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PRELIMINARY RISKS ASSOCIATED WITH POSTULATED TRITIUM RELEASE FROM PRODUCTION REACTOR OPERATION

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DISCLAIMER

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ABSTRACT

The Probabilistic Risk Assessment (PRA) of Savannah River Plant (SRP) reactor operation is assessing the off-site risk due to tritium releases during postulated full or partial loss of heavy water moderator accidents. Other sources of tritium in the reactor are less likely to contribute to off-site risk in non-fuel melting accident scenarios. Preliminary determination of the frequency of average partial moderator loss (including incidents with leaks as small as .5 kg) yields an estimate of ~1 per reactor year. The full moderator loss frequency is conservatively chosen as 5×10^{-3} per reactor year. Conditional consequences, determined with a version of the MACCS code modified to handle tritium, are found to be insignificant. The 95th percentile individual cancer risk is 4×10^{-8} per reactor year within 16 km of the release point. The full moderator loss accident contributes about 75% of the evaluated risks.

INTRODUCTION

The Savannah River Plant in Aiken, South Carolina, is currently the sole producer of tritium and plutonium for the U.S. Department of Energy. The tritium is produced by neutron irradiation of LiAl targets in heavy water-moderated, low-temperature production reactors. These reactors and supporting processing facilities at the Savannah River Plant (SRP) site are operated by E.I. du Pont de Nemours & Company.

As part of an ongoing Probabilistic Risk Assessment (PRA) of SRP reactor operation, this paper will present preliminary risk estimates due to postulated reactor accidents involving nominal operation of a SRP production reactor and its associated tritium inventories. Accidents involving melting of reactor core assemblies will not be addressed. Tritium releases not associated with core melts are covered. Core melts will be addressed by the PRA at a later date. Preliminary analyses have indicated that the contribution of tritium related health effects to risk for core melt accidents is small.

The SRP site currently contains three operating production reactors. These reactors are located near the center of the site and are no closer than 9 km to the site boundary. SRP also contains several other facilities associated with chemical separations, waste treatment, tritium processing, and fuel manufacturing.

PRA COMPONENTS

The PRA for the production reactors is composed of three components or levels of analysis. These three levels are shown below in Figure 1.

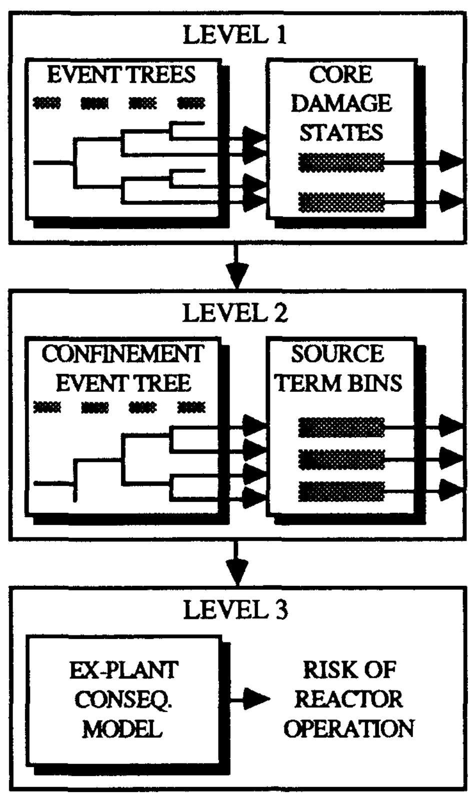

Figure 1. PRA Methodology Overview

The Level 1 PRA deals with the assessment of the likelihood of accidental releases of radioactive material from the reactor and its supporting systems. This assessment is performed using event tree methodology to model accident sequences. Event trees are developed for accident initiators which represent reactor off-normal events. Branch point probabilities in the event trees are generally determined by fault tree analysis of systems. An SRP specific data base provides the information used for the quantification of component failure probabilities and initiating event probabilities.

Each event tree sequence that results in a potentially significant release of radioactive material is assigned to a core damage state. These states of core damage are grouped by physical and temporal characteristics that can impact accident severity and release rates. The core damage states and their frequency of occurrence are the interface between the Level 1 and Level 2 PRA.

