8.0 ACCIDENT SAFETY ANALYSIS

8.1 INTRODUCTION

This section contains an analysis of postulated accidents in terms of the causes of such events, the consequences, and the ability of the GE-Morris Operation (GE-MO) organization to cope with each situation.

The function of GE-MO is to store and ship irradiated nuclear fuel. A primary requirement of these operations is to protect the public and employees from excessive exposure to ionizing radiation, as specified by the requirements of 10 CFR 72.106. Specifically, any individual at or beyond the controlled area boundary shall not receive a dose greater than 5 Rem to the whole body or any organ from any design basis accident (i.e., those accidents described in this section).

8.1.1 Release Pathways

Exposure of the public and employees might result from postulated accidents, by direct radiation from the fuel, by airborne release of radioactive material, or by release of radioactive material to groundwater. These postulated events are discussed in this section. None of these potential releases have off-site impacts which exceed the limitations of 10 CFR 72.104.

8.1.1.1 Direct Radiation

Exposure of the public and employees could be postulated to result from direct radiation from fuel in storage or by release of radioactive material to the environs. Direct radiation from the fuel would occur only if the water level in the storage basin became too low to provide adequate shielding. This would pose a hazard to persons only if they were in relatively close proximity to the basin. Loss of water could result from postulated drainage or evaporation of the basin water, but only when basin make-up water supply quantity or rate is not sufficient to keep up with the water loss. Sudden draining of water from the basin is not credible.

8.1.1.2 Airborne Release

Airborne release of radioactive material could result from fuel being mechanically damaged sufficiently to release fission gases from the plena of fuel rods. Of the gases released, only Kr-85 and I-129 would be of concern.

No mechanism exists in the fuel storage environment to cause an airborne release of particulate radioactive material in quantities sufficient to result in exposures approaching limits specified in 10 CFR 72.104. During certain cask operations (e.g., decontamination and venting) particulate releases might occur but in very small quantities, even under the most severe conditions that can be postulated. These quantities would be much too small for an off-site impact. A criticality incident could result from the dropping of a basket in such a way that all the fuel falls out of the basket and comes to rest in a critical array, or by the deformation of fuel baskets into a critical array by a tornado-generated missile. In reality, however, the above events have an extremely low probability of occurring and the impact of either would be substantially less than the limits of Part 72.104.

8.1.1.3 Waterborne Release

Vault intrusion water is normally disposed of in the sanitary lagoons, so that an off-site release would not be likely even in the unlikely event the water is contaminated.

Water from the storage basins can be released due to a leak in the basin structure, permitting water to escape to the surrounding rock.

8.1.2 Accident Description/Discussion

The following sections contain discussion of various postulated accidents and estimates of the quantity of radioactive material release and projected consequences. A summary of events resulting in postulated radiation exposures to the public is shown in Figure 8-1. No combination of normal and credible accident events has been developed that would result in an off-site release or direct radiation exposure that would exceed the regulatory limits for an accident (10 CFR 72.106).

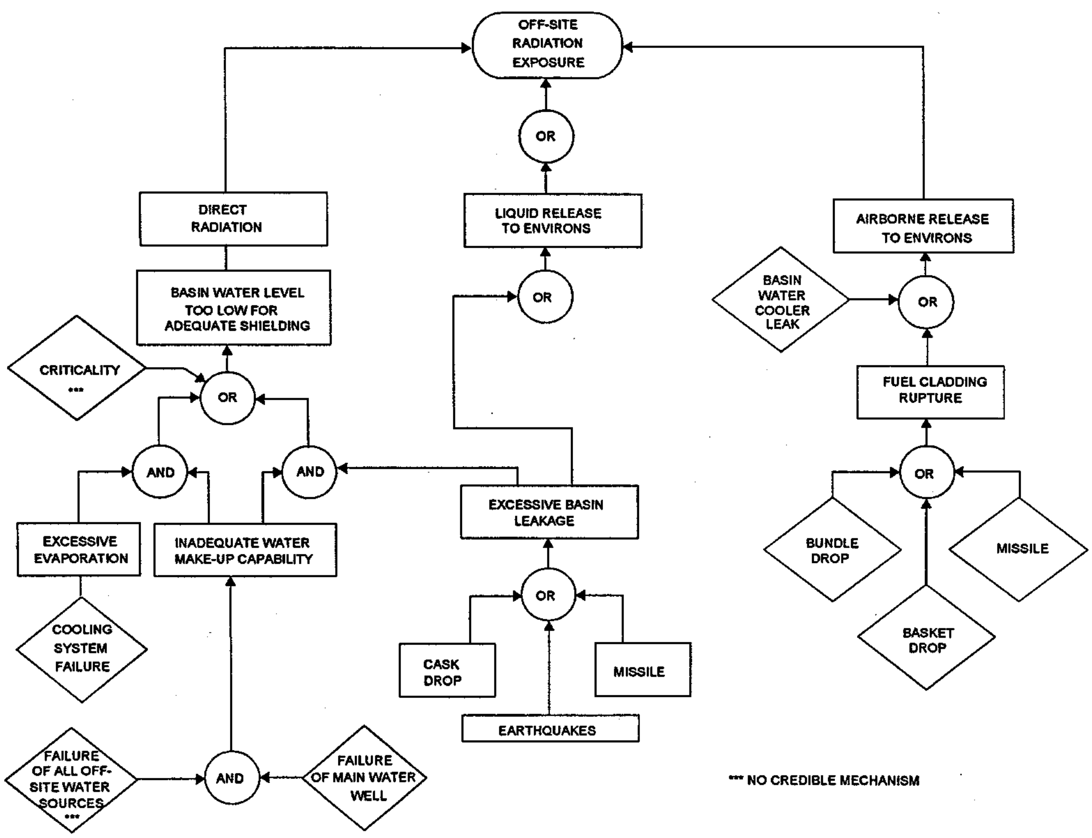

Figure 8-1. Event Diagram of Postulated Accidents

A release of noble gases and halogens from DNPS, similar to or greater than at TMI-2, would not affect fuel storage safety at Morris. The location and construction of the GE-MO control room, the availability of respiratory protective masks and systems, the availability of protective clothing, and other radiological emergency preparations at Morris would minimize the impact on GE-MO of any release from DNPS[1]. Even if it should become necessary to temporarily evacuate GE-MO, the slow loss of basin water by evaporation and the ease of replacement negates possible detrimental effects, and protects the public health and safety.

8.1.3 Exposure Paths

Of the possible exposure paths, only whole body exposure from external radiation and internal exposure through inhalation are considered credible at any off-site location. No mechanism has been identified that will cause radioactive contamination of farmlands, feed lots, or other sensitive areas, that could result in an ingestion dose greater than a small fraction of regulatory limits.

