Introduction

As part of its generic study of spent fuel pool accidents, undertaken to develop generic, risk-informed regulatory requirements for plants that are being decommissioned, the Office of Nuclear Reactor Regulation (NRR) had requested the Office of Nuclear Regulatory Research (RES) to perform an evaluation of the offsite radiological consequences of a severe spent fuel pool accident. Accordingly, RES completed an in-house analysis of offsite radiological consequences, which included sensitivity and uncertainty analysis to assess the effect of critical parameters and assumptions. On May 25, 1999, RES forwarded to NRR a summary of the evaluation. A primary objective of the evaluation was to assess the effect of extended storage in a spent fuel pool, and the resulting radioactive decay, on offsite consequences. The evaluation showed about a factor-of-two reduction in prompt fatalities if the accident occurs after 1 year instead of after 30 days. The evaluation also showed that beginning evacuation three hours before the release begins reduces prompt fatalities by more than an order of magnitude.

The purpose of this report is to document the detailed technical basis of the offsite consequence evaluation. This report documents the offsite consequence calculations we performed using the MACCS code (MELCOR Accident Consequence Code System) and includes the input files used. In addition, this report documents follow-up calculations, performed since our earlier letter, to evaluate the importance of cesium to better understand why the consequence reduction from a year of decay was not greater. These follow-up calculations showed that cesium with its long half-life (30 years) is responsible for limiting the consequence reduction. For the population within 100 miles of the site, 97 percent of the societal dose was from cesium.

Previous Consequence Assessments

Spent fuel pool accidents involving a sustained loss of coolant have the potential for leading to significant fuel heat up and resultant release of fission products to the environment. Such an accident would involve decay heat raising the fuel temperature to the point of exothermic cladding oxidation, which would cause additional temperature escalation to the point of fission product release. However, because fuel in a spent fuel pool has a lower decay power than fuel in the reactor vessel of an operating reactor, it will take much longer for the fuel in the spent fuel pool to heat up to the point of releasing radionuclides than in some reactor accidents.

Earlier analyses in NUREG/CR-4982[1] and NUREG/CR-6451[2] have assessed the frequency and consequences of spent fuel pool accidents. These analyses included a limited evaluation of offsite consequences of a severe spent fuel pool accident. NUREG/CR-4982 results included consequence estimates for the societal dose for accidents occurring 30 days and 90 days after the last discharge of spent fuel into the spent fuel pool. NUREG/CR-6451 results included consequence estimates for societal dose, prompt fatalities, and cancer fatalities for accidents occurring 12 days after the last discharge of spent fuel. The work described in this current report extends the earlier analyses by calculating offsite consequences for a severe spent fuel pool accident occurring up to one year after discharge of the last load of spent fuel, and supplements that earlier analysis with additional sensitivity studies, including varying evacuation assumptions as well as other modeling assumptions. The primary objective of this analysis was to assess the effect of extended storage in a spent fuel pool, and the resulting radioactive

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decay, on offsite consequences. However, as part of this work, the sensitivity to a variety of other parameters was also evaluated.

The current analysis used the MACCS code[3] (version 2) to estimate offsite consequences for a severe spent fuel pool accident. Major input parameters for MACCS include radionuclide inventories, radionuclide release fractions, evacuation and relocation criteria, and population density. The specification of values for these input parameters for a severe spent fuel pool accident is discussed below.