The Level 2 PRA evaluates the quantity and rate of radioactive material released from the reactor building for each of the core damage states. This information dealing with releases is called the accident source term and is expressed in terms of the following; Curies of isotopes of interest, chemical species, physical attributes (temperature, gas/liquid/aerosol/particulate, velocity), and timing of release. The impact of confinement system response (eg., filter effectiveness) is included in the determination of the source terms. This assessment is also performed using event tree methodology to model accident progression. Branch point probabilities in the event tree are determined by a combination of fault tree analyses of confinement systems and engineering calculations of accident phenomenology.

Each path through the confinement event tree yields a source term. Similar source terms are combined or binned for further evaluation in the Level 3 analysis. These source term bins and their frequency of occurrence are the interface between the Level 2 and Level 3 PRA.

The Level 3 PRA takes the source terms provided by the Level 2 analysis and assesses the health effects resulting from the releases. Health effects can be determined for average or worst case meteorological conditions. Health effects on on-site and off-site populations can be evaluated as well as evacuation plans. An atmospheric dispersion model for the site is used in this evaluation. The resulting Level 3 analysis health effects for a source term bin are then weighted by the frequencies from the corresponding Level 1 and Level 2 sequences yielding a measure of the risk.

The SRP PRA is currently in progress. This paper will use preliminary results from the Level 1 and Level 3 analyses for the evaluation of risks for tritium releases. The Level 2 methodologies are not sufficiently developed at this time to utilize in the following discussions. Any gaps left in this paper due to the omission of the Level 2 analyses will be noted along with any assumptions that are made in place of the Level 2 analyses.

TRITIUM INVENTORIES

Within a heavy water production reactor, tritium is found in a number of locations. A simplified drawing of the reactor tank in a SRP production reactor is shown in Figure 2 and those elements that contain a tritium source are shown in bold lettering.

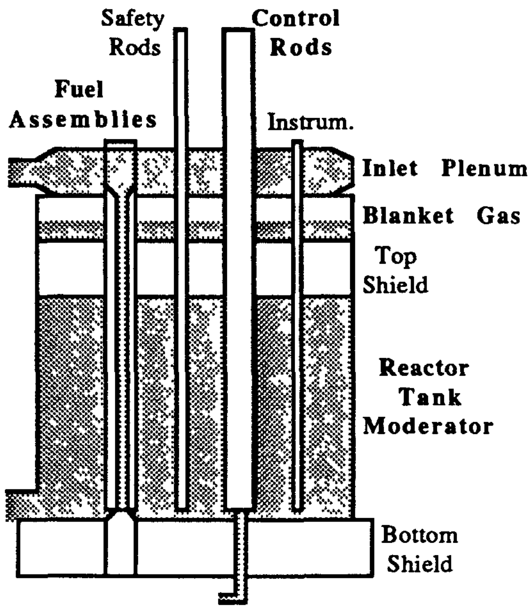

Figure 2. Tritium Locations

The blanket gas system which provides slight reactor pressurization contains the smallest inventory of tritium. This system contains a small amount of a mix of T_2, TH, DTO, and T_2O as impurity species. The blanket gas system is connected to a treatment facility for recombination of hydrogen gas from the system. Since this system is functioning at all times during reactor operation, there is insufficient time for the accumulation of significant quantities of tritium in the blanket gas. The tritium in this source is contained by the blanket gas plenum and the blanket gas system piping.

The next largest source of tritium is found in the reactor control rods. These control elements are composed of a lithium/aluminum alloy contained in an aluminum cladding. The tritium in the control rods is bound in the alloy and requires significant heat to become free. The clad and alloy are the only barriers to this tritium source because the control rod standpipes are open to the reactor building atmosphere.

The process water system (reactor tank moderator, process water piping, and inlet plenum) has the next largest inventory of tritium. A large amount (approximately 5 MCi) of tritium can reside in the process water system primarily in the form of DTO and T_2O. This source of tritium is not bound except by the constraints of the process water system piping.

The individual reactor lithium target assemblies (part of the reactor fuel assemblies) collectively contain the largest inventory of tritium. These assemblies are composed of a lithium/aluminum alloy contained in an aluminum cladding. This source differs from the control rods because the tritium is secondarily contained in the process water system piping which provides another barrier to release of this source.

INVENTORIES OF INTEREST

Due to the limited inventory of tritium in the blanket gas, this source is not of interest in computation of the probability of significant releases of tritium. Consequences from a release of the blanket gas inventory are insignificant.