8.2 LOSS OF FUEL BASIN COOLING

The basin cooling system is not critical to safety. When the cooling system is not in service, the water make-up system can be used to replace water lost by evaporation. Even if the water make-up system is out of service, there is adequate time to repair or replace both cooling and make-up systems or to provide make-up water from alternate on-site or off-site sources. (The water make-up system includes the water well and all equipment related to the normal make-up water supply to the basin.)

The time available to provide make-up water if the cooling and water make-up systems are out of service has been determined by measurement of evaporative losses with the fuel in storage as of June 2004. Based on actual measurement of basin heat-up rate, the time available to provide make-up water before reaching the technical specification (Section 10, ¶ 10.3.1) limit of 9 feet of water above the top of the fuel bundle upper tie plate is more than 60 days.

8.2.1 Basin Water Temperature

Maximum basin water temperature as measured in June 2004 after 60 days of operation with no cooling or makeup water was 123º F and more than 319,263 gallons of water would have to evaporate before the top of the fuel bundles upper tie plate would be exposed. This would require approximately 150 days.

The probability of excessively high radiation dose rates resulting from loss of fuel basin cooling is clearly quite small given ample time for repairs and water replacement.

8.3 DRAINAGE OF FUEL BASINS

There are no piping penetrations which could drain the fuel storage basins and there are no paths for siphoning water from the basin. Therefore, to inadvertently drain water from the basin, the basin structure must be penetrated. Since the basin structure is below grade and given low permeability of surrounding rock (except for the overburden) and high level of upper strata groundwater, leakage (even if it were undetected) would not uncover the fuel (Appendix A.13).

8.3.1 Basin Liner rupture Experience

An accident occurred in June 1972 that resulted in the rupture of the basin liner and demonstrated the ability of GE-MO to withstand and recover from such an incident. No measurable exposure to ionizing radiation was experienced by site personnel or the general public as a result of the incident and no groundwater contamination above background levels was detected.

8.4 CASK DROP INTO THE CASK UNLOADING BASIN

A postulated means of damaging the basin floor structure is dropping a shipping cask on either the cask unloading pit set off shelf or the floor.

The cask unloading pit set off shelf is protected by an energy absorbing pad designed to accommodate the impact of a cask. Detailed design analysis of the pad is given in Appendix A. Included in that appendix is an analysis of an impact on the corner of the shelf and an impact on the floor of the cask unloading pit. In each case, it is shown the integrity of the structure is not breached and in neither case is basin water released to the environs. Rapid recovery from a breach in the liner caused by a cask incident is discussed in Section 8.3.1.

8.5 FUEL DROP ACCIDENTS

Accidents could occur during fuel handling that might result in mechanical damage to the fuel and subsequent release of fission gases. Such accidents could happen during transfer of fuel from a storage basket to a cask, or during transfer of storage basket from basin to unloading pit. In any case, the postulated accident is assumed to occur in the fuel unloading pit since the fuel is lifted to greater height than in the storage basins.

During cask handling operations, there is no movement of a cask over fuel. The design of the fuel storage facility is such that a cask cannot be moved over the fuel storage basins. Further, administrative controls prevent cask movement when fuel is present in the unloading pit.

The following discussion addresses the fission gas inventory in the fuel, water decontamination factors, and assumptions that pertain to both fuel drop and basket drop analyses.

a. Fission Gas Inventory in the Fuel

   Fission gas inventory in the fuel is dependent primarily on the total fuel exposure. Of the radioisotopes present in the fission gas inventory, Kr-85 and I-129 represent the greatest curie inventory in fuel that has cooled 1 year or more. Figure 8-2 depicts the Kr-85 inventory as a function of cooling times for different fuel exposure levels. Amounts of I-129 in the fuel range from about 0.008 Ci/TeU for 8,000 MWd/TeU exposure to 0.04 Ci/TeU for exposure of 44,000 MWd/TeU and remain essentially constant with time.

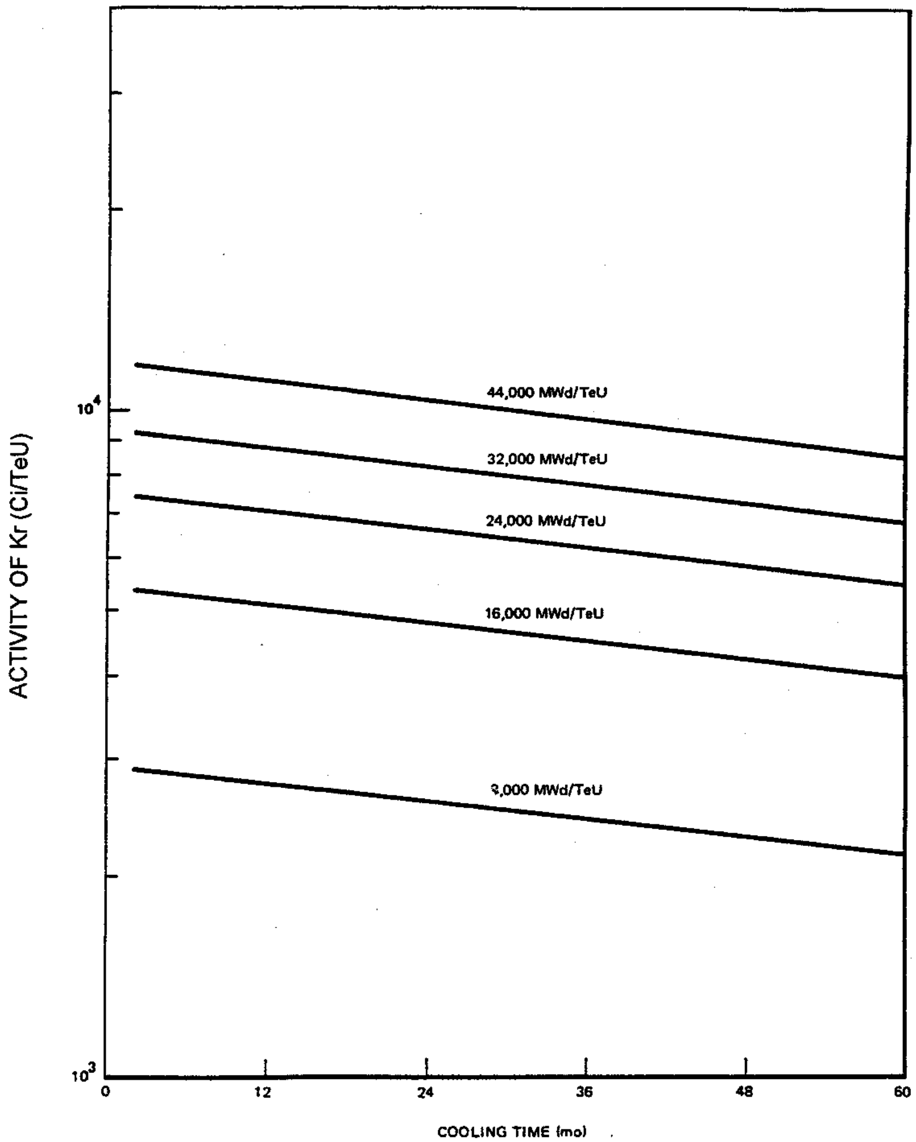

Figure 8-2. Kr-85 Activity as function of cooling time for different fuel exposures. (Total inventory in fuel rod.)

