TECHNICAL REVIEW OF RISK DUE TO EXPANSION OF THE MORRIS OPERATION SPENT NUCLEAR FUEL STORAGE

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SECTION 1

1.0 INTRODUCTION

The Morris Operation (MO) is a large facility originally designed for reprocessing spent fuel but subsequently licensed for spent fuel storage only. General Electric Company, who owns the MO, has applied for an increase in the licensed capacity and the Illinois Attorney General (IAG) has been recognized as an intervenor on behalf of the people of the State of Illinois in the licensing proceedings regarding this expansion. MHB Technical Associates has been retained by the IAG to review the reports and documentation of the Applicant in the MO licensing process, the environmental statement and safety evaluations by the regulatory bodies and the reports on spent fuel storage technology, and to perform a study assessing the extent to which the risk to health and safety of the public is impacted by expansion of MO.

The study has taken place over a six-month period and includes the benefit of a four-hour tour of the Morris facility, several meetings with members of the IAG office, and several hundred documents received through discovery requests. The following major aspects of the MO which could impact risk have been considered in this study: the site, the facility, major systems, major equipment, procedures, and operating history. An effort has been made to correlate the findings with the existing contentions in the intervention. This report is a summary of the findings of the study.

SECTION 2

2.0 HISTORY OF THE MORRIS OPERATION

The Morris Operation was originally designed by General Electric Company (GE) as a spent fuel reprocessing facility. Construction started in early 1968 on what was then called the midwest Fuel Recovery Plant (MFRP). In December of 1971, GE received an AEC license(1) for receiving, handling, and storing Special Nuclear Materials, mainly fissionable uranium and plutonium in the form of irradiated fuel. When construction was completed, the plant was subjected to preoperational tests and trial runs with test fuel rods constructed of depleted uranium. These tests and subsequent evaluation by a Task Force under the guidance of Dr. Charles Reed of GE disclosed technical problems with the MFRP that made it undesirable for GE to proceed with the reprocessing of commercial spent reactor fuel.(2) As a result of the GE decision, in November 1974 the AEC issued an order(3) requiring partial dismantling of the facility to render it inoperable, thus preventing any unauthorized activity involving Special Nuclear Materials. At about the same time, the GE license was amended(4) to permit the receipt, storage, and transfer of nuclear fuel from boiling water reactors (BWR's) and pressurized water reactors (PWR's).

The fuel storage basin at MO was originally designed for 100 MTU (metric tons of uranium)(5) to be contained in 32 fuel baskets with an adjacent pool designed to hold containers of high level waste (mainly strontium, cesium and iodine) resulting from the reprocessing operation. In December of 1975 the license was amended(6) to permit the conversion of this high level waste pool to a spent fuel storage pool by the addition of fuel storage racks and changes to the handling equipment. This modification increased the MO capacity from 100 to 750 MTHM. A November 1977 accounting by GE showed that the MO had 295 MTHM of stored fuel, 51 MTHM of space contracted, and 354 MTHM of reserve(7) space.

In April of 1977 GE requested of the NRC a license amendment to permit expansion of the MO by 1100 MTHM capacity of 1850 MTHM. This is to be accomplished by building an additional fuel storage pool to be attached to the existing pool. The addition of this storage pool (referred to as Basin 3) is to be done without transferring the existing stored fuel from Basins 1 and 2. The expansion creates new technical considerations and a potential increase in risk. Currently, the hearing process has been suspended at the request of the Applicant to await a national policy decision on waste disposal.

With the United States government decision to delay reprocessing indefinitely, many options are being evaluated for existing and new spent fuel storage facilities. GE has studied the conversion of the unused reprocessing canyon (the concrete hall where the reprocessing equipment is in a partially dismantled state) for the use as a dry storage facility for spent fuel,(8) and has considered using dense racks and soluble poison (borated solution for absorption of neutrons to prevent criticality) as a means of increasing storage capacity.[9] There is also the possibility that the government will take over the MO facility.[10]

The U.S. policy for spent fuel storage and long-term waste disposal has not yet evolved. The present federal regulations are lacking in authority and control over independent spent fuel storage systems.

Because of this, MO, the only operating Away-From-Reactor (AFR) storage facility, has an uncertain future. However, it is clear that MO is potentially attractive for early exploration to provide a stopgap solution to a serious national problem. Thus, each step in the revision or expansion of MO must be carefully reviewed and evaluated not only with regard to the near term effect on the health and safety of the Illinois public, but also as to the implications that such actions may have on the eventual U.S. waste disposal policy.

REFERENCES

1. U.S. Atomic Energy Commission Special Nuclear Materials License, SNM-1265, Dec. 27, 1971.

2. Midwest Fuel Recovery Plant Technical Study Report, July 5, 1974, General Electric Company.

3. Order Authorizing Dismantling of Facility to Render Inoperable, CSF-2, Nov. 21, 1972.

4. USNRC, SNM-1265, revised and reissued, Aug. 23, 1974.

5. MTU = Metric Ton of Uranium. Also referred to as TeU or tonnes uranium where 1 tonnes = 1,000,000 grams = 1 megagram. MTHM is used when other heavy metals (in addition to uranium) are included; MTHM = metric tons of heavy metal. A single fuel element weighs about 669 Kg for a PWR and 189 Kg for a BWR, and a reference bundle (mixture of 2:1, PWR and BWR bundles) is assumed to be 454 Kg (see ERDA 76-43, Vol. 1, May 1976, pg. 2.32).

6. USNRC, Materials License, Revised and Reissued for Increased Capacity of Facility, SNM-1265, Dec. 3, 1975.

7. Letter, D.M. Dawson (GE) to Richard W. Starostecki (NRC), 12/12/78, showing fuel from the following plants: Dresden II (145 MTHM), Connecticut Yankee (33 MTHM), San Onofre (76 MTHM + 51 under contract), and Point Beach (41 MTHM).

8. Summary of Environmental Report; Fuel Storage Facility Expansion, NEDO-21624, Apr. 1977, pg. 1-5.

9. Letter, Clark (NRC) to Miller (NRC), 1/6/75, documenting 12/19/74 meeting.

10. <u>Nucleonics Week</u>, Vol. 19, No. 50, Dec. 14, 1978, pg. 8 and 9.

SECTION 3

3.0 REVIEW OF THE CONSOLIDATED SAFETY ANALYSIS REPORT (CSAR)

The Consolidated Safety Analysis Report (CSAR)[1] is an abbreviated version of the Safety Analysis Report (SAR) originally created for the Midwest Fuel Reprocessing Plant.[2] The CSAR has been updated and edited to cover only the sections and information pertaining to the receipt, storage and transfer of spent nuclear fuel, thus reflecting the new mission of the Morris facility.

The CSAR is divided into three main subjects:

- description of the site and facility
- description of the procedures and design features incorporated to minimize radiation exposure during normal operation
- analysis of radiological impact of postulated accidents

This section of the study documents our review of the important aspects of the site and facility during normal operation. Section 4 documents our review of the accident analyses.

3.1 SITE

The MO site is located approximately 50 miles southwest of Chicago, as shown in Figure 3-1. This site has both favorable and unfavorable characteristics for a nuclear fuel storage facility. The positive site features include the following:

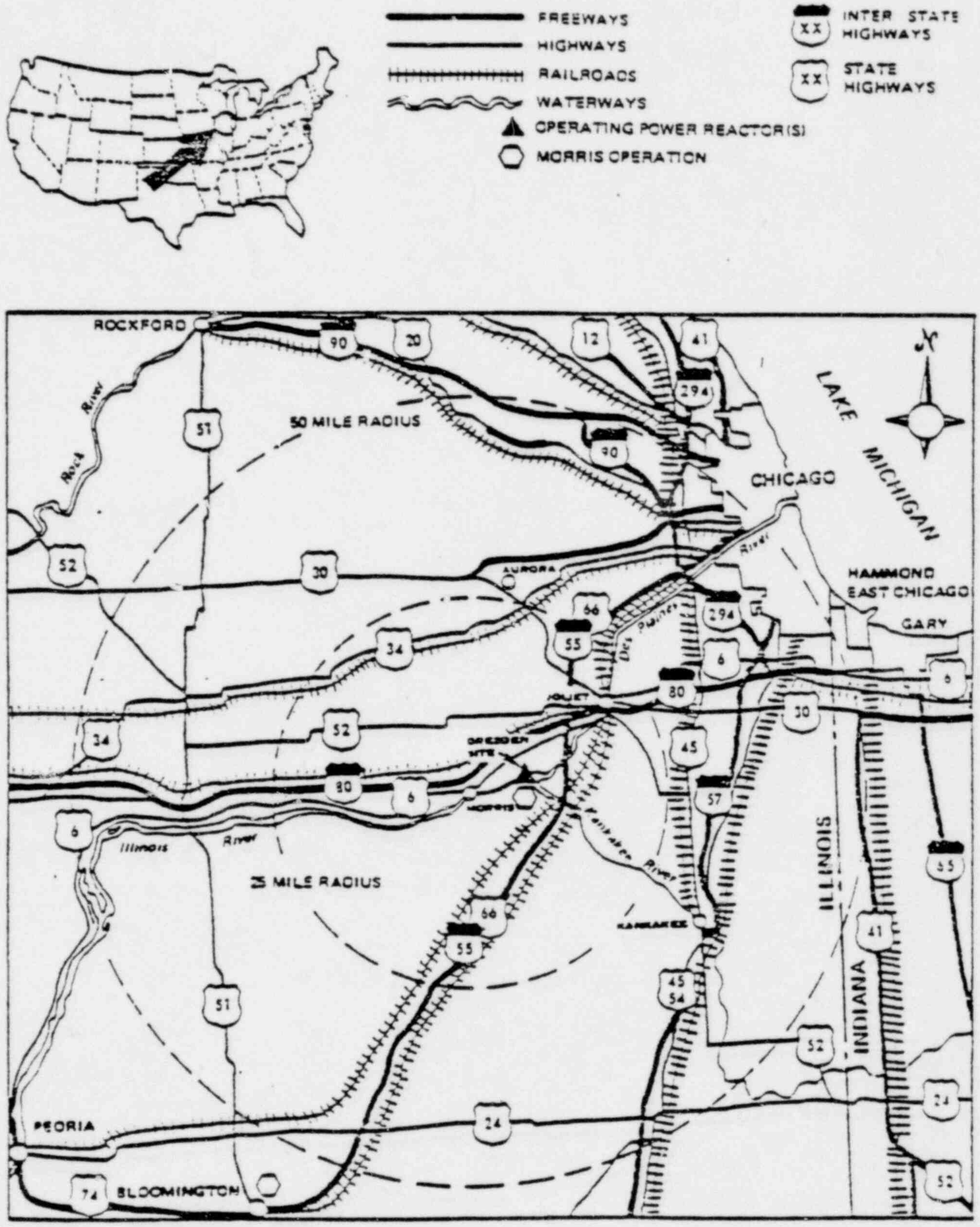


FIGURE 3-1

GENERAL LOCATION - MORRIS OPERATION

1. Only 5,000 people live within the five-mile radius of the Morris Operation.[3] The population in this area is estimated to increase to 6,500 by 1980.[4]
2. Fairly competent rock is found just below the surface.
3. Major earthquakes are uncommon in this area.[5]
4. Twenty-four nuclear monitoring facilities have recently been set up in the area to supplement the more than 50 federal and state monitoring stations active for more than a decade.[6]

There are also site features which are unfavorable to locating the expanded MO at this site:

1. Over 6 million people[7] live within 50 miles of the site (projected to increase to 8 million [8] by 1980).
2. Over 5 million people live in the 45° sector (NE and NNE of MO) where the wind blows effluents from the MO site more than 10% of the year.[9] A major base of radioactivity could impact these people if adequate precautions (e.g., evacuation or staying indoors) were not taken.
3. Tornadoes are common to the area with about one third of the 140 tornadoes reported in Illinois classified as destructive. Two tornados have been reported near the MO site but caused no damage