Released tritium from a control rod would vent out of a control rod assembly standpipe directly to the reactor building confinement system with subsequent release out a 61 m stack. Current reactor charges do not produce sufficient flux to melt a control rod even with a complete interruption of cooling flow to the rod. Release from this source is only possible in events that also include core melting. The contribution of health effects due to the tritium from this source during core melt accidents is small compared to fission product contributions and therefore is not considered for further analysis in this paper.

Single target assembly releases are bounded by a release from the process water system since they would contribute a small fraction of the process water system inventory to the release. Multiple target assembly failures will only occur with fuel melting in the reactor. The impact of fission product releases from fuel melts overwhelm the impact of any additional tritium releases and are addressed elsewhere in the PRA. Tritium releases occurring during fuel melting accidents are not addressed in this paper.

The process water system release of tritium is the major emphasis of this paper. This accident is called a loss of coolant accident or a LOCA. There is a potential for a LOCA to continue on to become a major core melt event but that aspect of the problem is not addressed here since the tritium release contribution to risk in a core melt event is not significant. A LOCA releases the tritium in the process water system to the reactor building where it is eventually vented out the 61 m stack. There is no credit taken for any retention of the tritium in the confinement system as will be discussed. LOCAs are divided into two categories for the purposes of this paper. The first category deals with LOCAs of sufficient size to release the entire process water system inventory. The second category deals with LOCAs which are limited in their size and may be termed minor leakage. This latter category of LOCA has an average release of 6,000 Curies of tritium and occurs about once per reactor year of operation. Preliminary results of detailed pipe analyses have indicated that the frequency of LOCAs large enough to release the full process water inventory is about 5×10^{-3} per reactor year. These larger LOCAs are primarily the result of breaks in process water system piping expansion joints.

LEVEL 2 ASSUMPTIONS

As stated above, the LOCA will dump the process water inventory onto the reactor building floor. For the expansion joint breaks, the water is initially dropped to the bottom of the reactor building. Building sump pumps automatically start and pump the water to an underground tank with sufficient capacity to hold the entire process water system inventory. An emergency coolant addition system will pump light water into the process water system and out through the pipe break. Eventually, the entire inventory of heavy water will be dumped into the building. When the underground tank fills, the water then flows to a large tank located outside the building. Both the underground tank and the large tank are vented back to the confinement system. If sufficient water is pumped out of the building to fill both tanks, the large tank overflows to a retention basin.

If the sump pumps fail, the heavy water will remain in the reactor building until the light water addition is sufficient to fill the building. The mix of heavy and light water will then flow through a drain system to the large tank until that tank overflows to the retention basin.

The confinement system draws a negative pressure on the reactor building and on the two tanks. Fans draw air from the building through a filter system and discharge to a 61 m stack. Failure of the fans would delay and lengthen the release of the tritium from the building. Releases from the building without fan operation would occur at or near ground level with some fraction going up the stack due to a chimney effect.

The Level 2 analyses will eventually evaluate all of these variations in release height, time, and duration. This paper will conservatively assume that the tritium release rapidly vents to the 61 m stack and that the filter system will not impact the tritium release in any way.

LEVEL 3 METHODOLOGY

Off-Site Consequences Code

Since the time of the WASH-1400 Reactor Safety Study[1] in the United States, code packages have been available to allow the risk assessment analyst to quantify off-site consequences given postulated reactor accident scenarios. The most complete and recent model for off-site impacts is contained in the MELCOR Accident Consequence Code System (MACCS)[2], issued as MACCS 1.4 in July, 1987. MACCS is a subsystem of stand-alone modules designed to be used after an accident-specific source term (quantity, identity, and timing) has been calculated for PRA or severe accident assessment purposes.

The MACCS user specifies radionuclide inventory, release and deposition parameters, and dispersion characteristics for 9 groups of nuclides to initiate the calculation. A site specific meteorological file must also be entered. Typically one year of hourly readings are entered in this manner for weather sampling purposes. MACCS can be directed to perform a sampling and sorting scheme to take the large hourly data base and map 8760 or so sets of data into categories defined by wind speed, atmospheric stability, and the location of rain. A population distribution about the reactor site, emergency response assumptions, and land usage characteristics are needed for the execution of remaining modules.