Other fission gases, including I-131, Xe-131m and Xe-133, decay relatively quickly. After one year cooling time, all three are decayed to insignificant levels as shown in Figure 8-3. The total fission gas inventory for a 1 year cooling time is given in Table 4-1, Section 4.

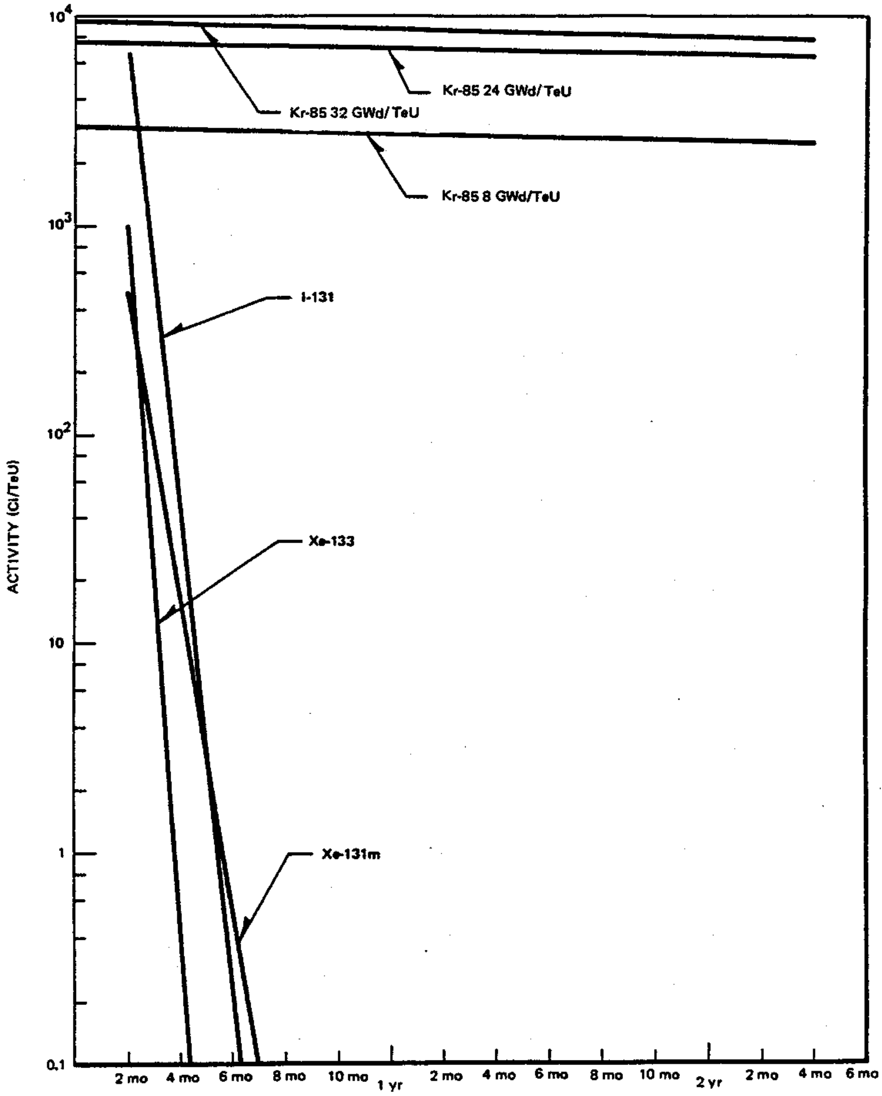

Figure 8-3. Iodine, Krypton and Xenon Decay

The amount of fission gas released from UO^2 fuel and accumulated in the plenum of each rod is dependent on the specific power (fuel temperature) during operation. At higher specific power, a greater fraction of gas will be released to the plenum. Calculations of fission gas inventory result in a release fraction that ranges from 20% to 45% depending on the irradiation history of the fuel rods. For example, a Westinghouse safety analysis report states that approximately 2.5% of Xe and approximately 3% of iodine are found in the gas plenum (Docket 50-295, "Zion Nuclear Power Station," Commonwealth Edison Co.).

GE uses plenum percentages for radioisotopes that are based on fission product release data from defective fuel experiments[2]. A comparison of these values with the NRC Regulatory Guide and the values used in the fuel drop analysis for GE-MO is shown below:

	GE Fuel Drop Analyses for Reactors	NRC Regulatory Guide	GE Fuel Drop Analyses For Morris Op
	PERCENT OF RADIOISOTOPES(S) IN PLENUM		
Radioiodine			
I-131	1.2	10	2
Kr-85	30	30	30
All other noble gases		10	
Xe-131m	3.9		
Xe-133	2.5		

These values are considered realistic values based on the analytical and experimental data contained in the references cited above. The value for radioiodine is also recommended by Appendix VIII, WASH-1400. The Kr value agrees with that in Regulatory Guide (RG) 1.25.

b. Water Decontamination Factor

Not all iodine released from a fuel rod would be released from the basin water. Being highly soluble, much of the iodine would dissolve and remain in the water. RG 1.25 recommends a factor of 100 for pool decontamination of iodine.

In analysis of a fuel handling accident, Westinghouse based decontamination factors on iodine tests conducted to determine the mass transfer from the gas phase to surrounding liquid[3]. That work resulted in the formulation of a mathematical expression for the iodine decontamination factor in terms of bubble size and bubble rise time. The equation is:

Decontamination Factor = (7.3) exp [0.313 t/d]

where t = rise time, and
d = effective bubble diameter.

Evaluating the decontamination factor for iodine released from a fuel bundle, a minimum factor of 760 is calculated for a water depth of 26 ft. However, for their "conservative analysis" the factor was reduced to 500.

For a fuel bundle drop at GE-MO, the worst-case accident occurs in the cask unloading pit. Minimum water depth in that pit is about 32 feet. Therefore, a decontamination factor of 500 is sufficiently conservative.

c. Assumptions

The following assumptions are made for the safety analysis:

1. The fuel bundle or basket drop occurs in the fuel unloading pit.

2. Because of the negligible particulate activity available for release from the fuel plena, none of the solid fission products are released.