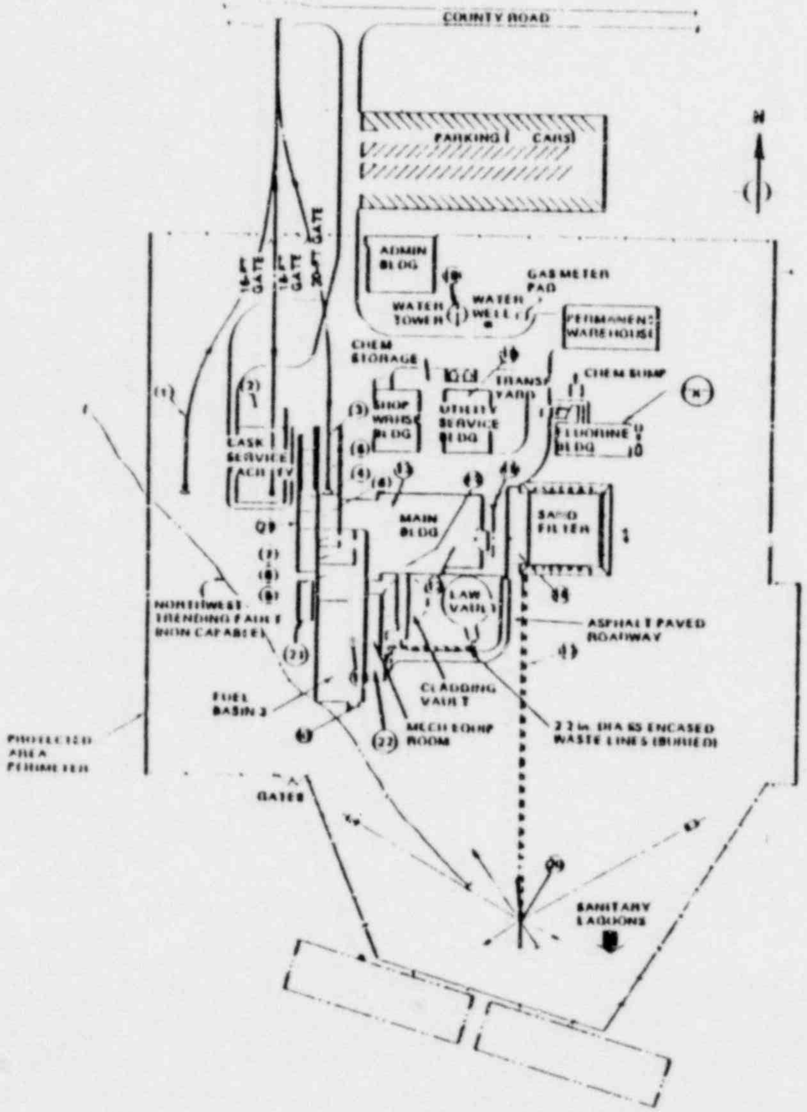
to the site.[10] (The existing structures covering the storage basin are not expected to survive a tornado.)[11]

4. The proposed basin addition would have one corner on an identified earthquake fault. However, this fault has been declared incapable of producing a major earthquake.[12]
5. The site is at the head waters of major state and national river systems with the Des Plaines and Kankakee Rivers feeding the Illinois River and eventually the Mississippi. A major radioactive release might contaminate the water used by hundreds of thousands of consumers downstream from the site.

These factors may not rule out the Morris site as a spent fuel storage site, but the factors do need to be carefully considered, especially in light of the possibility that the MO could become a de facto permanent waste disposal site.

3.2 FACILITY

Many aspects of the existing facilities will be utilized for the expansion. Figure 3-2 shows the MO with Fuel Basin 3 added (Item 11). The major structures and equipment which will be shared with the new basin and which thus are important to the safety of the proposed expansion are:



1 STORAGE SPUR – 8 CASK RR CAR CAPACITY
2 2 300 gal DIESEL FUEL TANK – LOCOMOTIVE REFUELING
3 CASK RECEIVING AREA
4 CAR BUMPER MOUNTED ON HEAVY CONCRETE BLOCK
5 125 ton BRIDGE CRANE – CASK HANDLING
6 DECONTAMINATION AREA
7 CASK UNLOADING BASIN
8 FUEL STORAGE BASIN 1
9 FUEL STORAGE BASIN 2 (FORMERLY WASTE STORAGE BASIN)
10 DOUBLE GATE TO PERMIT FUTURE EXPANSION OF BASIN AREA
11 FUEL STORAGE BASIN 3
12 EVAPORATOR, PART OF LOW ACTIVITY WASTE REDUCTION
13 PROCESS STEAM GENERATOR
14 CONTROL ROOM
15 VENTILATION AIR PASSAGE TO SAND FILTER
16 EMERGENCY EQUIPMENT: 600 kw DIESEL GENERATOR; AIR COMPRESSOR; 3 18,000 cfm EXHAUST BLOWERS
17 AIR TUNNEL TO 300 ft STACK
18 UTILITY BOILER, DEMINERALIZERS, AND SWITCHGEAR
19 50,000 gal WATER TOWER; 42,000 gal FIRE RESERVE
20 300 ft VENTILATION EXHAUST STACK
21 BASIN WATER FILTRATION FACILITY, BASIN 1 AND 2
22 BASIN WATER FILTRATION FACILITY, BASIN 3
23 BASIN WATER COOLERS
"X" FLUORINE BUILDINGS – NOT IN USE AND NOT RELATED TO FUEL STORAGE OPERATIONS

NOTE: SHOWING STATUS AT COMPLETION PROJ IV

FIGURE 3-2

MORRIS OPERATION FACILITIES

- cask receiving area (Items 3, 6 and 7)
- ventilation system
- sand filter and stack (includes Items 15, 17, and 20)
- evaporator (located in the main building, Item 12)
- LAW (low active waste) vault

There are several major systems which are to be added during the expansion:

- Basin 3
- basin water filtration facility (Item 22)
- water cooling system for basin
- ventilation system (extension)
- instrumentation including new area radiation monitoring and criticality instruments (not shown on Figure 3-2)

In the following subsections of this report, these and other major systems are reviewed to show how their normal and accident conditions can impact safety.

3.2.1 VENTILATION SYSTEM

The main sources of airborne radioactivity are:

- effluents from the LAW vault
- vented gas from shipping casks
- gases and volatiles from the pool and decontamination areas
- off-gas from leaking rods

The ventilation system is designed to flush clean air over the pool areas, through the canyon, and on to the sand filter and stack. A parallel path flows through the control room as shown in Figure 3-3.

There are some features of the system which are insufficiently described in the CSAR to evaluate their impact on safety. For example, Figure 3-3, taken from the CSAR, shows the ventilation system recirculating a portion of the air through the heating/air conditioning unit. This could cause a build-up of radioactive contaminants in the unit as well as distributing airborne contaminants from one basin to another. In the event the screen or sand filter should clog, the blowers fail, or the stack ice up or fail, it could be possible for the blowers on the heating/air conditioning unit to force air backwards through the control room. The impact would be small because there are not many airborne contaminants during normal operation, but could be a complicating factor during an accident.

By far, the major airborne contaminant emitted from ruptured or leaking spent fuel will be krypton-85.[13] Krypton activity in spent fuel will range from 1000 to 10,000 curies/MTHM depending on the exposure and cooling time (24,000 MWD/MT with one year cooling is about 7000 Ci/MTHM), with 20 - 45% assumed to be in the plenum and released from the basin in the event of a rupture of the fuel cladding. Krypton (Kr) is a relatively inert noble gas and therefore very difficult to filter or extract from the ventilation stream. As a result, it is simply

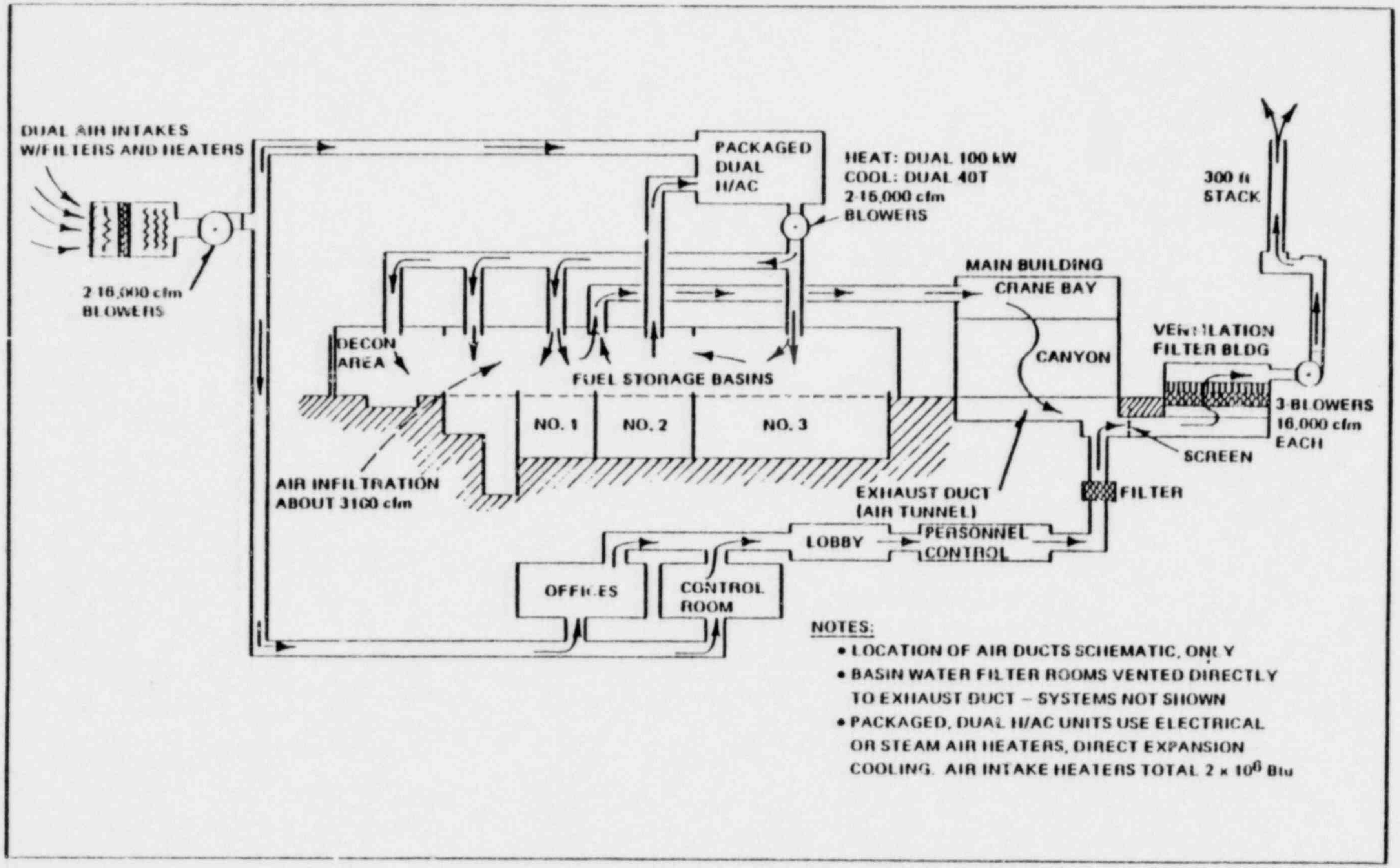


FIGURE 3-3

<u>Ventilation System</u>: Exhaust air passes from the basin area through the main building to the ventilation filter building (the sand filter), then exhausted through the stack. Negative air pressure is maintained in the enclosed areas.

passed through the sand filter and out the stack. The above mentioned ventilation system failures could concentrate the gases and cause a heal h risk to the employees.

The CSAR evaluates the risk to the public of airborne radioactivity release by camparing the dose averaged over 5 million people within 50 miles of the site to the background dose that same population receives.(14) This technique produces a deceptively favorable comparison because the impact of any one accident will not be felt by 5 million people. Another comparison used in the CSAR is to compare Kr release from the MO to that of the nearby Dresden Nuclear Power Station. Clearly, the release from MO should never be as large as that from a large reactor complex such as Dresden I, II, and III which is quoted as 1.68×10^6 curies (Ci) in the year 1973.(15) This comparison is further misleading because it implies the Dresden release is acceptable when in fact the EPA has recently proposed the upper limit for noble gas release should be lowered to 50,000 Ci/GW_e- YR(16) (curies per gigawatt electrical year), a factor of 20 lower than the figure quoted in the CSAR.

The CSAR analysis for Kr release assumes the maximum release as the release of the gases in the gap of only one basket of PWR fuel. The CSAR analysis should also consider a systematic mechanism which may corrode or weaken many fuel rods at one time to cause the simultaneous release of their contained gases. Factors which could contribute to such corrosive processes are discussed in a later section.

An additional concern about the ventilation system is that the air intake is at the top floor of the main building (east end) which is located about 400 feet north of the stack (see Fig. 3-2). This appears to be an undesirable position for an air pick-up point because the most prevalent direction for wind is from the south. The stack release is 300 feet above grade and the intake structures are about 50 feet above grade.(17)

3.2.2 BASIN WATER FILTER

In the basin filter system shown in Figure 3-4, no redundancy or cross-connections appear to have been provided. This makes both systems (the original system and the new Basin 3 system) single-failure prone with only a loose coupling through the basin gates permitting the two systems to share the other's load. Such a condition would only be of concern, however, if the basin were heavily loaded with high exposure, short-cooling-period fuel and even in this situation, only if the cooling cannot be restored in a reasonable period of time (probably several days).