In consideration of the radiation doses due to all or part of a reactor inventory of radionuclides, MACCS determines early exposure impacts due to: 1) direct external exposure to radioactivity in the passing cloud - cloudshine; 2) exposure due to inhalation; 3) exposure due to radioactive material deposited on the ground - groundshine; and 4) inhalation of resuspended material. Long-term exposure pathways are from 3 sources: 1) long-term groundshine; 2) long-term resuspension; and 3) ingestion of contaminated water/food. MACCS uses dose factors derived from ICRP-30 and health effects from NUREG/CR-4214.[3-7]

Version 1.4 of MACCS includes a sixty-nuclide dose conversion file appropriate for analysis of the LWRs that dominate the commercial power industry in the United States. Ten organ types are included and have dose factors for the pathways described earlier. To extend MACCS for computation of the consequences associated with operation of the SRP HWRs, tritium dose conversion data have been added for 3 pathways: 1) inhaled-acute; 2) inhaled-chronic; and 3) ingestion. The dose factors are based on a quality factor of 1.7 and are:

4.3×10^{-11} Sv/Bq (1.58×10^{-7} mrem/pCi) - inhalation pathways and
2.8×10^{-11} Sv/Bq (1.04×10^{-7} mrem/pCi) - ingestion pathway.[8]

The user must now specify a 61-nuclide input deck with elemental group, half-life, and inventory assignment information.

Code Limitations

The current modified-MACCS is an attempt to include tritium in Level 3 considerations for the SRP PRA of reactor operation. It does not model detailed tritium exchange behavior as a function of climatic humidity or rainfall. Also, the changes have not included adding tritium to MACCS models that are important metabolically in food ingestion. Therefore, long-term effects due to food ingestion remain dominated by ^{89}Sr, ^{90}Sr, ^{134}Cs, ^{137}Cs, ^{131}I, and ^{133}I. However, by making judicious choices for the available transport options in the modified code, viz. assigning tritium to the noble gas group and allowing wet deposition in tritium-only source term calculations, the incremental consequences due to tritium may be estimated conservatively for purposes of the current PRA.

APPLICATION INPUTS

Two tritium release profiles were developed as input for the modified-MACCS calculation. The tritium inventory is assigned to the MACCS Group 1 bin (noble gas) in both analyses. For the full moderator inventory analysis (Table 1a), the tritium is released as an early step function followed by a late, long-term release of moderate intensity. The release history follows that of a full-core noble gas release evaluated during a scoping phase of the PRA (cases designated by NG in this paper). A second series of cases for the noble-gas release analysis (labeled as high-release - HR) were also run to test the effects of an ~150 MW energy addition rate to the stacked plume beginning five minutes after scram. The second partial moderator loss analysis (Table 1b) models the average 2.2×10^{14} Bq release as a short 5-minute release, beginning 5 minutes after shutdown.

Transport away from the release point in all analyses will be conservatively fast in these calculations due to tritium assignment to the noble gas group of the MACCS standard nine-group input. The tritium species are modeled as noncondensible species. In reality, processes within the reactor building, and exchange processes in the environment will yield a predominantly condensible tritium form by the time the plume reaches the site boundary.

Table 1a Modified-MACCS Plume Bins for Postulated Moderator Tritium Release (Full Inventory Loss)

Bin No.	Time from Reactor Scram, (sec)		Plume Duration, (sec)		Tritium Fraction Stacked
1*	300	(5 min)	300	(5 min)	0.260
2*	300	(5 min)	600	(10 min)	0.190
3	1860	(31 min)	3960	(36 min)	0.460
4	2940	(49 min)	9000	(150 min)	0.082

* Accompanied by 79 MW Energy Addition Rate to the Plume in HR Cases

Table 1.b Modified-MACCS Plume Bins for Postulated Moderator Tritium Release (Partial Inventory Loss)

Bin No.	Time from Reactor Scram, (sec)		Plume Duration, (sec)		Tritium Fraction Stacked
1	300	(5 min)	300	(5 min)	0.0012

A set of five years of site meteorological data, 1982 through 1986 inclusive, has been characterized for use with the MACCS Latin hypercube sampling. These data, collected from on-site and off-site sampling points, were put in MACCS-required format.[9] Each year is represented by a file of hourly readings indicating day of the year, time, wind direction, wind speed, Pasquill Category (A-F), and accumulated precipitation. A final record specifies seasonal mixing layer heights for unstable and neutral conditions.