Radionuclide Inventories

As discussed above, the current analysis was undertaken to assess the magnitude of the decrease in offsite consequences that could result from up to a year of decay in the spent fuel pool. To perform this work, it was necessary to have radionuclide inventories in the spent fuel pool for a decommissioned reactor at times up to 1 year after final shutdown. The inventories in the NUREG/CR-6451 analysis have not been retrievable, so those inventories could not be used. NUREG/CR-4982 contains spent fuel pool inventories for two operating reactors, a BWR (Millstone 1) and a PWR (Ginna). Because the current analysis may also be used as part of the probabilistic risk analysis of spent fuel pool accidents for the Susquehanna plant which is a BWR, the spent fuel inventories for Millstone 1 which is also a BWR were used for this analysis. These spent fuel pool inventories for Millstone 1 are given in Table 4.1 of NUREG/CR-4982 and are reproduced in Table A6-1 below. Two adjustments were then made to the Table A6-1 inventories. The first adjustment was to multiply the inventories by a factor of 1.7, because the thermal power of Susquehanna is 1.7 times higher than that of Millstone 1. The second adjustment, described in the next two paragraphs, was needed because NUREG/CR-4982 was for an operating reactor and this analysis is for a decommissioned reactor.

Because NUREG/CR-4982 was a study of spent fuel pool risk for an operating reactor, the Millstone 1 spent fuel pool inventories shown in Table A6-1 were for the fuel that was discharged during the 11th refueling outage (about 1/3 of the core) and the previous 10 refueling outages. The inventories shown in Table A6-1 did not include the fuel which remained in the vessel (about 2/3 of the core) that was used further when the reactor was restarted after the outage. Because the current study is for a decommissioned reactor, the inventories shown in Table A6-1 were adjusted by adding the inventories in the remaining 2/3 of the core. This remaining 2/3 of the core is expected to contain a significant amount of short half-life radionuclides in comparison with the 11 batches of spent fuel in the spent fuel pool.

The radionuclide inventories in the remaining 2/3 of the core were derived from the data in Tables A.5 and A.6 in NUREG/CR-4982. Tables A.5 and A.6 give inventory data for the 11th refueling outage. Table A.5 gives the inventories for the entire core at the time of reactor shutdown. Table A.6 gives the inventories (at 30 days after shutdown) for the batch of fuel discharged during the outage. First, the inventories for the entire core at the time of shutdown were reduced by radioactive decay to give the inventories for the entire core at 30 days after shutdown. Then, the inventories (at 30 days after shutdown) for the batch of fuel discharged were subtracted to give the inventories for the remaining 2/3 of the core at 30 days after shutdown. Inventories for the remaining 2/3 of the core at 90 days and 1 year after shutdown were subsequently calculated by reducing the 30-day inventories by radioactive decay.

Table A6-1 Radionuclide Inventories in the Millstone 1 Spent Fuel Pool

Radionuclide	Half-Life	Spent Fuel Pool Inventory (Ci)		
		30 days after last discharge	90 days after last discharge	1 year after last discharge
Co-58	70.9d	2.29E4	1.26E4	8.54E2
Co-60	5.3y	3.72E5	3.15E5	2.85E5
Kr-85	10.8y	1.41E6	1.39E6	1.33E6
Rb-86	18.7d	1.01E4	1.05E3	3.84E-2
Sr-89	50.5d	8.39E6	3.63E6	8.33E4
Sr-90	28.8y	1.42E7	1.42E7	1.39E7
Y-90	28.8y	1.43E7	1.42E7	1.39E7
Y-91	58.5d	1.18E7	5.75E6	2.21E5
Zr-95	64.0d	1.94E7	1.00E7	5.10E5
Nb-95	64.0d	2.54E7	1.70E7	1.11E6
Mo-99	2.7d	1.49E4	3.12E-3	0
Tc-99m	2.7d	1.43E4	3.01E-3	0
Ru-103	37.3d	1.53E7	5.21E6	4.07E4
Ru-106	1.0y	1.72E7	1.53E7	9.13E6
Sb-127	3.8d	8.21E3	1.39E-1	0
Te-127	109d	2.21E5	1.45E5	2.52E4
Te-127m	109d	2.18E5	1.48E5	2.57E4
Te-129	33.6d	2.74E5	7.79E4	2.68E2
Te-129m	33.6d	4.21E5	1.20E5	4.12E2
Te-132	3.2d	3.74E4	8.64E-2	0
I-131	8.0d	1.22E6	6.35E3	0
I-132	3.2d	3.85E4	8.90E-2	0
Xe-133	5.2d	7.29E5	2.30E2	0