3. The overall effective decontamination factor for iodine is 500. Because water has a negligible effect on removal of the noble gases, the decontamination factor is 1.

4. Ventilation air flow exhaust rate from the basin areas is 7,600 scfm via the air tunnel, sand filter and the main stack. Duration of release is 2 hours.

5. Worst case $\overline{X}/Q$ is 2.8 x 10^{-5} sec/m^3. (See Appendix A.5, Section A.5.1b, Short-Term (Accident) Diffusion Estimates.)

6. Fuel characteristics are 44,000 MWd/TeU exposure, 1-year cooling.

7. Dose conversion factors are:

Species	Whole Body $\frac{mRem}{\mu Ci} - \frac{m^3}{sec}$	Thyroid $\frac{mRem}{\mu Ci} - \frac{m^3}{sec}$
Noble Gas	4.75 x 10^{-7}	-
Halogen	8.72 x 10^{-5}	4.472 x 10^{-1}

8.5.1 Fuel Bundle Drop Accident

a. It is highly unlikely fuel rods would be ruptured in a fuel drop accident. However, to establish an upper boundary in the consequence analysis, it is assumed all rods in the bundle have ruptured releasing all fission gases present in the plena to the basin. The following release is calculated:

	Amounts Released (Ci)	
Species	BWR	PWR
Noble Gases	684	1530
Iodine	3.3E-7	0.48E-7

It is assumed that all of the fission gases are expelled from the basin and passed through the sand filter and released from the main stack.

Using the assumed values for atmospheric diffusion and dose conversion factors, the maximum off-site dose rates are:

	Maximum Dose Rate (mRem/hr)	
Body Organ	BWR	PWR
Whole Body	4.5E-3	1.0E-2

Thyroid	1.8E-6	4.0E-6

If an individual off-site were exposed at the maximum dose rate for the duration of the accident (2 hr.), the maximum doses are estimated to be about 0.02 mRem whole body and 8.0 x 10^{-6} mRem thyroid. Such doses are clearly insignificant and well below the Part 72 guideline of 5 Rem for whole body or any organ.

If this accident were to occur with the ventilation system inoperable, the basin enclosure would contain the fission product gasses and act as a radiation source. Using Microshield v5.05 a Grove Engineering software program for estimating exposure from gamma radiation, the exposure from this source would be (mR/hr):

	BWR	PWR
Off-Site dose	.12	.26
Dose at Basin enclosure boundary	14.2	31.7

b. Actual Bundle Drop Experience

In actual fuel drops, some fuel bundles suffered minor damage, but in all cases, no major deformation of the fuel bundles occurred. For example, during the winter of 1973-1974 the Pilgrim Nuclear Power Station was down for a scheduled refueling and maintenance outage. During transfer of irradiated fuel from the core, a fuel bundle was accidentally dropped from the fuel grapple to the fuel pool floor. The bundle was carefully inspected. There was no indication of major fuel rod failure or distortion nor was there a measurable release of airborne activity as a result of this drop.

In the fall of 1974 during a scheduled outage of the Millstone Nuclear Power Station, an irradiated fuel bundle was dropped to the floor while being transferred from the fuel preparation machine to the fuel storage rack. Consequences of that drop included fracture of all the tie rods, separation at the upper tie plate, and minor permanent deformation at the upper tieplate. Although the fuel bundle appeared to be slightly bent and twisted, no major dislocation of rods, rod segments, or fuel pellets was indicated.

Early in the operation of the Garigliano reactor in Italy, a fuel drop occurred during transfer of fuel to the operating floor. A fuel rack containing five unirradiated fuel bundles dropped on a concrete floor, a distance of about 70 ft. in air. As a result, the rack was badly bent and twisted. Approximately 20% of the 36 fuel rods in each bundle split. Although some fuel pellets were expelled, most of the pellets remained within the fractured rods. Damage to each fuel bundle was confined to the lower one-third of the rods, the lower tieplates and spacers. The upper portion of the bundles remained intact with no apparent damage.

In another case, a fuel bundle was dropped more than 15 ft. and landed on a fuel rack. Consequences of that accident were damage to the nosepiece of the lower tieplate and a slight twist of the assembly. No deformation of the fuel rods or other bundle components was found.

8.5.2 Fuel Basket Drop Accident

After the cask is unloaded and the fuel placed in a storage basket, the basket is transferred to a fuel storage basin (Basin 1 or 2). During this transfer, the basket is less than 3 ft. above the basin floor. When in the cask unloading pit, the maximum height is about 22.5 ft. (equivalent drop height in air is about 12.6 ft.) above the cask unloading pit floor.

In the unlikely event that a basket is dropped in the cask unloading pit, there could be damage to the basin liner, the basket, and the fuel it contains. Damage to the basin liner would be less extensive than that analyzed for a cask drop accident. (See Section 8.4). The criticality aspect of a postulated basket drop accident is discussed in Section 8.9.

The fuel rods within a fuel bundle most likely would not break in a postulated basket drop accident. It has been concluded that fuel bundles in a shipping cask retain their integrity in a 30 ft. cask drop[4]. Since the pipe construction of the fuel basket offers support and protection for the fuel, the postulated drop should cause minor, if any, damage to the fuel.

Comparing actual fuel drops (see discussion in Section 8.6.1) with a postulated basket drop accident at GE-MO, conditions in the actual cases discussed were more severe in that drop heights were greater than the maximum drop height in the GE-MO cask unloading pit (12.6 ft. equivalent in air). Many of the actual drops involved fuel bundles that were unsupported and not as well contained as fuel would be in the GE-MO fuel storage basket.

A structure is installed in front of the entrance of the fuel storage basin (Figure 1-15) to restrain a basket in the event it is somehow dropped at the entrance and the top of the basket tips toward the cask unloading pit. The restraint prevents the basket from tipping in such a way as to disgorge the fuel it may contain.

To transfer a basket from the cask unloading pit, the basket is moved directly under the cask unloading pit doorway guard (Section 5.4.3.3) and lifted through the bottom of the structure. Then the basket is moved laterally into the fuel storage basin. Therefore, the orientation of the basket involved in a postulated drop accident is vertical (i.e., a side drop is not possible and is not analyzed).