The normal function of the filter is to remove detrimental contaminants from the water. Principal sources of radioactive contamination are:

- activated corrosion products
- fission products on the exterior of the fuel rods
- fission product leaking from the rods

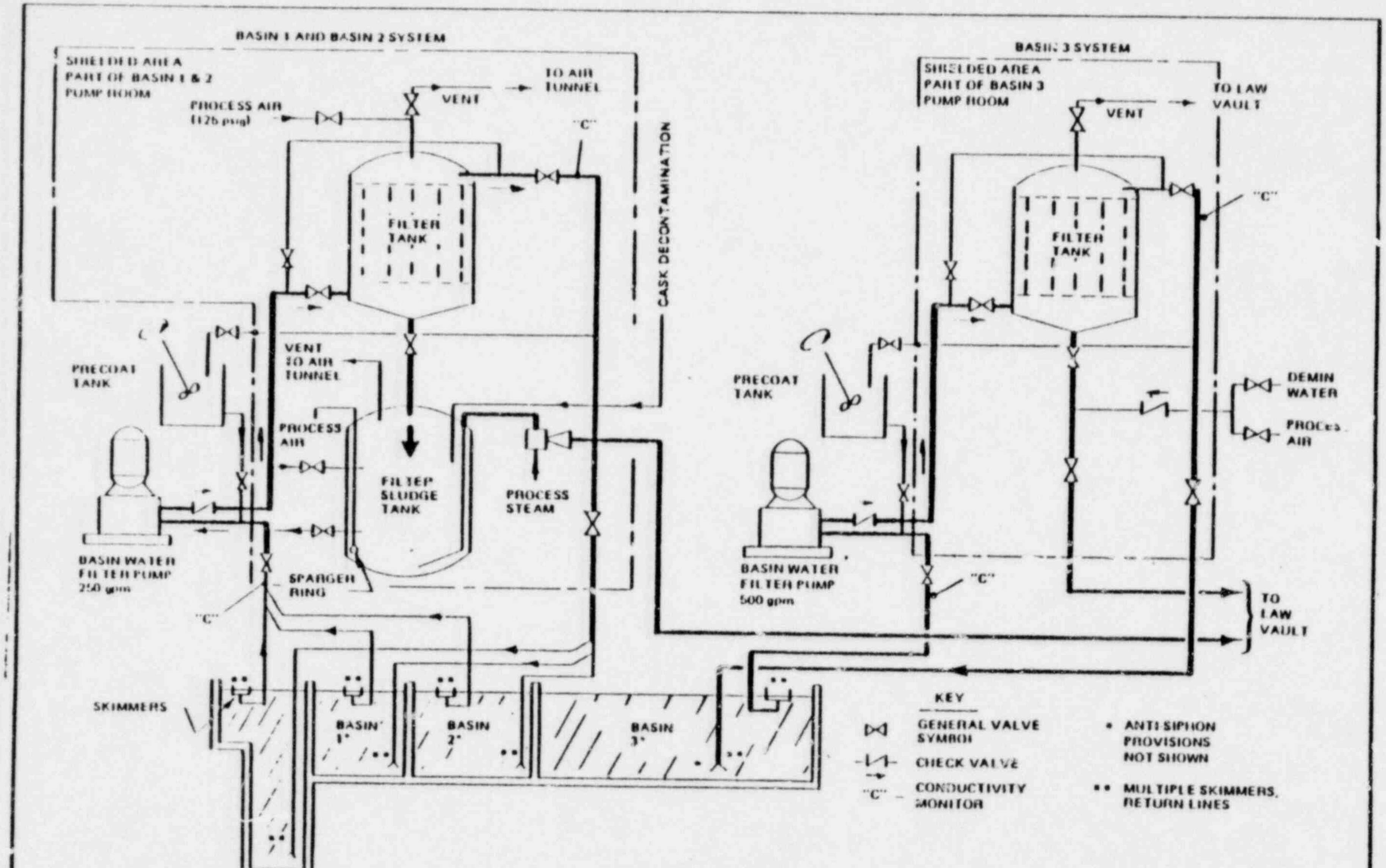


FIGURE 3-4

Basin Filter System: Water is continually drawn from basin skimmers filtered, and returned to the basin. Filter sludge and decontamination runoff are collected in the Basin 1 and 2 sludge tank, then jetted to the LAW vault. Basin 3 filter sludge discharged directly to the LAW vault. Provisions are included for flushing tanks and pre-coating filters.

The most prevalent contaminant in the basin water has been the fission product Cesium, which, in 1974 reached concentrations of 10^{-2} microcuries per millimeter. This is about 30 times the Maximum Permissible Concentration (occupational) for water (MPC_w).[18]

During the review of the original MO expansion (to 750 MTU), the NRC selected Argonne Labs to review the environmental impact. Their major concern was the water activity. The Argonne reviewers recommended additional ion exchanges (removal) since their analyses indicated a linear increase in radioactivity of the storage pool. The increased activity was felt to be a danger both to the environment and to the plant personnel.[19] In response to these concerns, the filter system at the MO was augmented by adding Zeolon to the Powdex filter. This change has brought the radioactive Cesium level within the acceptable range for the present fuel capacity. MO personnel are confident that this change will resolve the problem for the new basin as well.

3.2.3 BASIN WATER COOLING SYSTEM

For the Basin Water Cooling System shown in Figure 3-5, more consideration has been given to sharing of the two systems. However, in the new and existing systems there is no provision for precluding crud or contamination build-up in the individual components nor is there a shielded area for the equipment as was provided for the water filters (see Figure 3-4).

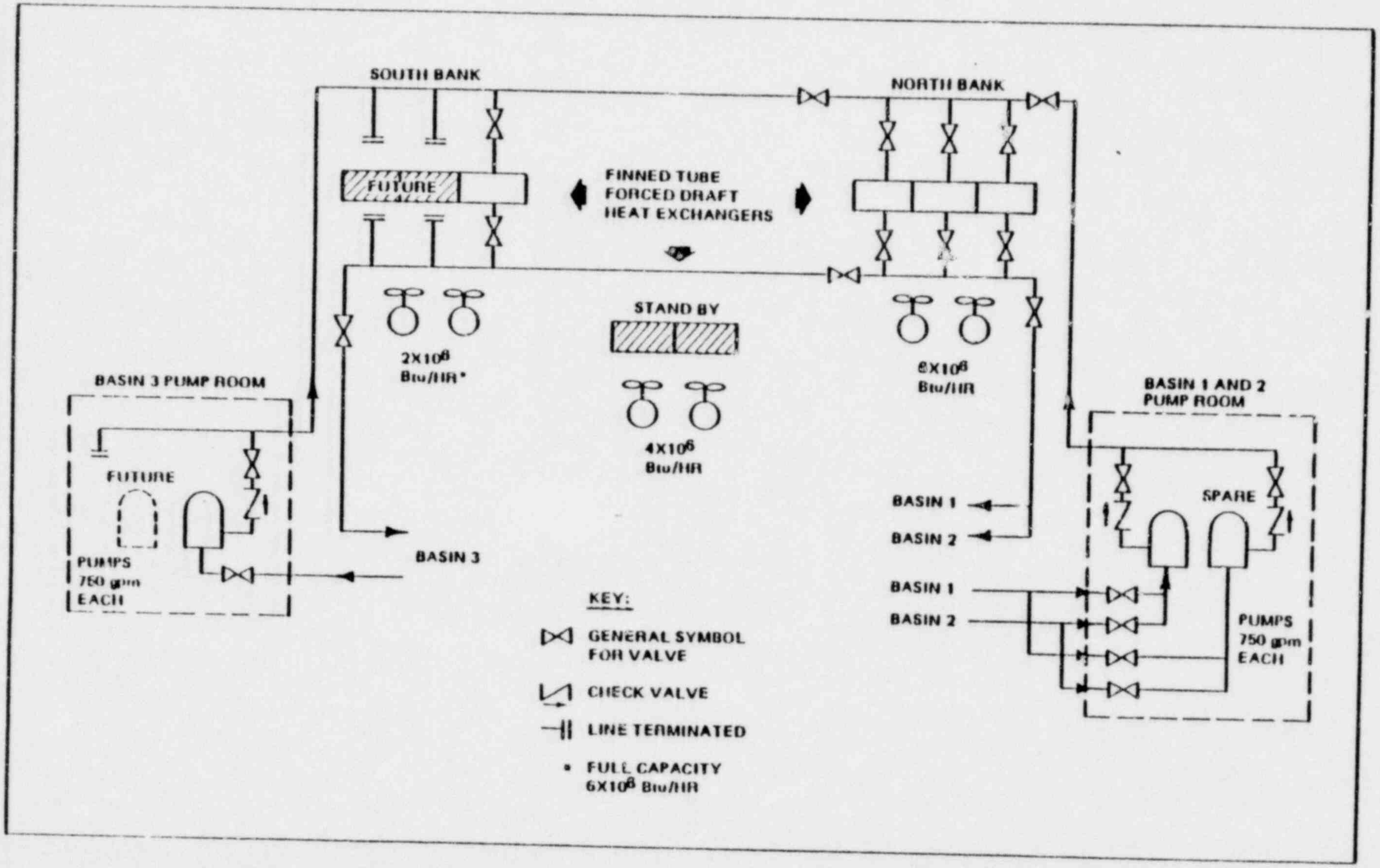


FIGURE 3-5

Basin Water Cooling Systems. Separate systems are provided for Basin 3, and Basins 1 and 2. Cooling capacity will be expanded as stored fuel increases. Valving (not all details are shown) provides flexibility to accommodate variou heat loads.

Radioactivity build-up in the Fin-Fan cooler for Basins 1 and 2 precluded extended periods in one of the main working areas next to Basin 2 at the time of the May 1978 tour of the MO (ref. Appendix B).

3.2.4 BASIN 3 ADDITION

The construction of Basin 3 will involve some very questionable activities. First, the radioactive fuel stored in Basins 1 and 2 is being left in place during construction. Second, most of the area to be occupied by Basin 3 must be blasted from the shale bedrock material. Third, the south wall of Basin 2, including the full height gate, will be exposed during portions of the construction. Fourth, it is possible the south wall could be overstressed by the hydro and shock loads caused by the construction process.[20] If the wall or the expansion gate were to fail, the failure could cause a very rapid loss of pool coolant to the excavated area with possible detrimental effects on fuel cladding integrity of newer or weakened fuel. However, the previous excavation and backfill of the area adjacent to the south wall may produce a sufficient mitigating effect to prevent overstress condition.

In August 1975, a leak developed in the expansion gate at the south end of Basin 2.[21] This is the same gate that must maintain the integrity of the basin during the construction process for Basin 3. Methods to avoid such problems during construction presently being considered include structures to hold the gate in place and secondary dams to catch leakage.[22]

Safe completion of the expansion program poses a dilemma. A reduced risk during construction would be provided by the removal of all fuel. However, the reason for the expansion program is that there is nowhere to send currently generated spent fuel, much less the spent fuel now at MO. This poses an even greater problem for the future when it could be necessary, due to some unforeseen event, to remove the fuel and drain the MO pools. Where would the fuel be shipped? Clearly, a contingency plan should be developed before more fuel is brought into the MO.

Basin 3 will also have an expansion gate at its south wall which would indicate that additional MO expansion may be contemplated by the Applicant.

3.2.5 CASK RECEIVING AREA

When a cask is received at the Morris site, the federal regulations require that it be smear tested for external contamination within 3 hours.[23] A review of selected documents from the NRC files on Morris reveals that several casks have shown up with external contamination as much as 5 times the limit. Table 3-1 gives a partial listing of reported contaminated casks. Shipments from the nearby Dresden Nuclear Station resulted in repeated violations of the contamination limits.

TABLE 3-1

PARTIAL LISTING OF REPORTS OF EXCESSIVE RADIOACTIVE CONTAMINATION* ON EXTERNAL SURFACES OF RECEIVED CASK SHIPMENTS

EXAMPLES AT THE MORRIS OPERATION:

Cask	Date	From	Max. CPM **
IF 100 - 0035	5/12/75	-	-
IF 100 - 0034	5/17/75	-	-
IF 100 - 0034	5/31/75	-	-
IF 300 - 301	9/10/75	-	-
-	8/7/76	Dresden	97,000
-	8/8/76	Dresden	87,000
-	3/25/76	San Onofre	117,000
-	3/23/76	San Onofre	314,000
NAC - IB	3/15/76	-	-
-	4/2/76	San Onofre	81,000
IF - 300	6/23/76	Dresden	52,000
-	5/29/76	San Onofre	314,185
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SOURCE. Reports by NRC Region III Inspection & Enforcement

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* Exceeding limits in 10 CFR 20.205 (less than 22,000 disintegration per second per 100 cm^2).

** Converstion factor from CSAR p. 7-7 is $6.43 \times 10^{-7} \mu$ Ci/CPM.

*** Three additional excessively contaminated casks were not reported, per Region III IE Inspection Report No. 070-01303/75-04.

**** DOT exempted several Dresden shipments from 10 CFR 20.205b(2) reporting limit, per Region III IE Inspection Report 070-1308/76-05.

Yet, the Department of Transportation decided that the Dresden shipments would be exempted from the reporting requirement (for references, see footnotes on Table 3-1). It is not clear how many casks, what level of contamination, or what amount of occupational exposure was involved in this decision, or what the basis for this ruling was since the shipments traveled through areas of public access, and thus represented a risk to the people in that area.