Table A6-1 (continued) Radionuclide Inventories in the Millstone 1 Spent Fuel Pool

Radionuclide	Half-Life	Spent Fuel Pool Inventory (Ci)		
		30 days after last discharge	90 days after last discharge	1 year after last discharge
Cs-134	2.1y	7.90E6	7.47E6	5.80E6
Cs-136	13.2d	2.05E5	8.13E3	3.91E-3
Cs-137	30.0y	2.02E7	2.01E7	1.97E7
Ba-140	12.8d	5.19E6	1.90E5	6.41E-2
La-140	12.8d	5.97E6	2.19E5	7.37E-2
Ce-141	32.5d	1.32E7	3.61E6	1.03E4
Ce-144	284.6d	2.64E7	2.27E7	1.16E7
Pr-143	13.6d	5.44E6	2.41E5	1.90E-1
Nd-147	11.0d	1.54E6	3.36E4	1.10E-3
Np-239	2.4d	5.59E4	2.88E3	2.88E3
Pu-238	87.7y	4.51E5	4.53E5	4.54E5
Pu-239	24100y	8.89E4	8.89E4	8.89E4
Pu-240	6560y	1.30E5	1.30E5	1.30E5
Pu-241	14.4y	2.29E7	2.27E7	2.19E7
Am-241	432.7y	2.88E5	2.94E5	3.21E5
Cm-242	162.8d	1.45E6	1.12E6	3.50E5
Cm-244	18.1y	2.27E5	2.25E5	2.19E5

MACCS has a default list of 60 radionuclides that are important for offsite consequences for reactor accidents. NUREG/CR-4982 contains inventories for 40 of these 60 radionuclides. Of these 40 radionuclides, 27 have half-lives from 2.4 days to a year and 13 have half-lives of a year or greater as shown in Table A6-1. The half-lives of the remaining 20 radionuclides range from 53 minutes to 1.5 days as shown in Table A6-2. Because the largest half-life of these 20 radionuclides is 1.5 days, omitting these 20 radionuclides from the initial inventories used in the MACCS analysis should not affect doses from releases occurring after a number of days of decay.

Table A6-2 Half-lives of MACCS Radionuclides Whose Inventories Were Not in NUREG/CR-4982

Radionuclide	Half-Life (days)
Kr-85m	.19
Kr-87	.05
Kr-88	.12
Sr-91	.40
Sr-92	.11
Y-92	.15
Y-93	.42
Zr-97	.70
Ru-105	.19
Rh-105	1.48
Sb-129	.18
Te-131m	1.25
I-133	.87
I-134	.04
I-135	.27
Xe-135	.38
Ba-139	.06
La-141	.16
La-142	.07
Ce-143	1.38

Release Fractions

NUREG/CR-4982 also provided the fission product release fractions assumed for a severe spent fuel pool accident. These fission product release fractions are shown in Table A6-3. NUREG/CR-6451 provided an updated estimate of fission product release fractions. The release fractions in NUREG/CR-6451 (also shown in Table A6-3) are the same as those in NUREG/CR-4982, with the exception of lanthanum and cerium. NUREG/CR-6451 stated that the release fraction of lanthanum and cerium should be increased from $1x10^{-6}$ in NUREG/CR-4982 to $6x10^{-6}$, because fuel fines could be released offsite from fuel with high burnup. While RES believes that it is unlikely that fuel fines would be released offsite in any substantial amount, a sensitivity was performed using a release fraction of $6x10^{-6}$ for lanthanum and cerium to determine whether such an increase could even impact offsite consequences.