8.5.2.1 Accident Analysis

In addition to the assumptions listed in Section 8.6.c, it is assumed the storage basket is full of fuel at the time the accident is postulated. It is unlikely any of the fuel rods would be damaged in such a drop. However, to conservatively evaluate consequences, all the rods in all the bundles are assumed to have ruptured and all the plenum fission gases are assumed to be released to the basin water.

a. The amount of fission gases released to the basin area is calculated to be:

	Amount Released to Basin Area (Ci)	
Species	BWR	PWR
Noble Gases	6156	6120
Iodine	3.01E-6	2.99E-6

b. The maximum off-site dose rates for 2 hr. release duration were calculated to be:

	Maximum Dose Rate (mRem/hr)	
Body Organ	BWR	PWR
Whole Body	4.05E-2	4.0E-2
Thyroid	1.62E-5	1.6E-5

An individual off-site who received the maximum exposure for the 2-hour period would receive less than 0.08 mRem to the whole body and 3.25E-5 mRem to the thyroid. Such an exposure is insignificant compared to the Part 72 guideline value of 5 Rem to the whole body or any organ.

If this accident were to occur with the ventilation system inoperable, the basin enclosure would contain the fission product gasses and act as a radiation source. Using Microshield v5.05 a Grove Engineering software program for estimating exposure from gamma radiation, the exposure from this source would be (mR/hr):

	BWR	PWR
Off-Site dose	1.05	1.04
Dose at Basin enclosure boundary	127.5	126.8

8.5.3 Recovery Practice

Specific procedures for recovering from a basket or bundle accident cannot be described because of the many variables involved (arrangement of bundles on the unloading pit floor, etc.). In general, however, recovery would involve picking up each bundle using appropriate grapples and inspecting each bundle for damage before inserting into a basket. Damaged bundles would be handled (canned or as otherwise appropriate) in much the same manner as for damaged incoming fuel.

8.6 TORNADO-GENERATED MISSILE ACCIDENT

An accident is postulated in which a tornado-generated missile is hurled into the fuel storage basin. Because the building covering the basins is not designed to withstand the forces of a tornado, it is assumed that the building has been blown away, leaving the fuel basins exposed.

The impact of a missile could cause damage to the basin liner or fuel, but not both concurrently. As indicated in the discussion of potential missiles in Appendix A-15, a missile would not have sufficient energy to damage both fuel and basin liner after striking one or the other.

Criticality aspects of this accident are discussed in Section 8.9. The analysis below concerns the consequences of a missile damaging the fuel. In the missile analysis given in Appendix A-

15, two missiles were analyzed. One was a 12 in. diameter by 20 ft. long section of a telephone pole weighing 630 lb. The other missile was a small automobile, 5 ft. by 5 ft. by 8 ft. in dimensions and weighing 1,800 lb. The spectrum of missiles has been expanded to include those listed in Table 8-1. The impact velocity given in the table is defined as that when the missile enters the water of the storage basin.

8.6.1 Accident Analysis

Each missile that was analyzed is listed in Table 8-1. The approximate velocities and kinetic energies at depths of 14 ft. and 21 ft. are given in Table 8-2. These values are those the missile could have if it entered the storage basin water in a vertical orientation. If the missiles entered the water in a horizontal orientation the drag force is greater in many cases and its velocity and kinetic energy would be less. Therefore, the values shown in Table 8-2 are "worst-case" values.

Postulated missile damage depends principally on the cross-sectional (or impact) area, its weight, and the amount of energy it could transfer to the fuel bundle. As indicated in Table 8-2, Missile F has the greatest amount of energy at a depth of 14 ft, which is the depth to the top of the fuel storage baskets. Because of its weight and frontal area (approximately 143 sq. in.), it could potentially cause the most damage. Yet, there is a limit to the number of fuel bundles such a missile could damage.

If the missile entered vertically into the pool, it could potentially strike as many as six BWR bundles or four PWR bundles. The storage basket would move under the impact and the pipes that make up the basket would probably break free. This action would likely absorb all the energy delivered by the missile.

Other missiles, mostly various sizes of pipe, could cause fuel rupture. However, the damage would be confined to one or two fuel bundles, except for Missile E, the 12 in. diameter pipe. This missile could potentially damage as many as six BWR or four PWR fuel bundles, which is comparable to that estimated for the utility pole, Missile F.

Table 8-1
LIST OF TORNADO-GENERATED MISSILES

Missile	Dimensions	Weight (lb)	Impact Velocity as Fraction of Tornado Velocity*
A-Wood Plank	4 in. x 12 in. 12 ft.	200	0.8
B-Steel Pipe	3 in. diam, 10 ft. long, Sched 40	78	0.4
C-Steel Rod	1 in. diam x 3 ft. long	8	0.6
D-Steel Pipe	6 in. diam, 15 ft. long, Sched 40	285	0.4
E-Steel Pipe	12 in. diam, 15 ft. long, Sched 40	743	0.4
F-Utility Pole	13.5 in. diam x 35 ft. long	1,490	0.4

G-Automobile	20 ft.^{2} frontal area	4,000	0.2

- Defined as rotational plus translational velocity.

Table 8-2

VELOCITIES AND KINETIC ENERGIES OF MISSILES IN WATER WHEN ENTERING FUEL POOL IN A VERTICAL POSITION

Missile	14 ft. Depth Velocity (ft./sec.)	Kinetic Energy (ft.-lb.)	21 ft. Depth Velocity (ft./sec.)	Kinetic Energy (ft.-lb.)
A	196	1.2×10^5	124	4.8×10^4
B	195	4.6×10^4	188	4.3×10^4
C	236	7.0×10^3	202	5.0×10^3
D	200	2.0×10^5	196	1.8×10^5
E	200	4.6×10^5	195	4.4×10^5
F	159	6.0×10^5	136	4.3×10^5
G	13	1×10^4	13	1×10^4

Missile G, the automobile, reaches a terminal velocity of about 13 ft./sec. within a depth of about 7 ft. It would then settle to the top of the fuel or to the floor. If it hit the fuel, the energy (one of the least of all the missiles) that it could transfer to the fuel is distributed over a 20 sq. ft. area. No fuel is expected to fail as a result of impact from this missile.

8.6.2 Assumptions

Assumptions used in the safety analysis include the following

a. All the fuel rods in six BWR bundles or four PWR bundles are ruptured. The impact of only one basket is considered.

b. The accident takes place in the fuel storage basin.

c. An average of 30% of the total Kr-85 and 2% of the I-129 activity is in the fuel rod plena and available for release.

d. No solid fission products are released (negligible particulate radioactive material is present in the fuel plena).

e. The overall effective decontamination factor is assumed to be 1 (the accident is assumed to occur in the fuel storage basin).

f. Fuel characteristics are 24,000 MWd/TeU exposure, specific power of 40 kW/kgU and one year cooling.

g. The storage basin is open (i.e., the sheet-metal building over the basin is assumed to have been blown away by the postulated tornado).

h. A maximum X/Q value is 4.0 x 10^{-4} sec/m^3 is taken from Appendix A.5, Section A.5.1 for a short-term ground level release.