After the smear test, the cask is vented and flushed to the Low Activity Waste (LAW) vault. If the flushed coolant measures too high a radiation level it indicates the cladding of the fuel contained within the cask may have failed. In the past, some failed fuel was shipped to Morris. For example, much of the Dresden 2 fuel was removed and shipped to MO because it failed. However, it is currently not planned to ship fuel known to be failed or leaking to the Morris facility. The cask flush test is supposed to detect fuel that has failed due to shipment and to warn the personnel involved that special procedures are required.(24) If the cask flush test indicates a large amount of fuel damage, the cask may not be received. When asked where it would go if not receivable, the Morris people did not have an answer.(25)

The flexible shielded lines involved in the cask flush are a potential source of radiation leakage in the event that a line rupture occurs when there is failed or leaking fuel in the cask being flushed.

3.2.6 CASK UNLOADING BASIN

Once the cask is flushed and the head bolts loosened, the cask is lowered onto the shelf of the unloading basin for removal of the head. From there it is lowered into the bottom of the "pit" or unloading basin for transfer of the fuel to baskets. Once the baskets are filled, they are lifted through a special gate into the storage basin. The special gate is designed to prevent tipping of the basket into the deep basin; an accident of concern since it could produce a criticality.

However, a perhaps less likely, but more serious accident could occur if the cask were to tip from the shelf into the deeper portion of the "pit". Such an event could occur during the head removal operation if one head bolt was left loosened and the crane, attempting to lift the head, instead tilted the cask (hung by the one loosened bolt). Failure of the single stud could release the cask in this tilted position. The falling cask could catch on the edge of the shelf long enough to disgorge the fuel bundles to the bottom of the pool. Misoperation of the radio-controlled cask handling crane during the movement from the shelf to the pit is another potential cause for this accident.

The resulting criticality could be more severe than that assessed by the Safety Evaluation Report (SER)[26] since it could involve up to 2½ times as many bundles (7 for the IF300 cask and 10 for NL 10/24 cask). The consequences would be increased if the cask subsequently fell onto the fuel causing physical failure of the cladding and releasing the fission products and gases within.

There has already been one reported incident in which an IF-200 cask was lifted during the head removal operation with one bolt inadvertently left in place.(27)

A considerable amount of effort has been put into establishing procedural controls and restraints on operation of the cranes, particularly in the unloading basin. Procedural control alone may not provide sufficient means of preventing accidents. The same consideration given to the basket tilt preventer gate is needed in the form of a cask tilt preventing device.

3.2.7 LOW ACTIVITY WASTE (LAW) VAULT

The radioactivity from the liquid processes of the MO is accumulated in the LAW vault. This 600,000 gallon underground tank is a reinforced concrete cylindrical vault incasing a carbon steel tank (phenolic coated) and is located in close proximity to the storage pool and the canyon of the main building. Contaminated liquids are sent to the LAW vault from the following sources:

- Basin water treatment waste
- Cask decontamination area(sump and cask flush)
- LAW evaporator and steampolisher bottoms
- Laboratory wastes
- Cell sumps (except LAW intrusion is sent to process sewer)

- Laundry
- Basin leak detection system
- Transfer of liquids from the cladding vault

Liquids are pumped from the LAW vault to the LAW evaporator in the canyon area. There they are concentrated with the vapors released to the stack via the sand filter and the residue pumped back to the LAW vault.

The current quantity of radioactivity in the LAW vault is reported at about 60 curies and increasing slowly.[28] This is a fairly low quantity and does not represent a large risk. However, over the years the vault and its piping will accumulate a larger and larger inventory of radioactive sludge. Using a nominal figure of 0.2 Ci/MTHM/yr stored,[29] this could accumulate as much as 10,000 Ci over a 25-year period. The accidental leakage or release of this material would then have a greater associated risk.

There are two stainless steel encased pipes which appear to run from the LAW vault to the basin water clean-up system and the cladding vault (see Figure 3-2). The design and protection of these pipes is not described in the CSAR, but they provide a potential leakage path during transfer to and from the LAW vault. Another area that is not well enough described in the CSAR to assess its importance is the security aspects of the LAW vault access hatches.

Although the LAW vault appears to be very substantial a structure, the liner appears vulnerable to chemical corrosion.

One example of accidental transfer of a small amount of acid solution has already occurred.[30] The accident was in conjunction with process testing and therefore is not likely to recur, but is nonetheless a concern.

REFERENCES

1. Consolidated Safety Analysis Report (CSAR) - Morris Operation NEDO-21326, Vols. 1 & 2, Jan. 1977, General Electric Company.

2. Final Safety Analysis Report - Midwest Fuel Recovery Plant - NEDO-10178, Dec. 1970, General Electric Company.

3. CSAR, p. 3-8.

4. NEDO-21624, Summary Environmental Report - Fuel Storage Facility Expansion, Apr. 1977, General Electric Company.

5. CSAR, Section 3.7.4 and map on page 3-55.

6. NEDO-21624, p. 8-1, 8-2.

7. CSAR, p. 3-10.

8. CSAR, p. 3-10. Other references are NEDO-21624 says 7.76 million by 1980, NRC Staff in their Environmental Impact Appraisal says 9.169

9. CSAR, p. 3-29.

10. CSAR, p. 3-24.

11. NEDO-10178, Amendment 3 to Supplement 1, 1978, General Electri Co., pgs. 2 and V-1. See also CSAR, App. A.15, Fig. A.15-1.

12. CSAR, Sections 3.7.3 and 3.7.4 and p. 3-57.

13. NEDO-20825, SER for M.O. Fuel Storage Expansion, Mar. 1975, General Electric Company, pgs. A-4 and A-5, see also NEDO-21624, pg. 9-3 and CSAR pgs. 7-3, 8-16, and 8-17.

14. CSAR, p. 7-36.

15. NEDO-21624, p. 9-3.

16. Final Environmental Statement - 40 CFR 190, EPA 520/4-76-016.

17. CSAR, Fig. 1-28, p. I-46.

18. CSAR, p. 7-8.

19. Letter from G. Montet (Argonne) to R. Cooperstein (NRC), April 30, 1975.

20. CSAR, p. A.15-26.

21. NRC Region III, IE Inspection Report No. 070-01308/75-04, pgs. 3 & 12.

22. CSAR, p. A.15-25.

23. 10 CFR 20.205b(2) requires 3 hours if arrival is during normal working hours and 18 hours if arrival in off hours.

24. CSAR 7-5, 7-6.

25. Appendix B, Morris tour notes.

26. The NRC in their SER for Morris expansion to 750 MTU, NR-FM-001, Dec. 1975, assumed 4 PWR bundles ($\sim$ 1.6 MTU) formed a critical mass resulting in 10^{19} fissions/MT.

27. NRC Region III IE Inspection Report No. 070-1303/75-04.

28. Appendix B, MHB Tour of MO.

29. NUREG-0116, Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle, Oct. 1976, pg. 4-110. They assume 200 ft^3 of solid waste per 1000 MTHM/yr at an average activity of 1 ci/ft^3.

30. NRC, Region III IE Inspection Report No. 070-01308/75-04.

SECTION 4

4.0 ACCIDENT SAFETY ANALYSIS

This section contains the evaluation of accident sequences which could cause risk to people in the vicinity of the Morris Operation or people who may be affected by the various pathways of radiation release. It is often stated in the CSAR that the storage of spent fuel is a passive and benign process. However, the present license allows 2.5 billion curies(1) of fission products to be stored in the MO pool; that is more radioactivity than in some reactors and is therefore worthy of cautious treatment. The proposed expansion will more than double this potential store of radioactive fission products plus adding several tens of tons of fissionable uranium and plutonium(2) to the MO inventory. The fissionable material must be kept apart to avoid accidental criticality.

For several of the radioactive elements in MO (e.g. cesium, strontium) small quantities dispersed to the environs can pose a health and safety hazard. Such a release of radioactivity may be the result of events such as accidents, errors, or sabotage.

The approach in this analysis is to evaluate different accident sequences by considering each accident initiator and defining the sequence of events which could follow, potentially leading to a radiation release to the environs. Combining these sequences creates what is termed an event tree.

Quantification of the events probability and consequences is a possible next step in evaluating overall risk. Although the absolute value of risk so derived is of questionable value, the development of a thorough event tree may be very useful in identifying vulnerable points in the system design. It is beyond the present scope of this study to quantify each branch of the event tree and its consequence, but the steps to accomplish this are discussed in Sections 4.2.2 nd 4.2.3.

This study uses as a base the safety analysis found in the CSAR, Chapter 8. The CSAR analysis defines the three basic steps leading to public exposure: Liquid pathway, the gaseous pathway, and direct radiation. From this base we have added other events which may contribute to the risk.

4.1 ACCIDENT INITIATORS AND SEQUENCES

The CSAR identifies nine major initiators, and in Chapter 8 discusses the sequence of events and possible consequence due to each of these.

- Criticality (basket spill)
- Cooling system failure
- Loss of make-up water sources (primary and back-up)
- Bundle drop accidents
- Basket drop accidents
- Cask drop accidents
- Missiles (tornado generated)
- Fin-fan cooler leak
- Earthquakes

To these, we add the following initiators:

- Construction-caused accidents
- Criticality (cask spill)
- Corrosion
- Tornado (evacuating pool)
- Sabotage
- Human error
- Pipe failure
- Cask handling error
- Cask overpressure venting

The following are brief descriptions of each of the added initiators with the postulated sequence of events which could produce a release of radiation and a risk to the public.

4.1.1 CONSTRUCTION-CAUSED ACCIDENTS

A major feature of the expansion is the addition of Basin 3 contiguous with Basin 2 and utilizing the expansion gate built into the end of the second basin. This presents a hazardous situation.

Section 3.2.4 described the construction plans which call for blasting the shale foundation material from the area of the new basin while the spent fuel remains in the old basin, immediately adjacent to the blasting. Several construction caused damage sequences can be envisioned: The failure of the south wall of Basin 2[3], the failure of the gate between

Basin 2 and the exposed pit for Basin 3, fracture of the foundation of Basin 2, rock missiles caused by the blasting and impacting Basin 2, fires in adjacent areas and equipment needed for the existing basins.

These accident initiators could cause damage to stored fuel, direct radiation exposure, leakage of contaminated pool water, and the need to remove the existing spent fuel to implement repairs. This last eventuality creates a dilemma. Obviously, if there were space available for such a transfer there would be no need for constructing Basin 3. The other question is whether or not casks could be found to make such a transfer in a short period of time. Clearly these are problems which need to be addressed in advance of any construction.

4.1.2 CRITICALITY (CASK SPILL)

The CSAR evaluates one general form of criticality and concludes it is not possible because of the grid spacing, poison effect of the basket material, procedural control, and conservative assumptions in the analysis. The NRC in their evaluation assumed a single basket of PWR fuel ($\sim$1.6 MT) is spilled and forms a critical mass but of low yield (10^{19} fission/MT) and of no serious consequences.(4)

The possibility exists for a larger number of bundles to be involved in a criticality accident. One way this could happen is during a cask spill. Section 3 discusses a sequence of events that could spill the contents of the cask onto the floor of the unloading pit, creating a criticality with the possible complication of physical damage to the fuel by the falling cask. The larger casks hold the equivalent of 2 to 2½ baskets of fuel (see Table 4-1). The possible consequences would be radiation exposure of personnel and gaseous release from the damaged fuel. It is unlikely but possible that a prompt criticality could produce a sufficiently large quantity of heat to form a steam bubble which would force some water out of the pool.

Another means of involving more than one basket is by the impact of a tornado-caused missile. The missile could shear off the hold-down pins on the impacted baskets, causing the tipping and spilling of adjacent baskets. The probability may be fairly low that such an event would produce the necessary configuration of bundles to create a major criticality. However, if the event did occur, there could also be added heat and radiation contributed by fuel adjacent to the spill but still in the baskets.

4.1.3 CORROSION

Despite the apparent passive nature of spent fuel storage, the fuel is subject to silent, slow, destructive forces such as corrosion of the fuel cladding. Most cladding which will be stored in the Morris facility is a zirconium alloy. Zirconium (or zircaloy) is less susceptible to corrosion than most metals, but one is dealing with the

TABLE 4-1

CHARACTERISTICS OF SPENT FUEL SHIPPING CASKS* & STORAGE BASKETS

Cask	Type	No. of Casks	Capacity (Assemblies) BWR	Capacity (Assemblies) PWR
NAC	Truck (Legal Weight) (NFS-4 type)	4	2	1
NFS-4	Truck (Legal weight)	2	2	1
NL 1/2	Truck (Legal weight)	3	2	1
IF-300	Rail	4	18	7
NL 10/24	Rail	4	24	10
TN 8	Truck (Overweight)	2	0	3
TN 9	Truck (Overweight)	3	7	0
NL 1/2	Truck (Legal weight)	2	2	1
Basket				
-	PWR Basket		-	4
-	BWR Baskets		9	-

*Source: Spent Fuel Storage Study 1976-1986, AIF, April 1977.

possibility of long periods of storage. If there is no clear resolution of the waste repository question, the MO may become the de factor waste disposal site for a large quantity of spent fuel. History for storing spent fuel in pools goes back only 20 years and the experience is primarily for low exposure fuel. There is less than 15 years experience with storage of high exposure fuels.(5) Therefore, it cannot be said with certainty what will happen to high exposure fuels stored for longer periods of time.