Table A6-3 Release Fractions for a Severe Spent Fuel Pool Accident

Radionuclide Group	Release Fractions	
	NUREG/CR-4982	NUREG/CR-6451
noble gases	1	1
iodine	1	1
cesium	1	1
tellurium	$2x10^{-2}$	$2x10^{-2}$
strontium	$2x10^{-3}$	$2x10^{-3}$
ruthenium	$2x10^{-5}$	$2x10^{-5}$
lanthanum	$1x10^{-6}$	$6x10^{-6}$
cerium	$1x10^{-6}$	$6x10^{-6}$
barium	$2x10^{-3}$	$2x10^{-3}$

Modeling of Emergency Response Actions and Other Areas

Modeling of emergency response actions was essentially the same as that used for Surry in NUREG-1150. The timing of events is given in Table A6-4. Evacuation begins exactly two hours after emergency response officials receive notification to take protective measures. This results in the evacuation beginning approximately .8 hours after the offsite release ends. Only people within 10 miles of the spent fuel pool evacuate, and, of those people, .5% do not evacuate. Details of the evacuation modeling are given in Table A6-5.

People outside of 10 miles are relocated to uncontaminated areas after a specified period of time depending on the dose they are projected to receive in the first week. There are two relocation criteria. The first criterion is that, if the dose to an individual is projected to be greater that 50 rem in one week, then the individual is relocated outside of the affected area after 12 hours. The second criterion is that, if the dose to an individual is projected to be greater that 25 rem in one week, then the individual is relocated outside of the affected area after 24 hours.

Table A6-4 Timing of Events

Event	Time (sec)	Time (hour)
notification given to offsite emergency response officials	0	0
start time of offsite release	2400	.7
end time of offsite release	4200	1.2
evacuation begins	7200	2.0

Table A6-5 Evacuation Modeling

Parameter	Value
size of evacuation zone	10 miles
sheltering in evacuation zone	no sheltering
evacuation direction	radially outward
evacuation speed	4 miles/hr
other	after evacuee reaches 20 miles from fuel pool, no further exposure is calculated

After the first week, the pre-accident population in each sector (including the evacuation zone) is assumed to be present unless the dose to an individual in a sector will be greater than 4 rem over a period of 5 years. If the dose to an individual in a sector is greater than 4 rem over a period of 5 years, then the population in that sector is relocated. Dose and cost criteria are used to determine when the relocated population returns to a sector. The dose criterion is that the relocated population is returned at a time when it is estimated that an individual's dose will not exceed 4 rem over the next 5 years. The actual population dose is calculated for exposure for the next 300 years following the population's return.

Offsite Consequence Results

MACCS calculations for a decommissioned reactor for accidents occurring 30 days, 90 days, and 1 year after final shutdown were performed to assess the magnitude of the decrease in the offsite consequences resulting from extended decay prior to the release. These calculations were performed for a Base Case along with a number of sensitivity cases to evaluate the impact of alternative modeling. These cases are summarized in Table A6-6. The results of these calculations are discussed below.

Table A6-6 Cases Examined Using the MACCS2 Consequence Code

Case	Population Distribution	Radionuclide Inventory	Evacuation Start Time	La/Ce Release Fraction	Evacuation Percentage
Base Case	Surry	11 batches plus rest of last core	1.4 hours after release begins	$1x10^{-6}$	99.5%
1	Surry	11 batches plus rest of last core	1.4 hours after release begins	$1x10^{-6}$	95%
2	Surry	11 batches	1.4 hours after release begins	$1x10^{-6}$	95%
3	100 people/mi^2	11 batches	1.4 hours after release begins	$1x10^{-6}$	95%
4	100 people/mi^2	11 batches plus rest of last core	1.4 hours after release begins	$1x10^{-6}$	95%
5	100 people/mi^2	11 batches plus rest of last core	3 hours before release begins	$1x10^{-6}$	95%
6	100 people/mi^2	11 batches plus rest of last core	3 hours before release begins	$6x10^{-6}$	95%
7	100 people/mi^2	11 batches plus rest of last core	3 hours before release begins	$1x10^{-6}$	99.5%

The Base Case was intended to model the offsite consequences for a severe spent fuel pool accident for a decommissioned reactor. To accomplish this, the Base Case used the Millstone 1 inventories from NUREG/CR-4982 adjusted for reactor power and the rest of the last core as discussed above. Accordingly, the Base Case used the Millstone 1 radionuclide inventories for the fuel from the first 11 refueling outages (1649 assemblies) together with the rest of the last core (413 assemblies). Because the Millstone 1 core design has 580 assemblies, the amount of fuel assumed to be in the spent fuel pool is equivalent to about 3.5 cores.