8.6.3 Dose Rate Calculations

Using the above assumptions, the amount of fission gases released was calculated to be:

	Amount Released (Ci)	
Species	BWR	PWR
Noble Gas	2.5E+3	3.7E+3
Iodine	1.2E-6	1.8E-6

Assuming an individual was present during the entire period during which the cloud passed, his maximum exposure is calculated to be approximately:

	Dose (mRem)	
Body Organ	BWR	PWR
Whole Body	0.5	0.8
Thyroid	2.3E-4	2.4E-4

Comparing these values with the Part 72 guideline values of 5 Rem to the whole body or any organ, they are clearly insignificant.

8.7 CHILLER SYSTEM LEAK

A water to freon heat exchanger system replaced the fin-fan coolers in 2000, and basin water no longer is piped outside the building to the original fin-fan coolers. The release of radioactive material into the atmosphere because of a leak in the basin chiller system - specifically, a leak in a water-to-freon heat exchanger is not possible. The operating pressure of the freon is greater than the basin water, so freon would leak into the basin water and not the reverse.

If the leakage occurred in the heat exchanger structure, the water would be channeled to a sump and automatically pumped to the Rad Waste System.

8.8 CRITICALITY ACCIDENT

The safety margin against an accidental criticality could potentially be reduced by receiving fuel that is more reactive than assumed in the design analyses or by mechanical damage to the storage basket or fuel sufficient to cause the stored fuel bundles to be forced into a critical configuration.

8.8.1 Fuel Handling Procedures

Nuclear safety in the cask unloading pit is maintained, in part, by handling one fuel bundle or one fuel basket at a time in accordance with approved procedures. However, fuel baskets are not limited to one fuel bundle when being transferred to storage: each basket can hold as many as four PWR fuel bundles or nine BWR fuel bundles.

The baskets are designed to rest in a grid installed in the fuel storage basins. A single grid section is installed in the cask unloading pit to hold a maximum of three baskets in line.

Fuel bundles are transferred, one at a time, from the shipping cask to the storage baskets. (See Section 1.) The baskets are removed from the cask unloading pit, one basket at a time, and placed in the fuel storage basin. Prior to moving the cask, all fuel must be removed from the cask unloading pit; either moved to storage in Basins 1 or 2, or loaded into the cask for transfer.

8.8.2 Reactivity Calculations

KENO calculations were performed by BNWL for a square array of four PWR bundles having 3/16 inch stainless steel plate between the bundles and around the array. For fuel having an enrichment of 1.575% U-235 and a K_∞ of 1.1996 the k_{eff} values for the array were as follows:

Bundle Pitch (in.)	k_{eff}
8.675	0.930 ± 0.004
9.250	0.923 ± 0.004
9.732	0.890 ± 0.005

The results calculated with the GE codes are about 5% more conservative than those calculated with the KENO code. Fuel characteristics for these calculations were as follows:

Rod Pitch:	0.604 in.
Rod o.d.:	0.448 in.
Pellet diameter	0.400 in.
Cladding Material	Zirconium
Rod Array:	14 x 14

PWR fuel having an initial k_∞ of 1.35 (2.8% U-235) and having undergone one cycle of irradiation (10,000 MWd/TeU) would have a post-irradiation k_∞ based on BNWL calculations using the LEOPARD code, of approximately 1.19. Calculations of uniform arrays of PWR fuel were made by GE personnel using proprietary reactor design codes, to describe the relationships between k_∞ spacing and K_{eff}. These calculations did not include the poisoning effect of the stainless steel in the baskets, which BNWL calculations indicated would reduce k_{eff} by 2.5%. Figure 8-4 depicts the relationship between k_{∞} and K_{eff} for PWR fuel bundle arrays with 2 in. separation. A 2.5% reduction in k_{eff} is included for the effect of stainless steel. The data shows that k_{∞} would have to exceed 1.21 for the array to be critical.

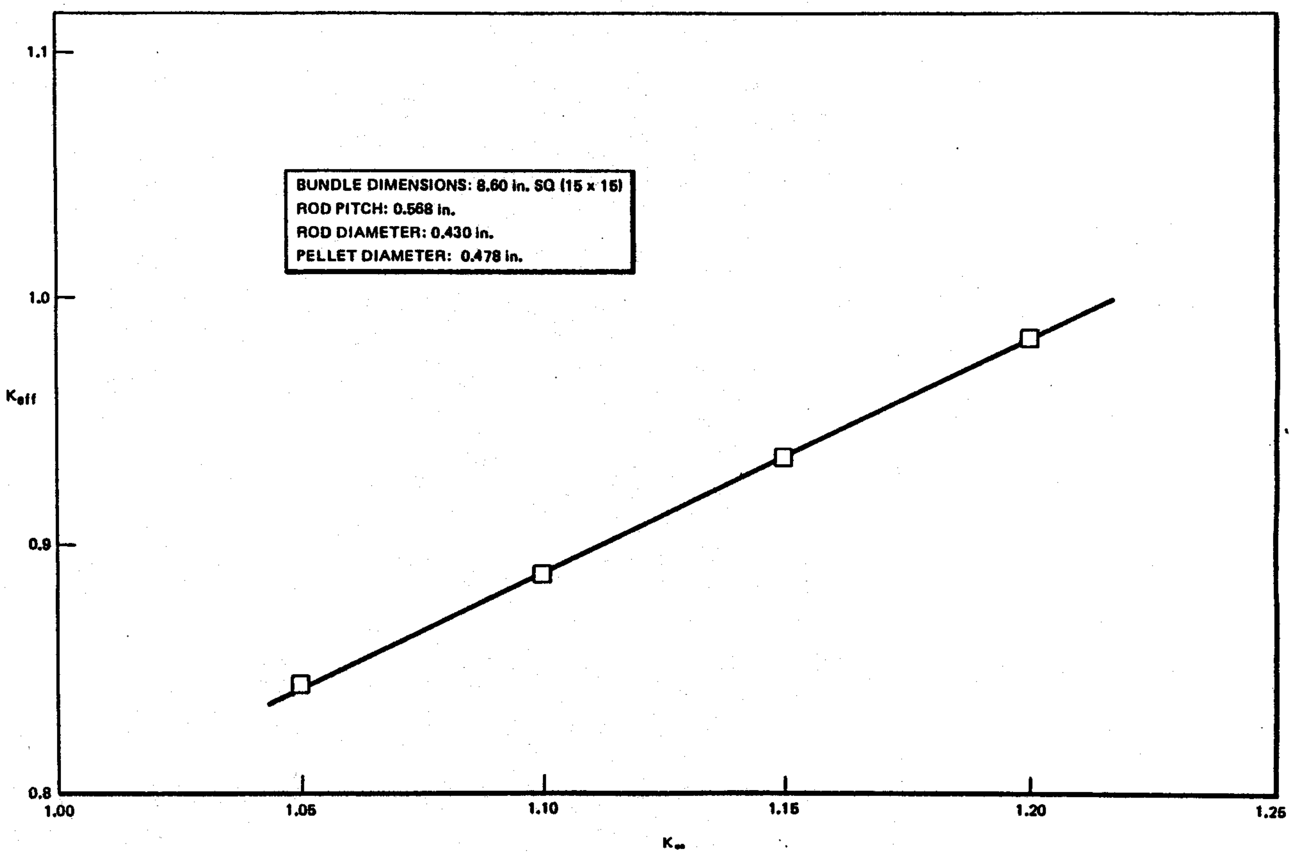

Figure 8-4. PWR fuel bundle array at 2-inch separation.