Water chemistry is vitally important to preventing corrosion of spent fuel cladding, and chlorine content is one key concern. Concentrations of less than one part per million ($<$ 1 ppm) of chlorine are recommended for long-term storage in pools.(6) The MO pools are reported to be between 4 and 5 ppm chlorine with the specified limit at 10 ppm.(7) This may prove to be a problem for long-term storage of higher exposure (longer reactor residence time) fuel. The chlorine can induce stress corrosion cracking or, in the case of weakened fuel, complete an already started process. Water acidity and iodine concentration are additional factors affecting cladding corrosion.(8) Once corrosion produces leaks through the fuel cladding, the gaseous fission by-products stored in the fuel gap and plenum have a chance to excape, pass through the water, through the sand filter and out the stack to the environs.

The CSAR has analyzed the release of radioactive fission gases due to a single basket drop and due to a tornado missile impact involving one basket (maximum of 6 bundles BWR fuel, 4 bundles PWR fuel) and concluded the accident would produce an insignificant result.(9) However, there were 1055 bundles stored in MO as of April 1977,(10) and the proposed pool expansion will permit more than quadruple that quantity.(11) Each fuel bundle contains from one to four thousand curies of Kr-85 depending on its exposure.(12) Thus, there may be millions of curies of fission gases in the fuel already stored at MO, and this number may also quadruple. If a major corrosion problem developed to the point of affecting the integrity of many fuel bundles, the leaks could result in the release of large quantities of fission gases, much larger than presently analyzed in the CSAR.

Several mitigating influences exist in the present design which could reduce the chances and impact of such a release. The chlorine concentration is measured periodically (although chlorine content and water conductivity are not listed as quantities monitored in the control room)(13) and any radiation releases would be sent out through the stack and thus greatly diluted. On the other hand, factors which increase the chances of a corrosion-caused release accident are the higher exposure fuel likely to be stored in the future, the possibility of human error introducing higher chlorine concentrations into the pool, and the possibility of malfunction of the make-up water demineralizer causing excessive chlorine levels (well water on the site is about 100 ppm chlorine, ten times the limit for the pool).(14)

4.1.4 TORNADO CAUSING REDUCED POOL WATER LEVEL

The CSAR analyzes the impact of missiles on the pool liner and spent fuel, but the CSAR does not consider the possibility of the tornado actually removing a large portion of the cooling water from the pools. The tornado analysis assumes that the sheetmetal building covering the pool is blown away by the force of the tornado.[15] The possibility of the combined event of a tornado markedly reducing the pool level as well as introducing missiles impacting the fuel and/or liner should be evaluated.

Water level reduction could have two effects: (1) to expose the fuel, thereby creating a cooling problem, and (2) to increase the damage that missiles would inflict on the fuel and/or liner. This accident sequence appears to be excluded from the CSAR.

Additional analysis is required to determine if fuel melting occurs in the event cooling is lost. Some experts feel that spent fuel discharged more than 3 months and stored in conventional racks might not melt even if provided only with air cooling. However, one expert concludes that if melting did occur, it would likely melt through the basin liner.[16] The impact of dense storage and longer exposure fuel would need to be considered before a conclusion could be reached for MO.

4.1.5 SABOTAGE

The CSAR includes sabotage in the accident analysis event diagram with the footnote:

> "Effects of sabotage [are assumed to be] equivalent to natural events or accidents." (17)

The CSAR treatment of the potential of sabotage is cursory because almost all of the accident analyses involve only single system failures or single fuel basket events. Clearly sabotage has the potential of involving numerous baskets, numerous systems, or even the entire inventory of spent fuel.

Recently, a great deal of attention has been focused on establishing deterrents, barriers and procedures against sabotage in order to meet the latest NRC regulations.(18) The effect of this action is to reduce the chances of a successful sabotage, but it does not eliminate it as a possible accident initiator. The regulatory attention and publicity may in fact result in an increase in the number of sabotage attempts.

The spent fuel pool is sometimes considered less susceptible to sabotage than many other targets in our society. The saboteur's motive might be to destroy the facility rather than any interest in the dispersion of the tens of tons of SNM which are contained in the fuel rods. The preceding motive, however, increases the number of options available to the saboteur. Vulnerable points include the expansion gates, cooling systems, LAW vault, and the fuel itself. For example, the storage pools lack a physical barrier between the observer and the water. The basin wall rises 2 or 3 feet above the walkway but otherwise there is litt' to prevent a person from dropping something into the pool. A more indirect approach could involve the high speed impact of a train car (such as a cask carrier) into the cask receiving area which could result in the launching of a missile or projectile into the unloading basin with the intent of damaging the pool liner and releasing the coolant.

Most likely, a successful sabotage attempt would require explosives and a person on the inside. During the construction of the proposed expansion, many construction personnel and a considerable quantity of explosives will be in the immediate vicinity of the existing spent fuel pool and supporting equipment. The applicant's security forces and surveillance requirements will certainly need to be expanded during the construction period.

4.1.6 HUMAN ERROR

The MO relies on procedural control of processes and personnel actions in many cases related to the handling and movement of radioactive materials. Procedures are only as good as the people attempting to follow them. There are numerous examples where people have violated the procedures or made errors which created a potentially dangerous situation. A very real possibility exists that human error could lead to a serious accident. Examples of human error related problems experienced at MO include:

- The cast tip accident in 1972 that ruptured the pool liner.

- Attempted to remove the head of an IF 200 cask with one bolt still engaged, causing entire cask to lift.
- Crane incident where NAC-1 cask head hit the scaffolding on the decontamination pad.
- A fuel basket hook was dropped during head replacement operation.
- Acid solution inadvertently transferred to the LAW vault while process testing.
- An acetone fire occurred during a welding operation.

None of these incidents resulted in a serious release. However, the possibility exists that other human errors could lead to serious accidents that are not as easily controlled and not as forgiving in their impact. In addition, no NRC regulations exist governing the licensing of ISFSF operators. The lack of a formal licensing procedure may increase the likelihood of human error.

4.1.7 PIPE FAILURE

Failures of some critical pipes could release fairly large quantities of radiation to the local evir The cask flush line carries the flushed coolant w ich be contaminated with radioactive gases (Kr-85 mainly) and fission products (e.g., Iodine and Cesium isotopes) due to failed fuel in the cask. Depending on the number and type of fuel failures, this could be 1000's of curies of gas and 100's of curies of non-gaseous fussion products.

The pipe which carries the flow between the LAW vault and the cladding vault is another example (see Figure 3-2).

This could contain concentrated waste and sludge and leaks from it would be in direct contact with the soil. There may be other systems and pipes in MO where failures could release radioactivity. There is at present no known large radiation source on the Morris site except the stored fuel itself. However, after a considerable period of time, the accumulated LAW vault inventory will increase to the point where it would be extremely hazardous if released.

The LAW vault sludge will consist of concentrated extractions from the fuel pool coolant, filters, resins, and cask flushing liquids. The largest contributing source of radioactive contamination of an ISFSF is shedding of the deposited radioactive material on the surface of the fuel elements. These deposits are referred to as "crud" and consist of activated corrosion products and the components of failed fuel. The major corrosion products are generally Co-58, Co-60, and Mn-54. The major fission products expected are Cs-134, Cs-137, Ru-106, Rh-106, Zr-95, Nb-95, Sb-124, Ce-144, I-129, I-131.[19] Other sources of contamination of these pools are contaminated coolant from shipping casks, some tritium in combined form, and some disolved radioactive gases. Using a buildup rate of 0.1 Ci/day,[20] this will result in a hundred curies buildup of LAW vault inventory every few years or thousands of curies over the lifetime of the facility. It takes more than a million gallons of water (about 3 acre-feet) to dilute just one curie of Cs-134 to an acceptable level.[22] Thus, inadvertent release of even a fraction of the LAW vault inventory via a pipe break, sabotage or other means could contaminate a large volume of surface or ground water.

4.1.8 CASK HANDLING ERROR

While in the cask, the spent fuel is in a potentially hazardous condition. The cask has a limited coolant supply, and the fuel is subjected to the vibration and shock of transit in the direction of its weakest dimension. The spent fuel must survive this following exposure to as high as 44,000 MWD/MTHM* with shipment from the reactor as early as three months after discharge.[22]

In receiving the cask and moving it into the unloading pool, there are several critical steps where an error in handling could expose the workers to the high radiation of the fuel, cause drop or damage of the cask, or release some of the radioactivity contained in the cask.

The critical steps involved include the lifting, venting, flushing, head bolt loosening, lowering onto the unloading shelf, head removal, lowering into the unloading pit, removal of fuel, and recovery of the cask. Many of these steps are comparable to the processes involved in moving fuel at a reactor. These

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MWD/MTHM: Megawatt days (thermal)/metric ton of heavy metal (mainly uranium).

processes have been shown to be subject to errors on at least seven occasions in the period 1974-1977, when BWR fuel bundles have been dropped while being moved.[23] Weaknesses in design procedures and personnel actions contributed to these reactor incidents. Similar errors and incidents are possible at the MO site.

4.1.9 CASK OVERPRESSURE VENT

Many casks vent into special containers which are attached indirectly to the cask, while other casks may vent to the atmosphere.

Failure to properly vent the cask or an unexpected pressure rise during receiving resulting in uncontrolled venting, could release radioactive gases and water to the environment, causing, as a minimum, exposure to operating personnel.

4.1.10 SPENT FUEL POOL COOLING LIMITATIONS

The heat generated by a bundle of spent fuel decays at a rate determined by the half lives of the various fission products, actinides and fissionable materials contained in the fuel bundle. The quantity of these materials present in any particular bundle is a function of the history of the bundle (i.e., the exposure in MWD/MTHM and the specific power rate of operation in MW/MTHM), and the length of time the fuel bundle has been discharged from the reactor.

The CSAR upper exposure limit for MO fuel in 44,000 MWD/MTHM with shipment to MO no sooner than 90 days after discharge from the . actor. However, many of the analyses in the CSAR used values as low as 2400MWD/MTHM and cooling and times up to one year.[24] Where less than maximum conditions are used, they are justified on the basis of the currently stored fuel (about 300 MTHM) being less than 15,000 MTHM average exposure. This bears further analysis as does the impact of adding higher exposure, shorter cooling time fuel to fill the expansion project.

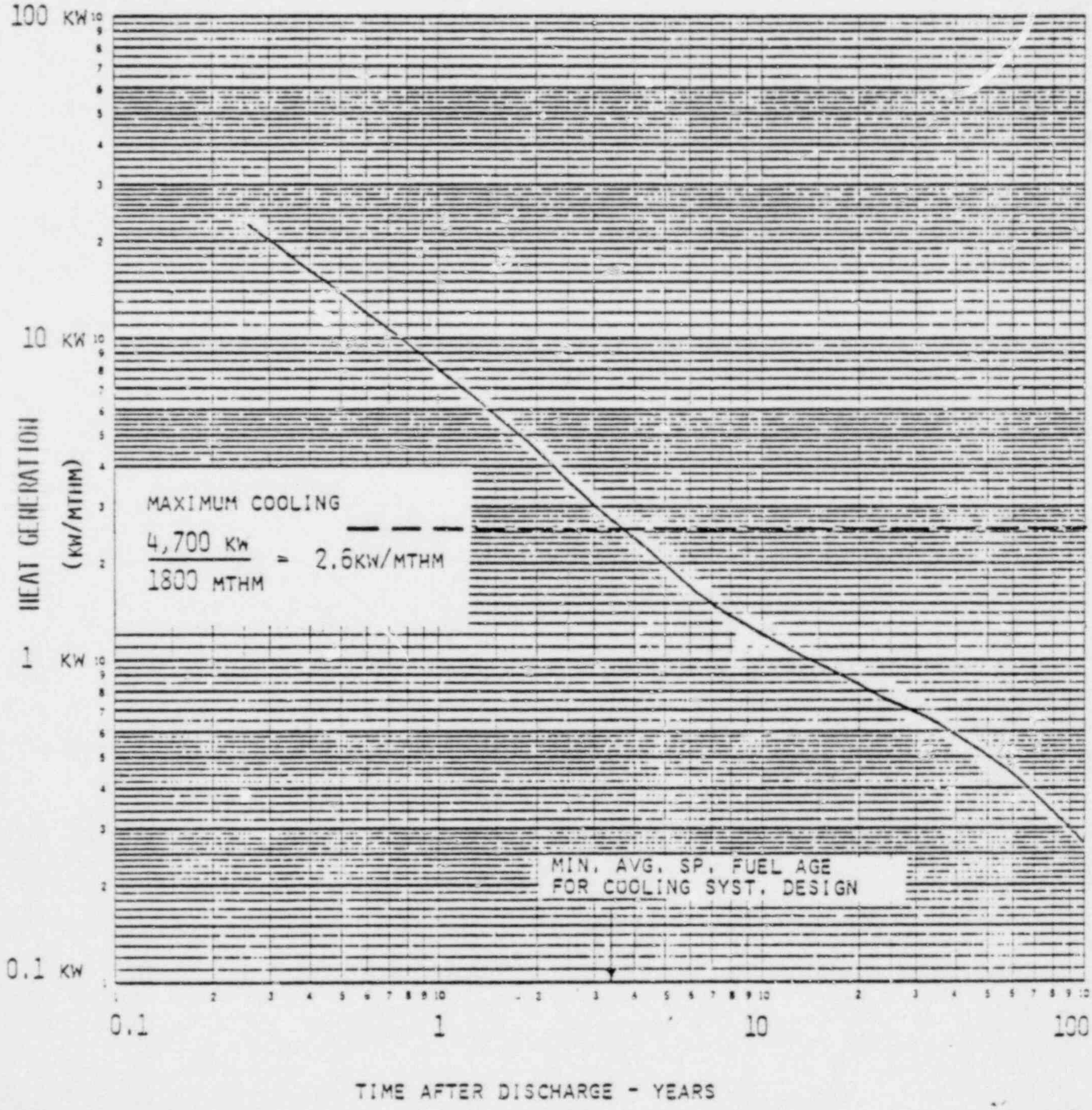
The average heat generation (KW/MTHM) of fuel as a function of time after discharge is shown in Figure 4-1. This curve is based on fuel with an exposure of 25,000 MWD/MTHM and 35 MW/MTHM specific power, which is consistent with many of the CSAR analyses and may represent an average condition, but is still not a worst case analyses. From this curve, the presently installed and planned cooling capacity of 4.7 MW is seen to be adequate for an average fuel cooling time (after discharge) of three to four years. Shorter average cooling time for the stored spent fuel may require additional cooling capacity.