A comparison was made of the 5 years with respect to MACCS meteorological categories for rainfall, wind change, Pasquill-Gifford category, and wind speed annual averages. Although the modified-MACCS calculations could have used any one of the five annual meteorological files, only 1984 was avoided due to its unrepresentativeness with respect to average conditions (esp. lower rainfall, higher percentages in high wind speed categories). Most of the current work is based on 1982 conditions.

Off-site population distributions in 16-22.5° sectors about the SRP site were derived from an Oak Ridge National Laboratory code that generates multisector, multi-radial region populations as a function of distance from a user-specified site in the U.S., as applied by C.E. Bailey of SRL.[10] The resulting population bar chart out to 805 km (500 miles) is shown in Figure 3.

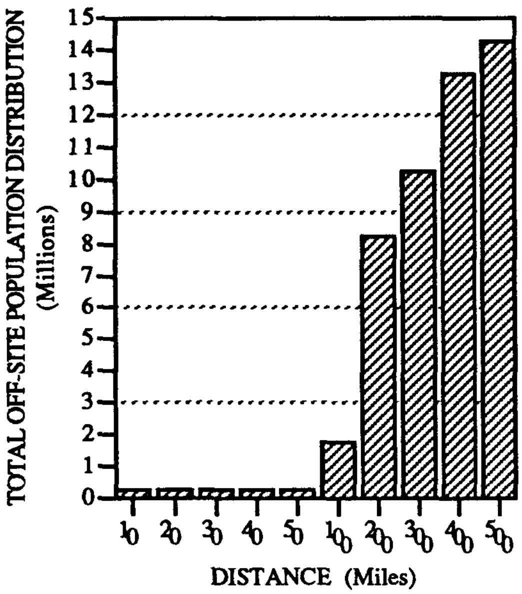

Figure 3. Off-Site Population Distribution

The source term for the loss-of-moderator reactor accident analysis considered here is assumed to be the entire tritium content of the process water system, released with the same release timing as noble gases (Xe, Kr) from the fuel assemblies would be released in a full core melt. The release history is conservative for tritium and will be refined over the balance of Level 3 of the PRA. A conservative estimate of tritium inventory of 1.85×10^{17}Bq (5MCi) is entered in the modified-MACCS input.[11]

CONDITIONAL CONSEQUENCES

The standard MACCS code and the version modified at SRL produce sets of early and late health effects on a scenario-by-scenario basis. These results, or consequences of the reactor accident are given probabilistically as complementary cumulative distribution functions (CCDFs). A CCDF gives the predicted frequency with which the corresponding magnitude is equaled or exceeded. Before an estimate of the frequency of the loss-of-moderator is applied, we shall use conditional CCDFs, i.e. CCDFs in which frequency is replaced by the probability given the occurrence of the accident sequence.

Two measures derived from the CCDF will be referred to in the balance of this paper. The first is the 50th percentile, or median, or magnitude of the consequences for which the probability of equaling or exceeding is 50%. The second is the 95th percentile, which is the magnitude of consequences for which the probability of exceedance is 5%.[12]

For moderator tritium analyses considered by the modified-MACCS code, only three types of measures of consequence are retained for the analysis. The others are either insignificant relative to the tritium inventory impact, or are identically zero. Those retained are: 1) cancer deaths from 0 to 805 km, given as total, lung, thyroid, breast, gastrointestinal, and leukemia; 2) population dose to red marrow in the regions bound by increasing radii, viz., 0 to 16 km, 0 to 32 km, 0 to 81 km, 0 to 161 km, 0 to 322 km, and 0 to 805 km; and 3) cancer fatality risk to the individual for 0 to 16 km or 0 to 32 km. For purposes of comparison with earlier studies, Runkle and Ostmeyer[6] allow that the red bone marrow dose equivalent can be used to replace the whole body dose for internal and external exposures.

Figure 4 is representative of the conditional CCDF output obtained in these analyses. The median conditional probability line indicates, from the intersection with no deposition (NG1) and with wet deposition (NG2) curves, that 5.3 and 11.5 Person-Sv, respectively may be expected to be exceeded 50% of the time. Similarly, 14 Person-Sv and 32 Person-Sv would be exceeded 5% of the time. Although each measure of consequence may be plotted as in Figure 4, we will present only median and 95th percentile information in tabular form for compactness.