8.8.3 Missile Impact

The close-packed, pipe sleeve construction of the fuel baskets makes it highly improbable that a missile could cause sufficient compaction of the fuel baskets to cause a criticality accident since the baskets would have to be compressed along two axes simultaneously. Conceivably, a single basket could be driven diagonally into a corner, causing the inner corners of two fuel bundles to be driven together at the top, while the inner corners of the other two elements would at least maintain the designed separation or tend to be spread apart.

Accurate predictions of the effects of the impact of a tornado-borne missile on a system as complex as an array of the fuel storage baskets would be difficult to make or to prove. To provide insight into the potential increase in neutron multiplication that could arise from reduced spacing, an analysis of three PWR bundles in a "T" configuration, closely spaced over their entire length, was done to estimate the effect of driving three assemblies into a corner. Since this example does not provide consideration of the fourth bundle in the basket, an example of reduced spacing involving four PWR bundles is provided. Such a condition represents an extremely improbable event since the fuel would have to be compacted into a corner from two directions 90° apart over a substantial portion of its length. Because such a compaction would result in separation of the fuel in the compacted array by more than 10 inches of water from the fuel in the closest baskets, the four-bundle array can be considered isolated. The results of calculations performed by GE personnel for a water-reflected, close-packed, square array of four PWR fuel bundles are shown in Figure 8-5.

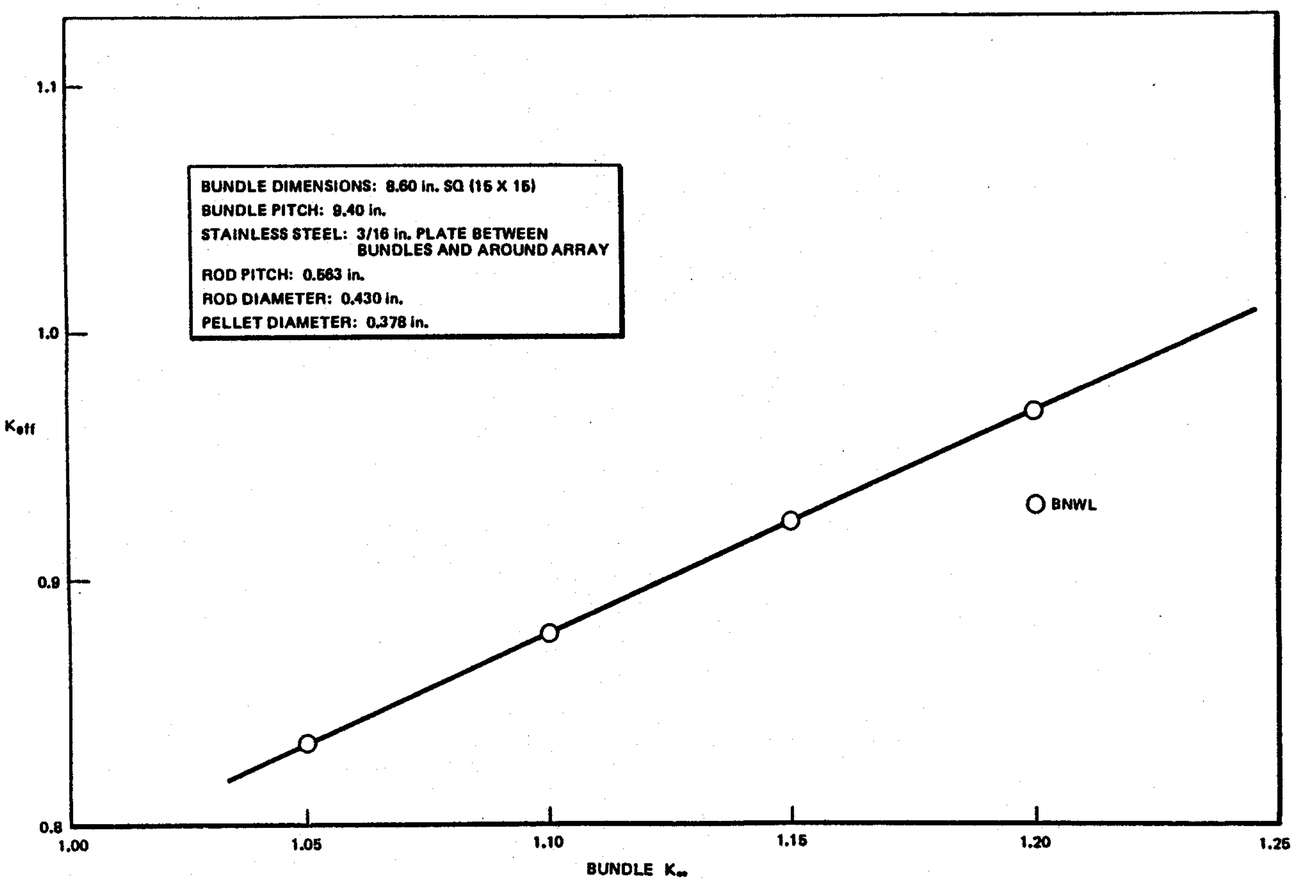

Figure 8-5. Close-Packed array of four PWR bundles.

For such a four-bundle array to become critical, the infinite multiplication factor must average at least 1.23. (Reactivity calculations are discussed in Section 8.9.2)

8.8.4 Consequences of a Criticality Accident

No criticality accidents have occurred in low enriched LWR bundle systems. Accidents have occurred in chemical reprocessing or critical assemblies involving plutonium or highly enriched uranium. Historical criticality incidents in nuclear separation facilities have had fission magnitudes estimated at 1.3 x 10^{17} to 4 x 10^{19} fissions. In no case has the reaction been of an explosive nature.

The accidents have either displaced the critical mass such that it was no longer in a critical geometry and thereby terminating the criticality, or the critical mass pulsed in and out of critical geometry.