Another critical variable is the rate at which spent fuel is received at the MO. Figures 4-2 A and 4-2 B analyze two cases which show the impact of rate of arrival of spent fuel. Assuming that one year old fuel is received at 200 MTHM per year, the pools will be filled in seven years, and the cooling system is marginally able to accomodate the resulting inventory of spent fuel and its generated heat.

Case B shows the same one year old spent fuel, but received at 100 MTHM per year. For this case, the pool cooling capacity is clearly adequate. However, for spent fuel which

SPENT FUEL HEAT GENERATION - KW/MTHM VS TIME

SOURCES: ERDA 76-43, PG 2.36
DOE/ER-0004/D PG 116
CSAR, PG 5-40



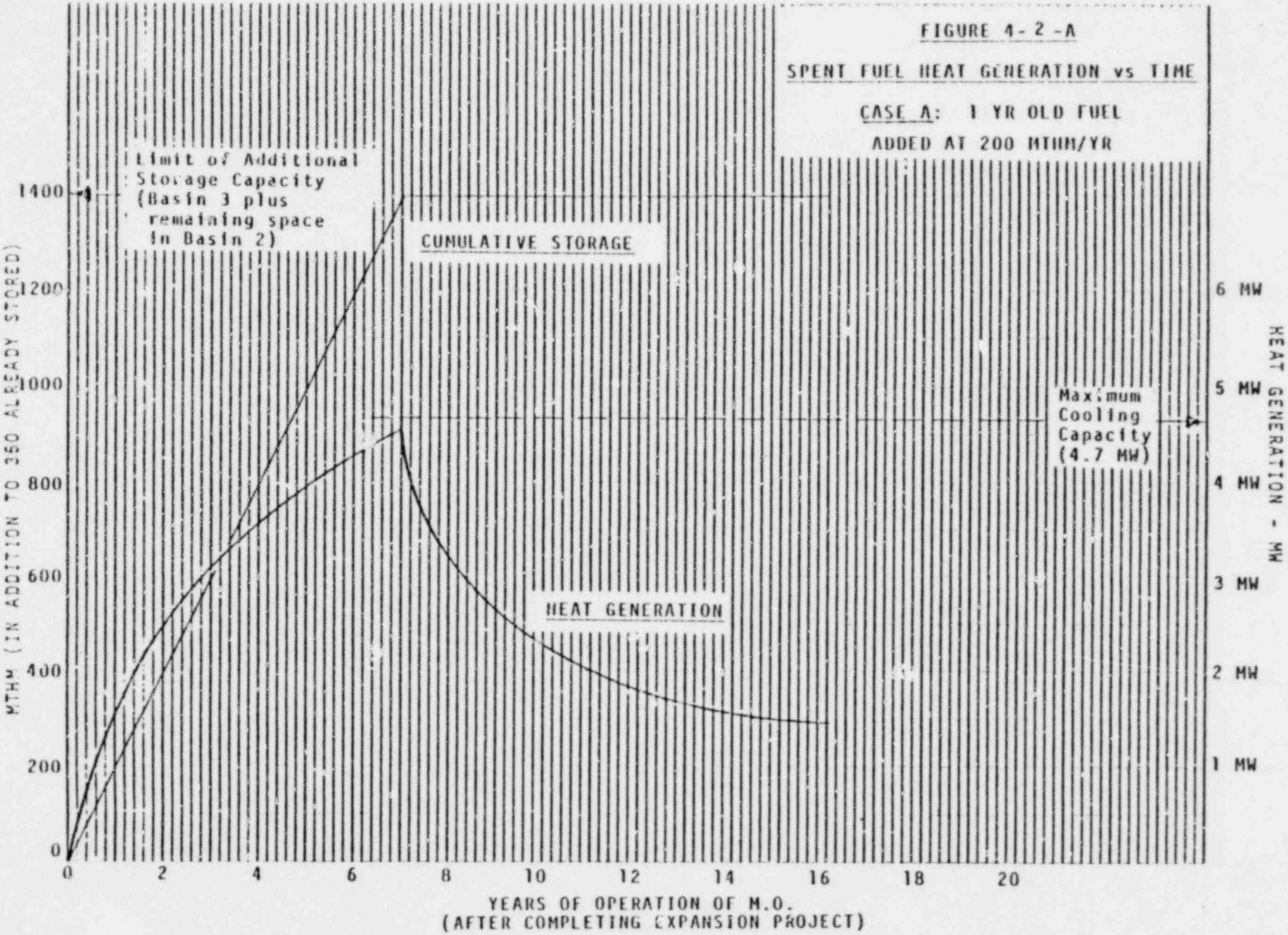


FIGURE 4-2-A
SPENT FUEL HEAT GENERATION vs TIME
CASE A: 1 YR OLD FUEL
ADDED AT 200 MTHM/YR
Limit of Additional Storage Capacity (Basin 3 plus remaining space in Basin 2)
CUMULATIVE STORAGE
HEAT GENERATION
Maximum Cooling Capacity (4.7 MW)
MTHM (IN ADDITION TO 350 ALREADY STORED)
0
200
400
600
800
1000
1200
1400
HEAT GENERATION - MW
1 MW
2 MW
3 MW
4 MW
5 MW
6 MW
0
2
4
6
8
10
12
14
16
18
20
YEARS OF OPERATION OF M.O.
(AFTER COMPLETING EXPANSION PROJECT)

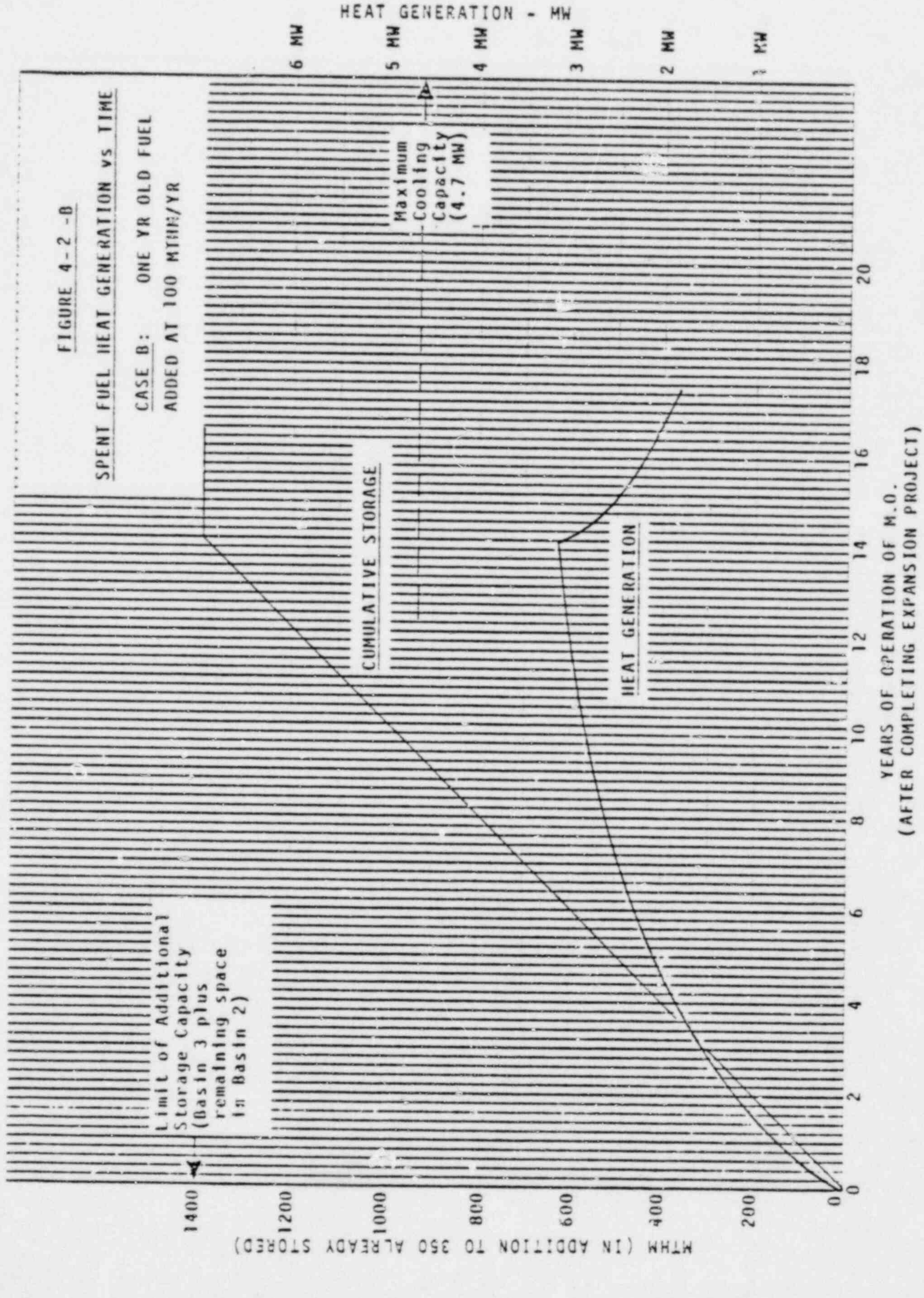


FIGURE 4-2-B
SPENT FUEL HEAT GENERATION vs TIME
CASE B: ONE YR OLD FUEL
ADDED AT 100 MTHM/YR
HEAT GENERATION - MW
6 MW
5 MW
4 MW
3 MW
2 MW
1 MW
Maximum Cooling Capacity (4.7 MW)
CUMULATIVE STORAGE
HEAT GENERATION
Limit of Additional Storage Capacity (Basin 3 plus remaining space in Basin 2)
MTHM (IN ADDITION TO 350 ALREADY STORED)
0
200
400
600
800
1000
1200
1400
0
2
4
6
8
10
12
14
16
18
20
YEARS OF OPERATION OF M.O.
(AFTER COMPLETING EXPANSION PROJECT)