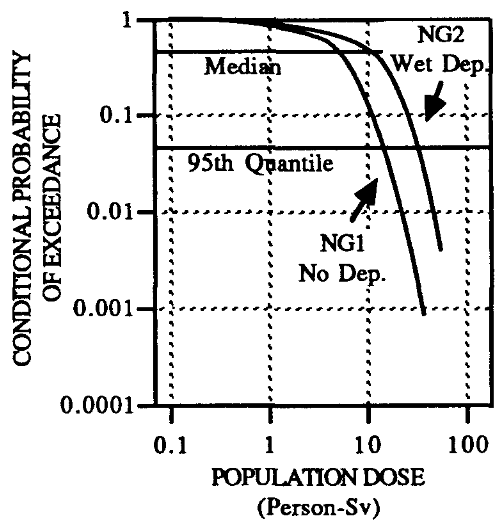

Figure 4. NG Cases - Tritium Release Population Dose to Red Marrow

Table 2a shows the comparison of off-site median and 95th percentile information for the three measures of consequence for the "noble gas" release profile. Cases NG1 and NG2 differ in that wet deposition may occur by rainfall in the latter; otherwise tritium in the plume has noble gas transport characteristics. Both sets of results are low from a PRA perspective. Comparatively, the effect of adding deposition to the global result (0 to 805 km) is to multiply the median by 2.1 to 2.5 and to about double the 95th percentile. At shorter distances, the effects are greater. The observed trend may be explained by noting the impact of rainfall on MACCS-generated plumes on the basis of the year-long set of input meteorological data. Rainfall occurring nearer to the release point will washout tritium-rich plumes and introduce a tritium component to the local population for inhalation and ingestion. This same component of the plume would have otherwise been transported to more distant points had wet deposition not been operative. In the NG cases, a quantitative measure of the importance of the deposition term may be gained by taking the ratio of case NG2 to NG1 for the median population dose to red marrow for the spatial intervals of (0 to 16 km / 0 to 161 km / 0 to 805 km). These ratios are (6.3 / 4.1 / 2.2).

Table 2a. Conditional Consequences for Full Moderator Loss

Measure of Consequence	Conditional Consequences Case NG1 Med.	Case NG1 95th	Case NG2 Med.	Case NG2 95th
Total Cancer Fatalities				
(0 to 805 km)	0.100	0.267	0.220	0.560
- Lung	0.010	0.028	0.024	0.057
- Thyroid	0.004	0.010	0.008	0.021
- Breast	0.031	0.086	0.070	0.164
- Gastrointestinal	0.030	0.081	0.066	0.161
- Leukemia	0.008	0.019	0.016	0.041
Total Cancer Fatalities				
(0 to 32 km)	1.3E-3	7.9E-3	7.4E-3	0.033
Population Dose (Sv) to Red Marrow				
(0 to 16 km)	4.8E-3	0.056	0.030	0.284
(0 to 32 km)	0.077	0.420	0.410	1.83
(0 to 81 km)	0.320	3.2	1.5	1.2E1
(0 to 161 km)	0.93	3.65	3.8	1.3E1
(0 to 322 km)	2.2	6.5	7.5	2.2E1
(0 to 805 km)	5.3	1.4E1	1.2E1	3.2E1
Individual Cancer Fatality Risk				
(0 to 32 km)	4.7E-8	2.5E-7	2.4E-7	1.1E-6

It was desirable to determine the sensitivity of the predicted consequences to the year of meteorological data used for modified-MACCS sampling. Case NG3 in Table 2b is identical to the input of NG2 with the exception of using a 1986 weather file for the SRP site. It is concluded by observation of the columns of data for case NG3 with the corresponding column data for case NG2 that there are no substantive differences in either median or 95th percentile for any of the relevant MACCS measures of consequence.