A criticality accident in the fuel storage basin of GE-MO is precluded by many factors, some of which include:

a. Geometric constraints imposed by the fuel bundles, storage baskets and holding grid

b. Design and operation of the storage system

c. Administrative procedures for fuel receiving and storage

d. Lower content of fissile material in the fuel bundle than assumed in calculations

e. Neutron poison content in the fuel not assumed in calculations

Nevertheless, a hypothetical criticality is postulated to provide a basis for evaluating the consequences of such an accident. Recovery from a hypothetical criticality would be much the same as from a basket or bundle drop (Section 8.5.2.1), except that a suitable tool suspended from the crane would be used to separate the critical assembly, stopping the reaction. Radiation levels at the pool surface would be low (up to 15 mRem/hr) so that no special protective measures would be required.

8.8.4.1 Assumptions

Primary assumptions used to evaluate a criticality accident include:

a. a point source is assumed at a depth of 16 feet; and

b. Fission gases released to the pool atmosphere as a result of the criticality are negligible. Release of fission gases due to the missile impact is covered by Section 8.7.

Since no reasonable mechanism exists for a criticality accident in GE-MO fuel storage pools, no meaningful values for characteristics such as reactivity insertion rates, specific power, etc., can be defined. However, a range of 10^{18} to 10^{20} fissions has been evaluated and adequately covers the range of total fissions for such a system.

A depth of 16 ft. was assumed because about 90% of the active fuel is below the 16 ft. level. The top of the active fuel is 14.5 ft. below the water surface.

It is assumed that all the fission products, including fission gases, would be contained within the UO_2 fuel matrix. Temperatures would not be sufficient to drive the fission products from that matrix. Any products that migrate from the fuel matrix would be contained within the fuel void spaces inside the fuel rod.

The gamma flux at the surface of the pool is approximated by the equation for a point source:

$$(\phi) = \left(\frac{BS}{4\pi t^2}\right)\left(\exp(-ut)\right)$$

where

ϕ = scalar flux (MeV/cm^2-sec);
B = build-up factor;
S = source strength (MeV/sec);
t = distance from source to pool surface (487.68 cm); and
μ = macroscopic cross section for shield material, water (cm^{-1})

Gamma-ray spectra for prompt fission photons are given in Table 8-3. Table data were found in Reactor Physics Contents, ANL-5800, Section 8. The four-group Spectrum B that is given in Table 8-3 was used to calculate the gamma flux. Values for the buildup factors were found in Rockwell's Reactor Shielding Design Manual, page 435.

The dose rate is:

D' = ϕ/c

where

D' = dose rate mR/hr

c = flux to dose conversion factor $\frac{\text{MeV/cm}^2\text{ - sec}}{\text{mR/hr}}$

Values for c for each energy group are:

$c_1 = 5.2 \times 10^2$

$c_2 = 6.2 \times 10^2$ $\frac{\text{MeV/cm}^2\text{ - sec}}{\text{mR/hr}}$

$C_3 = 7.8 \times 10^2$

$C_4 = 8.6 \times 10^2$

The dose rate in terms of mR/fission is given by:

$$\frac{BM(E)e^{-ut}}{4\pi t^2 c(3600)}$$

where

M(E) = energy/fission, or MeV/fission

Table 8-3

PROMPT FISSION GAMMA-RAY SPECTRA

	Spectrum A		Spectrum B	
E (MeV)	N(E) (γ/fission)	M(E) (MeV/fission)	E (MeV)	M(E) (MeV/fission)
0.5	3.1	1.55	-	-
1.0	1.9	1.90	1.0	3.451
1.5	0.84	1.26	-	-
2.0	0.55	1.10	2.0	3.085
2.5	0.29	0.725	-	-
3.0	0.15	0.450	-	-
3.5	0.062	0.217	-	-
4.0	0.065	0.260	4.0	1.035
4.5	0.024	0.108	-	-
5.0	0.019	0.095	-	-
5.5	0.017	0.094	-	-
6.0	0.007	0.042	6.0	0.256
6.5	0.004	0.026	-	-
	7.028	7.827		7.827

Values of M(E) are given in Table 8-3 for Spectrum B. The calculated doses in terms of mR/fission at the surface of the water in a storage basin are given in Table 8-4. The calculated doses at the surface of a basin from 10^{18} fissions, 10^{19} fissions, and 10^{20} fissions are 0.413 mR, 4.13 mR, and 41.3 mR, respectively. These doses are obviously not of serious consequence.

For comparison, extrapolation of actual measurements from an experiment produced a gamma-ray tissue dose rate of 0.18 mRad/hr. These data were taken from Figure 8.8 in Section 8, ANL-5800, showing plots of centerline attenuation data for water measured in the Bulk Shielding Facility at ORNL.[5]

The curves in Figure 8.9 of ANL-5800 also give data for fast neutron dose rate and thermal neutron flux. These data are given as a function of watts for the source, which is a reactor in this case. As indicated, the thermal neutron flux for 16 ft. (approximately 488 cm) is 5×10^{-8} n/sq cm - watt. The fast neutron tissue dose curve drops sharply and ends at a value of 2×10^{-7} erg/gm - hr watt for approximately 175 cm. The fast neutron dose at a distance of about 488 cm is negligible.

Table 8-4

DOSE, mR, PER FISSION,
AT BASIN SURFACE

Group	Dose: mR/fission
1	2.118×10^{-25}
2	6.780×10^{-22}
3	1.391×10^{-19}
4	2.736×10^{-19}

A criticality of 10^{18} fissions produces about 8.9 kWh of energy. If it is assumed the event lasts 3 hours, the power level for those 3 hours is about 3 kW. The thermal neutron flux was determined to be approximately ($1.5 \times 10^{-4 \text{ n/sq. cm.}}$) - sec at the surface of the pool. The corresponding dose rate is about 6.2×10^{-7} mRem/hr.

The consequences of a postulated criticality in the storage basin are no more serious than the short-term operation of a low-power, swimming-pool type nuclear reactor commonly used at some universities.

8.9 REFERENCES

1. According to recent studies in the U.S. and abroad, significant evidence indicates that consequences of a hypothetical fuel melting accident may be less than currently predicted by at least one or two orders of magnitude, see appendices E, F, and G, Report of the President's Commission on the Accident at Three Mile Island.

2. N. R. Horton, W. A. Williams, and J. W. Holtzelaw, Analytical Methods for Evaluating the Radiological Aspects of the General Electric Boiling Water Reactor, March 1969 (APED-5756).

3. RESSAR-41, April 1974.

4. See "IF-300 Shipping Cask Consolidated Safety Analysis Report," NEDO-10084-2, Chapter V.

5. Attenuation in Water of Radiation from the Bulk Shielding Reactor: Measurements of the Gamma-Ray Dose Rate, Fast-Neutron Dose Rate and Thermal Neutron Flux, July 8, 1958 (ORNL-2518).