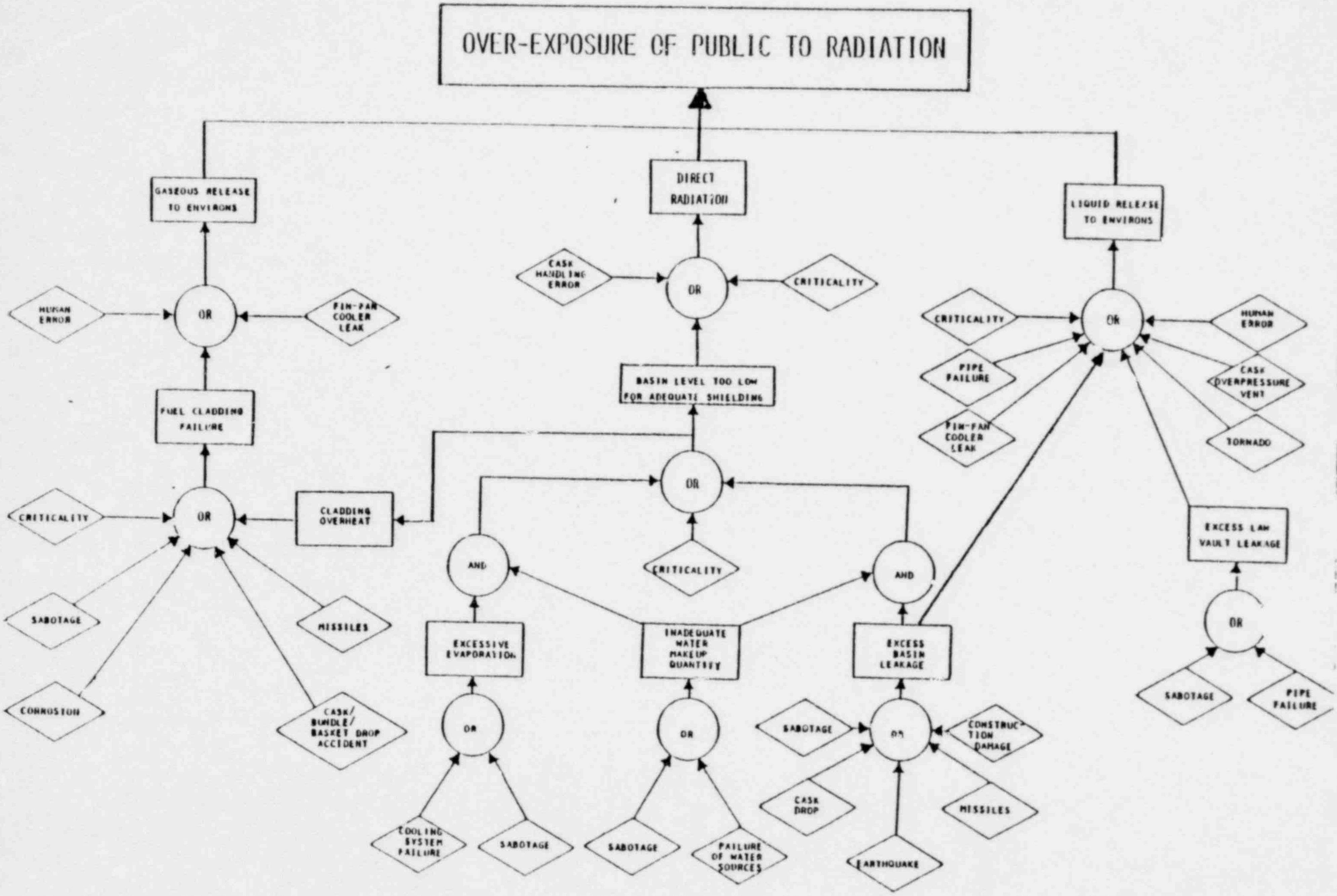
has been less than one year out of the reactor and/or received at a rate greater than 200 MTHM/year,[25] the cooling capacity appears to be marginal or inadequate. The CSAR does not analyze the maximum rate of receipt of spent fuel with the upper limit conditions of exposure, specific power and recency of discharge.

4.2 ACCIDENT PROBABILITIES AND CONSEQUENCES

4.2.1 EVENT DIAGRAMS

Each accident initiator discussed in the preceding subsections contributes to one or more pathways which could result in release of radiation to the environment. The event sequences are defined in CSAR as the Liquid, Gaseous, and Direct Radiation pathways. By defining intermediate states of the critical variables, postulated accident sequences or events can be diagramed. Figure 4-3 shows the resulting Event Diagrams for the three pathways. For ease of comparison with the CSAR Event Diagram,[26] the same symbolism is used. The diagrams are read from bottom to top with initating events shown in diamonds and "or" symbols indicating that any one of the inputs to the "or" symbol can cause the output state to exist.

Whereas some of the event sequences may mainly affect the workers at the site (e.g., cask overpressure venting), in general, the accident sequences lead to exposure of the general public in the vicinity of the MO. Over exposure may be defined as exceeding the permissible limit as given in 10 CFR 100.



OVER-EXPOSURE OF PUBLIC TO RADIATION
GASEOUS RELEASE TO ENVIRONS
DIRECT RADIATION
LIQUID RELEASE TO ENVIRONS
OR
HUMAN ERROR
FIN-FAN COOLER LEAK
CASK HANDLING ERROR
CRITICALITY
PIPE FAILURE
CASK OVERPRESSURE VENT
TORNADO
BASIN LEVEL TOO LOW FOR ADEQUATE SHIELDING
FUEL CLADDING FAILURE
CLADDING OVERHEAT
SABOTAGE
MISSILES
CORROSION
CASK/ BUNDLE/ BASKET DROP ACCIDENT
AND
EXCESSIVE EVAPORATION
INADEQUATE WATER MAKEUP QUANTITY
EXCESS BASIN LEAKAGE
EXCESS LAW VAULT LEAKAGE
COOLING SYSTEM FAILURE
FAILURE OF WATER SOURCES
CASK DROP
CONSTRUCTION DAMAGE
EARTHQUAKE

FIGURE 4-3

Because data is lacking, quantification of the potential range of accident probabilities consequences was not attempted at this time. The risks resulting from these additional sequences should be thoroughly addressed during the future licensing proceedings to ensure they are adequately evaluated and that proper precautions are taken to prevent their occurrence. The general steps for this risk assessment are described in the following sections.

4.2.2 ACCIDENT PROBABILITY

To complete the assessment, each of the accident initiators should be evaluated in terms of likelihood of occurrence (in units such as events/year, events/demand, or events/fuel assembly moved). Actual experience may not be available for all the initiators considered. In such cases, comparable experience or engineering estimates must be made. Multiplying the probabilities of the events in the event tree (assuming independent events) will give the overall probability of the sequence occurring. Clearly, this will require consideration of both release and dispersion mechanisms which requires that meteorological and demographical data be analyzed. This process should be completed for each sequence starting with each initiator.

The resulting probability data will indicate the most likely paths and thus the weakest links in the existing design. But this alone is not sufficient to justify changes. One must first look at the consequences due to each sequence.

4.2.3 ACCIDENT CONSEQUENCES

The calculation of consequences involves an estimation of the amount of radioactive material released, the manner in which it is dispersed, the population it comes into contact with, and the expected health and safety effects it creates. For the different categories of release, the health effects would be evaluated for early fatalities and illnesses as well as long term fatalities and illnesses. Having both the probability values and the consequence values, the relative values risk (estimated as the product of probability and consequence), are useful in deciding what action or improvements in the facility or operating procedures are desirable to reduce the risk to the public.

REFERENCES

1. SER - Div. of Fuel Cycle and Material Safety, US NRC NR-FM-001, Dec. 1975 page 1. The level of 2.5×10^9 curies of fission products is the limit based on a heat production rate of 4.4 megawatts. This applies to the current MO license as amended to cover 750 MTHM of stored spent fuel.

2. If the licensed limit is raised to 1850 MTHM of spent fuel (originally enriched to an average of 3%), and approximately one-half of the original fissionable material remains, M.O. inventory would be roughly 1850 x .03 x .5 $\cong$ 28 MT fissionable uranium plus the plutonium 239 generated during operation. The present licensing revision allows 37.5 MT U-235 and 9MT Pu-239, as discussed in the SER by Div. of Fuel Cycle & Material Safety, NR-FM-001, 1975.

3. CSAR, Appendix A.15, Section A.15-4.3, page A.15-26 identifies this as a potential weakness.

4. SER, NR-FM-001, USNRC, Div. of Fuel Cycle and Material Safety, Dec., 1975, pgs. 21 & 22.

5. Refer to Appendix C for a review of the corrosion problems and extent of experience with spent fuel.

6. Appendix C.6, J. R. Weeks, Corrosion of Materials in Spent Fuel Storage Pools, July 1977.

7. NRC Region III, IE Inspection Report No. 070-1308/76-01.

8. Appendix C.6 and C.7.

9. CSAR, Sections 8.7 and 8.8.

10. SER, Fuel Storage Facility Expansion, NEDO 21624, April 1977, p. 1-1, and CSAR p. 5-6.

11. ERDA 76-43, Vol. 1, pg. 232 defines an average bundle as about 0.3 MTHM. Thus, 1850 MT could have over 5,000 bundles depending on the ratio of BWR to PWR bundles.

12. A typical fuel assembly may have several thousand curies of krypton gas after discharge from an operating reactor. See, for instance, CSAR, p. 4-2, pgs. 8-16 through 8-19, and ERDA, 76-43, Vol. 1 pg. 235.

13. CSAR, p. 5-57a.

14. CSAR, p. 3-41

15. CSAR, p. 8-26 and p. 8-30.

16. David McCloskey, Sandia Labs., presentation to California Energy Commission, March 10, 1977. Transcript pgs. 112-114.

17. CSAR, Figure 8-1, p. 8-3.

18. 10CFR 73.55.

19. ERDA-76-43, Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle. May, 1976, Vol. 1, pg. 2.39.

20. ERDA 76-43, Vol. 1, pg. 2.42 shows an expected value of solid waste from a spent fuel pool with 800 fuel assemblies to be $0.6 m^3/yr$ with about $4\ Ci/m^3$.

21. Maximum permissable concentration of Cs-134 (MPC_w(occupational) = $3 \times 10^{-4}\ \mu Ci/ml$ or $\sim 1 Ci/10^6$ gal.)

22. Analyses have been done in the CSAR using periods as short as 90 days from the time of discharge, but CSAR, pg. 7-10 states that current practices are to prohibit shipping fuel with less than 120 days cooling time.

23. Inspection & Enforcement Circular 77-12, USNRC, September 15, 1977.

24. CSAR, p. 4-4a.

25. The CSAR uses the receiving rate of 300 MTHM/YR in some analyses, which is a carry over from the reprocessing plant design.

26. CSAR, Figure 8-1, p. 8-3.

SECTION 5

5.0 FUTURE RISKS

The primary purpose of this risk analysis has been to (a) review the Morris existing and proposed expansion program (Project IV) facilities and to (b) evaluate the accident possibilities for the present and future spent fuel storage program. This assessment is necessarily limited by the uncertainties that face the spent fuel/waste disposal program in the United States. There are many factors which contribute to the uncertainty of MO risk assessment. Several of the major factors are discussed in the following subsections.

5.1 HIGHER EXPOSURE FUELS

The current state of the nuclear fuel cycle is forcing a change on the nature of spent fuel that will be entering the storage cycle. The elimination of the reprocessing step provides incentive to the utility to drive existing and future fuels to high burnups so as to extract as much fission energy as possible. Early fuel performance, characterized by frequent failures, has been evaluated and it is now apparent that most failures have been influenced by the rate of core power level change.(1) Change rate restrictions have been imposed and fuel life appears to be increasing.

Fuel design goals now are aiming at exposures of 40,000 MWd/MTHM or greater.(2) These factors, coupled with the trend towards stretched refueling cycles to improve plant availability, all contribute towards more severe fuel duty and higher exposures. This condition is confirmed in the most recent Department of Energy report on spent fuel [(3)] which indicates average exposure of 31000 MWD/MTHM (270 for BWR, 33000 for PWR) forms the basis of their planning.

Long-term (greater than ten years) fuel storage experience in the U.S. is predominantly with low-exposure fuel (~ 10,000 MWD/MTHM.)[(4)] The addition of 30 - 50,000 MWD/MTHM exposure fuel to the stock will mean substantially higher per-unit inventories of fission products and a partially degraded (weaker) primary release barrier (the cladding). These changing conditions with storage times beyond current experience introduce some uncertainty into Morris' future operation.

5.2 OTHER THAN LWR FUELS

There are many unknowns in the source and type of future fuels that may be operated and stored. One gas-cooled reactor is now in operation generating spent fuel. Breeder reactors may eventually go into operation and produce extremely high burn-up fuel. Decommissioning of nuclear facilities may produce some highly radioactive materials which will require long-term storage. Storage of the radioactive materials resulting from any of these operations would be possible and likely at MO, especially if MO should turn out to be the only licensed AFR with available capacity.

5.3 USE OF MO AS A SURF

The Spent Unreprocessed Fuel (SURF) Facility is a fuel storage concept wherein spent fuel is encapsulated an stored in one of various forms of retrievable storage, pen decisions on ultimate disposal or reprocessing. The optic considered include both dry and wet storage, above and bel ground. A recent study[5] considered the details of sever alternates at a Hanford, Washington site. One of these is air-cooled vault storage facility which consists of a long heavily-shielded concrete vault with air flow passing thro from one end to the other for fuel cooling. It takes litt imagination to substitute the MO main reprocessing canyon (slightly renovated) as an alternative or supplement to th Hanford reference plant. The schedule of this same study envisioned a 1985 start in converting the 9,000 MTU spent expected to be in AFR storage at that date. Complete tran of all AFR stored fuel plus additional fu produced would complete the transition by 1998.

GE has already conducted studies of converting the canyon to an air storage facility for fuel cooled over fiv years,[6] a plan very similar to the SURF concept. If thi were to be done, the time commitment for the life of MO wo be considerably extended since the design life of a SURF i expected to be at least 50 years. Transition by default o such a facility into a permanent repository is a definite possibility.

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5.4 CHANGING REGULATIONS

Present NRC regulations cover processing facilities and reactors, but do not specifically address independent storage facilities for reactor spent fuel. This results in less rigid review of such facilities. For example, the first Licensed Fuel Facility Status Report,[7] lists the MO as a licensed facility, but does not require MO to report its effluent data even though it is licensed to have 2.5 billion curies of fission products in inventory.