Table 2b. Conditional Consequences for Full Moderator Loss - Meteorological Sensitivity

Measure of Consequence	Conditional Consequences Case NG2		Case NG3	
	Med.	95th	Med.	95th
Total Cancer Fatalities				
(0 to 805 km)	0.220	0.560	0.213	0.544
- Lung	0.024	0.057	0.022	0.056
- Thyroid	0.008	0.021	0.008	0.021
- Breast	0.070	0.164	0.065	0.155
- Gastrointestinal	0.066	0.161	0.063	0.150
- Leukemia	0.016	0.041	0.015	0.039
Total Cancer Fatalities				
(0 to 32 km)	7.4E-3	0.033	0.011	0.035
Population Dose (Sv) to Red Marrow				
(0 to 16 km)	0.030	0.284	0.053	0.280
(0 to 32 km)	0.410	1.83	0.57	2.03
(0 to 81 km)	1.5	1.2E1	1.80	1.1E1
(0 to 161 km)	3.8	1.3E1	3.98	1.3E1
(0 to 322 km)	7.5	2.2E1	7.2	2.2E1
(0 to 805 km)	1.2E1	3.2E1	1.1E1	3.0E1
Individual Cancer Fatality Risk				
(0 to 32 km)	2.4E-7	1.1E-6	3.4E-7	1.2E-6

A similar comparison of deposition impacts is shown in Table 3 for an off-site population given the high-release scenario profile to the tritium release (cases HR1 and HR2). This variation from the NG class of cases is significant when calculated with the full source term including MACCS groups 5 through 9. However, with tritium alone the effects are minor: ~150 MW of energy is added to the tritium plume at the stack. Local populations are therefore subject to less of the tritium plume because of the buoyancy factor accompanying the plume. The net results are less radiation dose to the local population and fewer cancers induced due to the initial higher plume height. In absolute terms, the Table 3 conditional consequences are virtually the same as the NG cases.

Table 3. Conditional Consequences for Full Moderator Loss (Energy Addition to Plume)

Measure of Consequence	Conditional Consequences Case HR1 No Deposition		Case HR2 Wet Deposition	
	Med.	95th	Med.	95th
Total Cancer Fatalities				
(0 to 805 km)	0.100	0.267	0.246	0.590
- Lung	0.010	0.028	0.026	0.061
- Thyroid	0.004	0.010	0.009	0.023
- Breast	0.030	0.086	0.073	0.176
- Gastrointestinal	0.030	0.082	0.068	0.170
- Leukemia	0.007	0.020	0.018	0.043
Total Cancer Fatalities				
(0 to 32 km)	0.001	0.006	0.007	0.032
Population Dose (Sv) to Red Marrow				
(0 to 16 km)	0.004	0.040	0.027	0.211
(0 to 32 km)	0.065	0.347	0.383	1.88
(0 to 81 km)	0.312	3.13	1.56	1.3E1
(0 to 161 km)	0.939	3.64	4.36	1.5E1
(0 to 322 km)	2.19	6.46	8.01	2.4E1
(0 to 805 km)	5.29	1.4E1	1.3E1	3.3E1
Individual Cancer Fatality Risk				
(0 to 32 km)	4.1E-8	2.2E-7	2.3E-7	1.0E-6

The partial moderator loss conditional consequences are given in Table 4. The results of this analysis are approximately proportional to the tritium loss in the full moderator analyses. A ~$2^1/_2$ h release gave similar results to Table 4 showing insensitivity to tritium plume duration.

Table 4. Conditional Consequences From Average Moderator Leak - 2.22×10^{14} Bq

Measure of Consequence	Conditional Consequences (Median	95th Percentile)
Total Cancer Fatalities		
(0 to 805 km)	2.6E-4	7.4E-4
- Lung	2.7E-5	7.7E-5
- Thyroid	9.7E-6	2.5E-5
- Breast	8.6E-5	2.3E-4
- Gastrointestinal	8.4E-5	2.3E-4
- Leukemia	2.1E-5	5.6E-5
Total Cancer Fatalities		
(0 to 16 km)	7.0E-7	6.7E-6
(0 to 32 km)	1.0E-5	4.4E-5
Population Dose (Sv) to Red Marrow		
(0 to 16 km)	3.6E-5	3.4E-4
(0 to 32 km)	5.3E-4	2.2E-3
(0 to 81 km)	1.9E-3	1.7E-2
(0 to 161 km)	4.7E-3	2.1E-2
(0 to 322 km)	9.2E-3	3.0E-2
(0 to 805 km)	1.4E-2	3.7E-2
Individual Cancer Fatality Risk		
(0 to 16 km)	7.7E-10	7.8E-9
(0 to 32 km)	3.2E-10	1.3E-9

PRELIMINARY RISK DUE TO TRITIUM RELEASE

The risk in terms of some effect or consequence per reactor-year for this class of postulated tritium release ($3H_{rel}$) is developed formally from

$$\text{Risk} = \text{Partial Moderator Loss Risk} + \text{Full Moderator Loss Risk} \quad (1)$$

For the first term of Eqn. 1 above, the tritium release frequency is

$3H_{rel}$ Freq. = ~1 release/reactor-year, preliminary average leak frequency for small &

medium leaks.

and the conditional consequences are taken from Table 4. Similarly, the full moderator inventory loss of tritium is the product of

$3H_{rel}$ Freq. = $5x10^{-3}$ releases/reactor-year, preliminary full moderator leak frequency

and the conditional consequences found in Table 2 or 3.