Other modeling in the Base Case, such as the population distribution, the evacuation percentage of 99.5% of the population, and the meteorology, are from the NUREG-1150 consequence assessment model for Surry. The input files for the Base Case are given in Appendix A. The results of the Base Case are shown in Table A6-7.

Table A6-7 Mean Consequences for the Base Case

Decay Time in Spent Fuel Pool	Distance (miles)	Prompt Fatalities	Societal Dose (person-Sv)	Cancer Fatalities
30 days	0-100	1.75	47,700	2,460
	0-500	1.75	571,000	25,800
90 days	0-100	1.49	46,300	2,390
	0-500	1.49	586,000	26,400
1 year	0-100	1.01	45,400	2,320
	0-500	1.01	595,000	26,800

Table A6-7 shows the offsite consequences for a severe spent fuel pool accident at 30 days, 90 days, and 1 year following final reactor shutdown. The decay times for fuel transferred to the pool during the 11th refueling outage were 30 days, 90 days, and 1 year, respectively. The decay times for spent fuel in the pool from earlier refueling outages were much longer and were accounted for in the inventories used in this analysis.

These results in Table A6-7 show virtually no change in long-term offsite consequences (i.e., societal dose and cancer fatalities) as a function of decay time, because they are controlled by inventories of radionuclides with long half-lives and relocation assumptions. However, these results also show about a factor-of-two reduction in the short-term consequences (i.e., prompt fatalities) from 30 days to 1 year of decay. (All of the prompt fatalities occur within 10 miles of the site.) As a rough check on the prompt fatality results, the change in decay power was evaluated for an operating reactor shut down for 30 days and for 1 year. The decay power decreased by about a factor of three. This is consistent with a factor-of-two decrease in prompt fatalities. The factor-of-three decrease in decay power by radioactive decay will also increase the time it takes to heat up the spent fuel, which provides additional time to take action to mitigate the accident.

The results of Case 1, which used a lower evacuation percentage than the Base Case, are identical to the results of the Base Case shown in Table A6-7. Case 1 used an evacuation percentage of 95%, while the Base Case used an evacuation percentage of 99.5%. Although it might be expected to see an increase in prompt fatalities from reducing the evacuation percentage, no such increase was observed. This is due to the assumption that the release ends at 1.2 hours, while the evacuation does not begin until 2 hours.

Case 2, shown in Table A6-8, used a radionuclide inventory that consisted of 11 batches of spent fuel, but did not include the remaining two-thirds of the core in the vessel. This was done to facilitate comparison of the consequence results with the results of the analyses in NUREG/CR-4982 and NUREG/CR-6451. This also allowed examination of the relative contribution of the short-lived radionuclides to consequences. Because the length of time between refueling outages is on the order of a year, short-lived radionuclides in the spent fuel pool will decay away between refueling outages. As a result, all of the short-lived radionuclides are in the core at the start of the 11th refueling outage for Millstone 1. When Millstone 1 discharged one-third of its core at the beginning of the 11th refueling outage, two-thirds of its short-lived isotopes remained in the vessel. Therefore, use of 11 batches of fuel in Case 2 without the remaining two-thirds of the core represents about a factor-of-three reduction in short-lived radionuclides in the spent fuel pool from what was modeled in Case 1. As shown in Table A6-8, use of 11 batches of spent fuel without the remaining two-thirds of the core resulted in a factor-of-two reduction in the prompt fatalities and no change in the societal dose and cancer fatalities. This factor-of-two reduction in prompt fatalities is consistent with the factor-of-three reduction in the inventories of the short-lived radionuclides when the remaining two-thirds of the core in the vessel is not included in the consequence calculation.

