F	G		I	J	K	L	M	N
235	386,000	482,000	892,000	967,000	From Ref. 14									
236														
237	100	Iodine Partition Factor for Leakage through intact steam generators												
238	1	Iodine Partition Factor for Leakage through affected steam generators												
239	3.58E+02	I-131 Eq. Release over 0-2 hour period (Ci)												
240	1.08E+03	I-131 Eq. Release over 2-8 hour period (Ci)												
241	1.97E+05	Xe-133 WB Eq. Release over 0-2 hour period (Ci)												
242	5.92E+05	Xe-133 WB Eq. Release over 2-8 hour period (Ci)												
243	1.27E+05	Xe-133 β-Skin Eq. Release over 0-2 hour period (Ci)												
244	3.82E+05	Xe-133 β-Skin Eq. Release over 2-8 hour period (Ci)												
245														
246	1.08E+06	Inhalation Dose Conversion Factor (Rem/Ci) for I-131												
247	5.77E-03	Air Submersion WB Dose Conversion Factor (Rem-M^3 / Ci-Sec) for Xe-133												
248	9.70E-03	Air Submersion β-Skin Dose Conversion Factor (Rem-M^3 / Ci-Sec) for Xe-133												
249														
250														
251														
252	**CASE 3 DOSE ASSESSMENT RESULTS**													
253	**EAB**	**LPZ**	**CR**	**TSC**	**EOF**									
254	6.17E-04	1.40E-04	4.08E-03	1.81E-03	1.10E-04	X/Q (sec/cu.m.)								
255	3.47E-04	3.47E-04	3.47E-04	3.47E-04	3.47E-04	Breathing Rate (cu.m./sec)								
256	1	1	81.8	96.0	14.7	Iodine Protection Factor								
257	**8.28E+01**	**7.53E+01**	**2.68E+01**	**1.01E+01**	**4.04E+00**	**Cooldown Period Thyroid Dose for Case 3**								
258	1	1	25.8	25.0	35.3	Geometry Factors								
259	**7.03E-01**	**6.38E-01**	**7.20E-01**	**3.30E-01**	**1.42E-02**	**Cooldown Period WB Dose (rem) for Case 3**								
260	**7.63E-01**	**6.92E-01**	**2.02E+01**	**8.95E+00**	**5.44E-01**	**Cooldown Period β-Skin Dose (rem) for Case 3**								
261														
262	**8.45E+01**	**7.57E+01**	**2.70E+01**	**1.02E+01**	**4.06E+00**	**Total Thyroid Dose (rem) Initial Blowdown + Cooldown**								


North
Roof 286' el
Roof 305' el
Pool A
Pool B
Floor 286'
Pool C
Pool D
Decon Pit
New Fuel Pit (dry)
WASTE PROCESSING BUILDING
Roof 286' el
Roof 286' el
Grade=260'
Main Stack
(Discharge for RABEES and for FH BEES)
RAB HVAC
Hatch 317'el
UNIT 1
C.R. OAI
C.R. HVAC
C.R. HVAC
Floor 286'
MS Relief Valves
MS Valve Area
Roof 305'
Process Instrm.
Process Instrm
Office Area
C.R. OAI
Grade=260'
Roof 305'
Process Instrm
Process Instrm
305' EL
HVAC
RAB OAI
MS Dump Silencers
Process Instrm.
Control Room
Office Area
▼ =RABEES
DO NOT SCALE
Dwg generally includes only those features necessary to establish potential transport paths.
CAR 2165 G022
CAR 2165 G019
CAR 2165 G914
South C.R. Emergency Air Intake el. 313'
CAR 2165 G522A
TURBINE BUILDING
MS Lines
North C.R. Emergency Air Intake el. 318'
CAR 2168 - G507 502

The primary coolant I-131 activity prior to the accident is 0.35 μCi/g dose equivalent (D.E.) I-131.

The initial mass in the RCS is 1.729E8 g.

The activity levels of the Iodine nuclides (based on 1% fuel defects) are listed in the table below and converted to 1.0 μCi/g D.E. I-131 by multiplying the activity for each isotope by its associated dose conversion factors and dividing by the I-131 dose conversion factor. The total activity corresponding to 1.0% fuel defects is 3.002 μCi/g D.E. I-131 and is calculated by adding together all of the converted to dose equivalent I-131 activities. The activity corresponding to a total of 1.0 μCi/g D.E. I-131 is found by dividing the 1% fuel defect activity for each nuclide by 3.002. The activity corresponding to a total of 0.35 μCi/g D.E. I-131 is found by multiplying the 1.0 μCi/g D.E. I-131 fuel defect activity for each nuclide by 0.35.

