Revised and gave this revision to G. Hubbard at 11:00 a.m on 2/9/00, per his request. 2/9/00

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.

representative   with a thermal power of 3441 MWt   was originally intended to

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 (a) part of ~~the~~ probabilistic risk analysis of spent fuel pool accidents for ~~the Susquehanna plant[2] 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 1 below. Two adjustments were then made to the Table 1 inventories. The first adjustment was to multiply the inventories by a factor of 1.7, because the thermal power of ~~Susquehanna[2]~~ 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.

the representative BWR

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 1 were for the fuel that was discharged during the 11[th] refueling outage (about 1/3 of the core) and the previous 10 refueling outages. The inventories shown in Table 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 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 11[th] 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.

F/20

identical to those of Case 5 shown in Table 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 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.

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

Table 12. Mean consequences for Case 7.

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 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~~ the representative BWR and Millstone 1. Therefore, Case 3 was rerun with the radionuclide inventory of NUREG/CR-4982. As shown in Table 14, the Case 3 rerun results generally compared well with the NUREG/CR-4982 results.

```
RDCORINV037     I-135          0.000E+00
RDCORINV038     Xe-133         6.600E+16
RDCORINV039     Xe-135         0.000E+00
RDCORINV040     Cs-134         3.810E+17
RDCORINV041     Cs-136         1.610E+16
RDCORINV042     Cs-137         8.580E+17
RDCORINV043     Ba-139         0.000E+00
RDCORINV044     Ba-140         6.100E+17
RDCORINV045     La-140         6.230E+17
RDCORINV046     La-141         0.000E+00
RDCORINV047     La-142         0.000E+00
RDCORINV048     Ce-141         1.550E+18
RDCORINV049     Ce-143         0.000E+00
RDCORINV050     Ce-144         2.350E+18
RDCORINV051     Pr-143         5.910E+17
RDCORINV052     Nd-147         1.770E+17
RDCORINV053     Np-239         6.470E+15
RDCORINV054     Pu-238         1.760E+16
RDCORINV055     Pu-239         3.870E+15
RDCORINV056     Pu-240         5.400E+15
RDCORINV057     Pu-241         9.470E+17
RDCORINV058     Am-241         1.080E+16
RDCORINV059     Cm-242         7.320E+16
RDCORINV060     Cm-244         8.700E+15
*
*  SCALING FACTOR TO ADJUST THE CORE INVENTORY FOR POWER LEVEL
*
RDCORSCA001   1.711  *  convert from Millstone to Susquehanna 3441 MWt BWR
*                       by multiplying by ratio of powers
*                       (3441MWt/2011MWt)
*
*
RDAPLFRC001   PARENT     (apply rel fracs the same as prior versions)
*
* RELEASE FRACTIONS FOR ISOTOPE GROUPS IN RELEASE
*
*  ISOTOPE GROUPS:
*
*           XE/KR    I      CS     TE     SR     RU     LA     CE     BA
*
RDRELFRC001 1.0E+0 1.0E+0 1.0E+0 2.0E-2 2.0E-3 2.0E-5 1.0E-6 1.0E-6 2.0E-3
****************************************************************************
* OUTPUT CONTROL DATA BLOCK, LOADED BY INPOPT, STORED IN /STOPME/, /ATMOPT/
*
* FLAG TO INDICATE THAT THIS IS THE LAST PROGRAM IN THE SERIES TO BE RUN
*
OCENDAT1001  .FALSE. (SET THIS VALUE TO .TRUE. TO SKIP EARLY AND CHRONC)
*
OCIDEBUG001  0
*
* NAME OF THE NUCLIDE TO BE LISTED ON THE DISPERSION LISTINGS
*
OCNUCOUT001  Cs-137
*
*             NUM0
TYPE0NUMBER     2
*
*            INDREL   INDRAD
TYPE0OUT001    1        9
TYPE0OUT002    1        10        XCCDF
****************************************************************************
* METEOROLOGICAL SAMPLING DATA BLOCK
*
* METEOROLOGICAL SAMPLING OPTION CODE:
```