Table A6-8 Mean consequences for Case 2

Decay Time in Spent Fuel Pool	Distance (miles)	Prompt Fatalities	Societal Dose (person-Sv)	Cancer Fatalities
30 days	0-100	.89	44,900	2,280
	0-500	.89	557,000	25,100
90 days	0-100	.78	44,500	2,250
	0-500	.78	554,000	25,000
1 year	0-100	.53	43,400	2,180
	0-500	.53	567,000	25,500

The results of the next case, Case 3, are shown in Table A6-9. This case used a generic population distribution of 100 persons/mile^2 (uniform). This was done to facilitate comparison of the consequence results with the results of the analyses in NUREG/CR-4982 and NUREG/CR-6451. Use of a uniform population density of 100 persons/mile^2 results in an order-of-magnitude increase in prompt fatalities and relatively small changes in the societal dose and cancer fatalities.

Table A6-9 Mean Consequences for Case 3

Decay Time in Spent Fuel Pool	Distance (miles)	Prompt Fatalities	Societal Dose (person-Sv)	Cancer Fatalities
30 days	0-100	11.7	50,100	2,440
	0-500	11.7	449,000	20,300
90 days	0-100	10.6	50,300	2,460
	0-500	10.6	447,000	20,200
1 year	0-100	8.19	49,000	2,380
	0-500	8.19	453,000	20,500

The results of the next case, Case 4, are shown in Table A6-10. This case includes the remaining two-thirds of the core in the vessel. This was done to facilitate comparison of the consequence results with the results of the analysis in NUREG/CR-6451. As discussed above in the comparison of Case 1 with Case 2, this increases the prompt fatalities by about a factor of two with no change in the societal dose or cancer fatalities.

Table A6-10 Mean Consequences for Case 4

Decay Time in Spent Fuel Pool	Distance (miles)	Prompt Fatalities	Societal Dose (person-Sv)	Cancer Fatalities
30 days	0-100	18.3	53,500	2,610
	0-500	18.3	454,000	20,600
90 days	0-100	16.3	52,100	2,560
	0-500	16.3	465,000	21,100
1 year	0-100	12.7	50,900	2,490
	0-500	12.7	477,000	21,600

Heat up of fuel in a spent fuel pool following a complete loss of coolant takes much longer than in some reactor accidents. Therefore, it may be possible to begin evacuating before the release begins. Case 5, which uses an evacuation start time of three hours before the release begins, was performed to assess the impact of early evacuation. As shown in Table A6-11, prompt fatalities were significantly reduced and societal dose and cancer fatalities remained unchanged.

Table A6-11 Mean Consequences for Case 5

Decay Time in Spent Fuel Pool	Distance (miles)	Prompt Fatalities	Societal Dose (person-Sv)	Cancer Fatalities
30 days	0-100	.96	48,300	2,260
	0-500	.96	449,000	20,200
90 days	0-100	.83	47,500	2,220
	0-500	.83	460,000	20,700
1 year	0-100	.67	46,700	2,180
	0-500	.67	473,000	21,300

As noted above, NUREG/CR-6451 estimated the release of lanthanum and cerium to be a factor of six higher than that originally estimated in NUREG/CR-4982. Case 6 was performed to assess the potential impact of that higher release. The Case 6 consequence results were identical to those of Case 5 shown in Table A6-11. Therefore, even it were possible for fuel fines to be released offsite, there would be no change in offsite consequences as a result.

The final case, Case 7 was performed to examine the impact of a 99.5% evacuation for a case with evacuation before the release begins. This sensitivity (see Table A6-12) showed an order of magnitude decrease in the prompt fatalities. Again, as expected, no change in the societal dose or cancer fatalities was observed.