The RCS activity is calculated by taking the mass in the RCS, 1.729E8 g, and multiplying by the nuclide activity based on dose equivalent I-131 and then dividing by E6 (unit conversion).
For example, 1.729E8 g * 0.200 μCi/g = 3.458E7 μCi / E6 μCi/Ci = 34.6 Ci.

Nuclide	Activity – 1% Fuel Defect (μCi/g)	ICRP-30 Dose Conversion Factors (rem thyroid/Ci inhaled)	Activity – D.E. I-131 (μCi/g)	Activity Corresponding to a total of 1.0 μCi/g D.E. I-131	Activity Corresponding to a total of 0.35 μCi/g D.E. I –131	RCS Activity Corresponding to 0.35μCi/g (Ci)
I-131	1.71	1.07E6	1.71	0.570	0.200	34.6
I-132	2.47	6.29E3	0.01452	0.823	0.288	49.8
I-133	7.23	1.81E5	1.223	2.408	0.843	145.8
I-134	0.567	1.07E3	5.67E-4	0.189	0.066	11.4
I-135	1.84	3.14E4	0.054	0.613	0.215	37.2
Total			3.002			278.8

The RCS activity (A) is related to the appearance rate by:

A = Iodine Appearance Rate (P) / Removal rate (λ)
where $\lambda = \lambda_{purification} + \lambda_{decay}$
with $\lambda_{purification} = [(1 - 1/DF)(F) + L] / V$
where DF = Maximum Decontamination Factor provided for iodine

DF = infinite, thus 1/DF = 0
F = Maximum purification mass flow rate (letdown flow)
Max purification flow is 120 gpm and this is conservatively increased by 10% to 132 gpm (cold conditions are assumed so density = 62.4 lb/ft^3)
F = (132 gpm)(0.13368 ft^3/gal)(62.4 lb/ft^3) = 1101.1 lb/min = 66,066 lb/hr

L = Leakage from the primary coolant system
= 42 gpm at cold conditions – assume 62.4 lb/ft^3
= (42 gpm)(0.13368 ft^3/gal)(62.4 lb/ft^3) = 350.3 lb/min
= 21,018 lb/hr

V = RCS water mass = 1.729E8 g

Thus, $\lambda_{purification}$ = (66,066 + 21,018) lb/hr * 453.6 g/lb / 1.729E8 g = 0.2285 hr^{-1}

The values for λ_{decay} are combined with the purification term to create a total removal term for each isotope:

λ_{I-131} = 0.2285 + 0.00359 = 0.2321 hr^{-1}
λ_{I-132} = 0.2285 + 0.303 = 0.5315 hr^{-1}
λ_{I-133} = 0.2285 + 0.0333 = 0.2618 hr^{-1}
λ_{I-134} = 0.2285 + 0.791 = 1.0195 hr^{-1}
λ_{I-135} = 0.2285 + 0.105 = 0.3335 hr^{-1}

The RCS inventory is (from table above)
A_{I-131} = 34.6 Ci
A_{I-132} = 49.8 Ci
A_{I-133} = 145.8 Ci
A_{I-134} = 11.4 Ci
A_{I-135} = 37.2 Ci

Normal appearance rate is calculated by P = Aλ
P_{I-131} = (34.6 Ci)*(0.2321 hr^{-1}) / (60 min/hr) = 0.134 Ci/min
P_{I-132} = (49.8 Ci)*(0.5315 hr^{-1}) / (60 min/hr) = 0.441 Ci/min
P_{I-133} = (145.8 Ci)*(0.2618 hr^{-1}) / (60 min/hr) = 0.636 Ci/min
P_{I-134} = (11.4 Ci)*(1.0195 hr^{-1}) / (60 min/hr) = 0.194 Ci/min
P_{I-135} = (37.2 Ci)*(0.3335 hr^{-1}) / (60 min/hr) = 0.207 Ci/min

The appearance rates are assumed to increase by a factor of 500 for the SGTR. The iodine spike appearance rates are thus:
P_{I-131} = 67.0 Ci/min
P_{I-132} = 220.5 Ci/min
P_{I-133} = 318.0 Ci/min
P_{I-134} = 97.0 Ci/min
P_{I-135} = 103.5 Ci/min