Table 5 is the summed risk over small-to-medium moderator loss events and postulated full moderator losses. The partial loss (95th percentile) conditional consequences found in Table 4 multiplied by the appropriate frequency of 1.0 per reactor year are added to the full moderator risk calculated from wet-deposition case NG2 conditional consequences multiplied by $5x10^{-3}$ per reactor year.

Table 5. Risk to Off-site from Hypothetical Tritium Release from SRP Reactor Operation - Average Partial Moderator Loss Risk + Full Moderator Loss Risk

Measure of Consequence	Partial Moderator (Consequence / Rx-y) 95th	Full Loss 95th	Total Off-site Risk (Rx-y-1)
Total Cancer Fatalities			
(0 to 805 km)	7.4E-4	2.8E-3	3.5E-3
- Lung	7.7E-5	2.9E-4	3.7E-4
- Thyroid	2.5E-5	1.1E-4	1.4E-4
- Breast	2.3E-4	8.2E-4	1.0E-3
- Gastrointestinal	2.3E-4	8.1E-4	1.0E-3
- Leukemia	5.6E-5	2.1E-4	2.7E-4
Total Cancer Fatalities			
(0 to 16 km)	6.7E-6	2.8E-5	3.5E-5
(0 to 32 km)	4.4E-5	1.7E-4	2.1E-4
Population Dose (Sv) to Red Marrow			
(0 to 16 km)	3.4E-4	1.4E-3	1.7E-3
(0 to 32 km)	2.2E-3	9.2E-3	1.1E-2
(0 to 81 km)	1.7E-2	6.0E-2	7.7E-2
(0 to 161 km)	2.1E-2	6.6E-2	8.7E-2
(0 to 322 km)	3.0E-2	1.1E-1	1.4E-1
(0 to 805 km)	3.7E-2	1.6E-1	2.0E-1
Individual Cancer Fatality Risk			
(0 to 16 km)	7.8E-9	2.9E-8	3.7E-8
(0 to 32 km)	1.3E-9	5.5E-9	6.8E-9

Two NRC safety goals are typically addressed in determining the risks to off-site populations from reactor operation. These are: 1) the risk of prompt fatality to the average individual and 2) the risk of induced cancer to the average individual. Within the framework of calculations discussed above, no prompt fatalities were found for either the high release or noble gas release scenario.

A comparison can be made to the latent cancer safety objective. The NRC cancer objective is that the average risk of a cancer fatality in a populations near a nuclear power plant (defined to be within 16 km (10 miles)) be less than 0.1% of the cancer fatality risk from all other causes; this corresponds to a cancer risk of $2x10^{-6}$ per year.[13] The last two lines of entries show the analogous results for the 95th percentile-based cancer risk to the average individual located within 16 km and 32 km of the SRP release point. From 0 to 16 km, the risk is $4x10^{-8}$ per reactor year. From 0 to 32 km, the risk per individual is $7x10^{-9}$ per reactor year.

CONCLUSIONS

Preliminary risks associated with postulated partial and full loss-of-moderator accidents in a SRP reactor have been determined. Although these analyses are subject to further development, this work has produced the following insights:

1. The off-site risk is due largely to the contribution from the full moderator loss of $1.9x10^{17}$Bq. However, the component representing plant operating experience with small-to-medium leaks contributes ~25% to the risk.

2. Although finite, the calculated measures of risk determined in this study are significantly less than the NRC safety goals.

3. Initial adaptation of the MACCS computer code to handle off-site consequences due to tritium release can be viewed as an useful tool to provide conservative first estimates to support PRA studies. Tritium plume transport, isotopic exchange phenomena, and incorporation into the food-chain models may be desirable additions.

4. Off-site impacts, integrated over the 800 km distance covered in the MACCS data base, are proportional to tritium quantity released. The results are insensitive to the plume duration.

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