Table A6-12 Mean Consequences for Case 7

Decay Time in Spent Fuel Pool	Distance (miles)	Prompt Fatalities	Societal Dose (person-Sv)	Cancer Fatalities
30 days	0-100	.096	48,100	2,250
	0-500	.096	449,000	20,200
90 days	0-100	.083	47,400	2,210
	0-500	.083	460,000	20,700
1 year	0-100	.067	46,600	2,170
	0-500	.067	473,000	21,300

Comparison with Earlier Consequence Analyses

As a check on the above calculations and to provide additional insight into the consequence analysis for severe spent fuel pool accidents, the above calculations were compared to the consequence results reported in NUREG/CR-4982 and NUREG/CR-6451. Table A6-13 shows the analysis assumptions used for BWRs in these earlier reports together with those of Cases 3 and 4 of the current analysis.

NUREG/CR-4982 results included consequence estimates for societal dose for an operating reactor for severe spent fuel pool accidents occurring 30 days and 90 days after the last discharge of spent fuel into the pool. The Case 3 results were compared against the NUREG/CR-4982 results, because they use the same population density (100 persons/mile^2) and 11 batches of spent fuel in the pool. However, one difference is that Case 3 uses a radionuclide inventory that is a factor of 1.7 higher than NUREG/CR-4982 to reflect the relative power levels of Susquehanna and Millstone 1. Therefore, Case 3 was rerun with the radionuclide inventory of NUREG/CR-4982. As shown in Table A6-14, the Case 3 rerun results generally compared well with the NUREG/CR-4982 results.

Table A6-13 Comparison of Analysis Assumptions

Parameter	NUREG/CR-4982 (BWR)	NUREG/CR-6451 (BWR)	Case 3	Case 4
population density (persons/ mile^2)	100	0-30 mi: 1000 30-50 mi: 2300 (city of 10 million people, 280 outside of city) 50-500 mi: 200	100	100
meteorology	uniform wind rose, average weather conditions	representative for continental U.S.	Surry	Surry
radionuclide inventory	11 batches of spent fuel	full fuel pool after decommissioning (3300 assemblies)	11 batches of spent fuel, increased by x1.7	11 batches of spent fuel plus last of rest core, increased by x1.7
exclusion area	not reported	.4 mi	none	none
emergency response	relocation at one day if projected doses exceed 25 rem	relocation at one day if projected doses exceed 25 rem	NUREG-1150 Surry analysis (see above)	NUREG-1150 Surry analysis (see above)

Table A6-14 Comparison with NUREG/CR-4982 Results

Decay Time in Spent Fuel Pool	Distance (miles)	Societal Dose (person-Sv)		
		NUREG/CR-4982	Case 3	Case 3 Rerun
30 days	0-50	26,000	20,900	16,700
	0-500	710,000	449,000	379,000
90 days	0-50	26,000	20,400	16,500

The NUREG/CR-6451 results included consequence estimates for societal dose, cancer fatalities, and prompt fatalities for a decommissioned reactor for a severe spent fuel pool accident occurring 12 days after the final shutdown. The Case 4 results for 30 days after final shutdown were compared against the NUREG/CR-6451 results, because (1) they included the entire last core in the spent fuel pool and (2) Case 4 had a uniform population density which could be easily adjusted to approximate that in NUREG/CR-6451. Differences between Case 4 and NUREG/CR-6451 included the population density, the amount of spent fuel in the pool, and the exclusion area size. To provide a more consistent basis to compare the NUREG/CR-6451 results with the Case 4 results, Case 4 was rerun using population densities, an amount of spent fuel, and an exclusion area size similar to NUREG/CR-6451.

The average population densities in the NUREG/CR-6451 analysis were about 1800 persons/mile^2 within 50 miles and 215 persons/mile^2 within 500 miles. Also, NUREG/CR-6451 used an inventory with substantially higher quantities of long-lived radionuclides than the 11 batches of spent fuel in NUREG/CR-4982. NUREG/CR-6451 stated that it used an inventory of Cs-137 (30 year half-life) that was three times greater than that used in NUREG/CR-4982. To provide a more consistent basis to compare with NUREG/CR-6451 long-term consequences, Case 4 was rerun using uniform population densities of 1800 persons/mile^2 within 50 miles and 215 persons/mile^2 outside of 50 miles and a power correction factor of 3 instead of 1.7. As shown in Table A6-15, Case 4 rerun is in generally good agreement with NUREG/CR-6451. These calculations indicate a very strong dependence of long-term consequences on population density. Remaining differences in long-term consequences may be due to remaining differences in population density and inventories as well as differences in meteorology and emergency response.

Table A6-15 Comparison with NUREG/CR-6451 Results (long-term consequences)

Dist. (miles)	Societal Dose (person-Sv)			Cancer Fatalities		
	NUREG/ CR-6451	Case 4	Case 4 Rerun	NUREG/ CR-6451	Case 4	Case 4 Rerun
0-50	750,000	23,600	389,000	31,900	1,260	20,800
0-500	3,270,000	454,000	1,330,000	138,000	20,600	44,900

To provide a more consistent basis to compare with NUREG/CR-6451 short-term consequences, Case 4 was again rerun, this time using a uniform population density of 1000 persons/mile^2 and an exclusion area of .32 miles. As shown in Table A6-16, Case 4 rerun is in generally good agreement with NUREG/CR-6451. Overall, these calculations indicate a very strong dependence of short-term consequences on population density and a small dependence (about 10% change in prompt fatality results) on exclusion area size. Remaining differences in short-term consequences may be due to remaining differences in population density and inventories as well as differences in meteorology and emergency response.

Table A6-16 Comparison with NUREG/CR-6451 Results (short-term consequences)

Dist. (miles)	Prompt Fatalities		
	NUREG/CR-6451	Case 4	Case 4 Rerun
0-50	74	18.3	168
0-500	101	18.3	168

Effect of Cesium

Cesium is volatile under severe accident conditions and was previously estimated to be completely released from fuel under these conditions. Also, the half-lives of the cesium isotopes are 2 years for cesium-134, 13 days for cesium-136, and 30 years for cesium-137. Therefore, we performed additional sensitivity calculations on the Base Case to evaluate the importance of cesium to better understand why the consequence reduction from a year of decay was not greater. The results of our calculations are shown in Table A6-17. As shown in this table, we found that the cesium isotopes with their relatively long half-lives were responsible for limiting the reduction in offsite consequences.

Table A6-17 Mean Consequences for the Base Case with and Without Cesium

Decay Time in Spent Fuel Pool	Distance (miles)	Prompt Fatalities	Societal Dose (person-Sv)	Cancer Fatalities
1 year	0-100	1.01	45,400	2,320
1 year (without cesium)	0-100	0.00	1,460	42

Conclusion

The primary objective of this evaluation was to assess the effect of extended storage in a spent fuel pool, and the resulting radioactive decay, on offsite consequences of a severe spent fuel pool accident at a decommissioned reactor. This evaluation was performed in support of the NRR generic evaluation of spent fuel pool risk that is being performed to support related risk-informed requirements for decommissioned reactors. This evaluation showed about a factor-of-two reduction in prompt fatalities if the accident occurs after 1 year instead of after 30 days. Sensitivity studies showed that cesium with its long half-life (30 years) is responsible for limiting the consequence reduction. For the population within 100 miles of the site, 97 percent of the societal dose was from cesium. Also, this evaluation showed that beginning evacuation three hours before the release begins reduces prompt fatalities by more than an order of magnitude.

References

(1) NUREG/CR-4982, Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82, July 1987.

(2) NUREG/CR-6451, A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants, August 1997.

(3) NUREG/CR-6613, Code Manual for MACCS2, May 1998.