The licensing of independent spent fuel storage facilities has been performed to date totally without the benefit of specific licensing regulations. As stated by the NRC in the Draft Generic Environmental Impact Statement on the handling of spent reactor fuel:[8]

> "Pertinent sections of 10 CFR Parts 19, 20, 30, 40, 51, 70, 71, and 73 now apply to spent fuel storage installation. These regulations cover the possession of special nuclear materials, but were promulgated to cover such possession incidental to manufacturing type operations. These regulations do not specifically cover spent fuel storage only type operations under static storage conditions." (emphasis added)

The NFS West Valley plant was licensed under 10 CFR Part 50, and the GE Morris plant under 10 CFR Part 70. Title 10, CFR, Part 50 covers the licensing of production and utilization facilities and, as such, is primarily intended for use in the licensing of electrical production facilities, while Part 70 applies to the possession of special nuclear materials. When the decision was made by General Electric to forego the start-up of the fuel reprocessing plant, it was obvious that the

Appendix F to Part 50, which specifically addresses only fuel reprocessing plants, would not apply. Accordingly, the Morris operation was licensed under Part 70 even though, as pointed out in NUREG-0404,[9]:

> "........... the pertinent requirements of 10 CFR Part 70 are worded in general language and require interpretations in specific licensing actions."

NUREG-0404 identifies the need for a more definitive regulation base and indicates[10] that:

> "a proposed new rule 10 CFR Part 72, Licensing Requirements for Storage of Spent Fuel in an Independent Spent Fuel Storage Installation, is being prepared."

The establishment of these regulations will provide a basis for the licensing of an AFR and, presumably, for a spent fuel waste repository.

These new part 72 regulations have just been issued for comment. To the extent that the regulations may dictate facility modification, new risk may be incurred.

5.5 DE FACTO WASTE STORAGE SITES

As resolution of the waste disposal dilemma continues to be delayed, the likelihood of MO becoming a de facto waste storage site increases. With many years backlog of military wastes to dispose of, the probability is high that the military waste will take priority in the permanent disposal program. A series of AFR's and expanded reactor spent fuel storage pools is therefore the most probable scenario for the mid-term future.

REFERENCES

1. Roberts, A. & Ocken, H., Improving Nuclear Fuel Performance, EPRI Journal, Oct. 1978.

2. Zirconium Behavior in a Nuclear Environment, EPRI Journal, Jun. 1978.

3. DOE/ET-0055, Preliminary Estimates of the Charge for Spent-Fuel Storage and Disposal Services, Jul. 1978, page 14.

4. BNWL-2256/UC-70, Johnson, A.B., Behavior of Spent Nuclear Fuel in Water Pool Storage, Sep. 1977.

5. Spent Unreprocessed Fuel Facility - Engineering Studies, Rockwell Hanford Operation and Kaiser Engineers, RHO-LD-2, Feb. 10, 1978.

6. SER - Fuel Storage Facility Expansion - NEDO-21624.

7. Licensed Fuel Facility Status Report, NUREG-0430, Vol. 1, No. 1, May 1978.

8. NUREG-0404, Draft GEIS on Handling and Storage of Spent Light Water Power Reactor Fuel, Vol. 1 & 2, Mar. 1978.

9. Ibid, page 3-13.

10. Ibid, page 3-13.

SECTION 6

6.0 DECOMMISSIONING

As discussed in Section 5, MO will undoubtedly operate well into the next century. During this time, it will receive large quantities of spent fuel and, consequently, will accumulate radioactive material through operation of the process loops. If current regulations prevail, ultimately MO will need to be decommissioned. The financial responsibility of this will fall on whomever has title to the facility at that time.

There are presently no specific NRC regulations covering the licensing and ultimate decommissioning of a spent fuel storage facility although such regulations are scheduled for issuance in 1980-81[1] and proposed Part 72 Licensing Requirements for ISFSI (facilities) were published for comment in the October 6 Federal Register.[2] The regulations, when established, will define requirements for the ultimate disposition of the Mo facility.

Decommissioning uncertainty is reflected in the wording of sections of the CSAR, Chapter 4.5.2, "Proposed Decommissioning Methods," refers to plans of sealing, immobilizing, restricting access, and/or solidifying in place. In general, the plan described follows "entombment" philosophy rather than the return of the site to original or unlimited access condition.

A review of MO operations identifies several areas where radioactive material will accumulate in relatively large quantities. These are:

- LAW vault
- Evaporator
- Cladding vault
- Sand filter
- Pool filters
- Grid structures (pool)
- Fin-fan cooler and piping
- Drains and sumps
- Chemical vault

Potential problems associated with MO decommissioning, assuming total removal of all radioactive components, are discussed in the following subsections.

6.1 LAW VAULT

The LAW vault consists of a 600,000 gallon steel tank, housed in an underground reinforced concrete structure. The slurry (sludge) of all contaminated waste liquids are to be collected, concentrated and stored in the LAW vault throughout the life of the facility. Disposition of the LAW vault sludge will probably be accomplished by remote pumping and mixing with concrete in barrels for burial at another waste site.

Depending on the age of the slurry and the extent of solidification of the bottoms, this may be more or less effective in remote removal of most of the sludge. Semi-manual removal of solidified material may be required.

The steel tank (walls), sumps, pumps, and piping will also need to be removed as will portions (if not all) of the concrete strucutes. Full removal could require blasting with chance for release of some of the surface contamination to the environs. Removal will probably not be an insurmountable problem provided proper decontamination is performed in advance so as to control or minimize the release of contamination. Such decontamination, however, will probably require the addition of temporary processing equipment since the LAW vault would not then be available to receive such effluent. In general, the cost and effort required for LAW vault disposal are large but not insolvable.

6.2 EVAPORATOR

The location of the evaporator in the canyon will be an aid in decommissioning the device. This provides the capability for decontamination with good ventilation control by passing the gases through the sand filter before release. Care will be required during removal to avoid contamination of other non-contaminated equipment, or of the massive concrete canyon structure itself.

6.3 SAND FILTER

The sand filter is a large building (75 x 80 x 15 ft) filled with graded gravel and sand. For decommissioning, the problem will be to remove the filter media (sand) without remobilizing the particulates the filter has removed from the airstream over the life of the plant. A water backwashing scheme is proposed in the CSAR which would generate a large volume of contaminated water. The water would be collected in the canyon decontamination cell for subsequent treatment (presumably involving the evaporator if it has not been previously decommissioned).

An alternative is to mix the filter bed material into concrete in 55-gallon drums and ship it away for burial. This method will require control of the air to prevent contamination of the environment by the remobilized particles. In actual practice, a combination of both methods will probably be required.

6.4 POOL GRID STRUCTURES

The old grids from Basin 1 and 2 were cut up and shipped to a waste burial site. The same plan could be implemented for disposal of grids and liner plate in the future. However, they will have received much longer exposure and much greater crud buildup, requiring greater care in handling.

6.5 PIPES, PUMPS, FILTERS, ETC.

Most structures of this size are small enough or can be cut up into small enough pieces to be drummed and hauled to a waste burial site for disposition.

6.6 SEQUENCE OF DECOMMISSIONING

Perhaps the most difficult part of a total decommissioning plan is to sequence the operation properly so as to take advantage of existing clean-up and radiation control equipment. Specifically, the evaporator, sand filter and LAW vault are vital systems that will be needed during the decommissioning process and must be kept in service as long as possible. This will help to reduce on-site/off-site exposure--an essential goal of decommissioning. Ultimately, decommissioning will necessitate the procurement and utilization of temporary waste receiving, ventilation, and control equipment to permit the removal of these essential items.

The feasibility and cost of decommissioning a similar facility, the Barnwell Reprocessing Plant, was studied by BNWL for the NRC.[3] The sequencing, schedule, and cost considered in that study would probably be similar. It was estimated that complete dismantlement would take approximately $58 million. Assuming that estimate is accurate, MO dismantlement would probably be somewhat less than that due to the fact that less highly-contaminated equipment would be required to be handled.

REFERENCES

1. Report to the President by the Interagency Review Group on Nuclear Waste Management, TID-28817 (Draft), October 1978, pages 84-95.

2. Federal Register, Vol. 43, No. 195, Friday, October 6, 1978, pages 46309-46321.

3. Technology, Safety, and Costs of Decommissioning a Reference Nuclear Fuel Reprocessing Plant, NUREG-0278, October 1977.

SECTION 7

7.0 SUMMARY AND RECOMMENDATIONS

Compared to reactor operation, the operation of the MO spent fuel facility is a relatively passive process. MO utilizes neither high technology nor complicated quipment. However, even before the proposed expansion, the quantity alone of authorized fissionable material (46.5 MT of U-235 and Pu-239) and fission products (2.5 billion curies) require careful control and operation to prevent their release to the environment.

This study has reviewed the present and planned Mo facilities and identified important accident sequences which contribute to the public risk. The study has focused on identibying the possible problem areas and accident initiators but has not attempted to quantify the probability or consequences. The following is a summary of the findings of the study.

7.1 SUMMARY

The Morris Operation (MO) expansion program will increase the spent fuel storage capacity from 750 metric tons (MTHM) to 1850 MTHM. To complete the expansion, an additional storage pool must be constructed contiguous to the existing storage basins. The addition involves heavy construction work, including blasting. Several supporting systems are being added or expanded to handle the new spent fuel. These include the basin filter system, basin water-cooling system, ventilation system, basin crane and radiation monitoring instrumentation.

Some of the major risks of operation of MO are evaluated in the Consolidated Safety Analysis Report (CSAR). However, in some cases the CSAR assumptions are incomplete and some accident sequences have been omitted entirely. To accurately evaluate the risk to the public the additional accidents must be identified, evaluated and quantified. Listed below are summaries of the important findings of the study.

- A major uncertainty in risk is caused by the proposed construction program which includes blasting in the vicinity of the existing pool without removing the existing spent fuel. The possibility of damage to the basin walls, gates and foundation as well as sabotage must be considered.

- The uncertainty of the fuel storage policy of the U.S. makes the future role of MO uncertain. Additional construction (if it is decided to go beyond the present expansion plan) in the future would be working around an inventory of spent fuel as much as 2½ times the present limit.

- Perhaps the greatest risk of MO is that it will become a de facto permanent waste storage site or a SURF, thus greatly extending the length of time the spent fuel remains on site.

- A criticality accident can be caused by other mechanisms than identified in the CSAR. Major effort has gone into evaluating and preventing a criticality due to a basket tipping into the cask unloading pit. However, there are additional tipping accidents that could occur and potentially result in a criticality accident. These are a cask spill into the unloading pit and basket spill into the basin. In addition, it is possible a missile could impact more than one bundle in the pool causing a larger spill and criticality than analyzed in the CSAR.

- There are no apparent plans for transferring the spent fuel to another site in the event of a major accident which could require emptying the pool and repairing the basin. Under these conditions, delays in implementing a transfer could be detrimental and result in public exposure to radiation.
- The present security precautions of limited access and mechanical sniffing for explosives will be ineffective during the major construction project required to blast, excavate and construct Basin 3.
- The building covering the existing pools (Basin 1 and 2), is a metal-sided building providing only nominal physical security and doubtful environmental protection (it is assumed to be blown away in the tornado analysis). Thus, externalities such as tornados, missiles, and sabotage are more likely to inflict damage on the stored fuel.

- Some accident sequences which have not been evaluated are an airplane crash impacting the pool, a tornado causing evacuation of a large percentage of the pool coolant, and gross corrosion of the pool.
- The MO chlorine concentration limit seems too high to ensure against corrosion of the fuel cladding. Failure of demineralizing systems could lead to excessive levels of chlorine in the pool.
- The LAW vault and sand filters represent the major accumulations of radioactivity (outside of the fuel itself) and will require great care and planning in decommissioning. The present plan for decommissioning by entombment is inadequate if the current trend toward total facility removal is followed. The NRC regulations do not cover Independent Spent Fuel Storage Facilities (ISFSF) but are likely to be revised to do so. The revisions may significantly impact the design and operation of the MO facility.

7.2 RECOMMENDATIONS

As a result of the study, several recommendations have been identified to improve the safety and to reduce the risk of operation of the Morris facility. Listed below are the major recommendations:

1. Develop a contingency plan for removing the fuel and draining the basin in the event of a major problem.

2. Develop a master plan for Morris Operation. Consider one major renovation to its ultimate condition, rather than piecemeal additions which keep the MO in a constant state of change with extra fuel handling and risk associated with each change.
3. Review the plan to construct expansion Basin 3 while fuel is stored in Basin 1 and 2. Evaluate the possibility of removing the spent fuel prior to construction.
4. Evaluate increased security during periods of construction and blasting.
5. Harden the storage building to protect against externalities which could release the 2.5 million curies of fission products authorized to be stored at MO.
6. Add devices to prevent tipping of cask into unloading pit and tipping of baskets into storage basin.
7. Review chloride concentration levels in terms of long-term corrosion impact.
8. Perform an analysis of a tornado evacuating water from the pool.
9. Analyze the event of a missile causing multiple basket tips and bundle spills and possible criticality. (The SER NR-FM-001 Dec. 1975 hypothesizes a one basket criticality.)
10. Revise NRC regulations to cover spent fuel facilities (e.g. require reporting of effluents from spent fuel storage facilities).

11. Complete an analysis of airplane crash effect on the pool and stored fuel.
12. Cover the pool and circulate air under cover rather than expose workers to breathing releases so as to be consistent with ALARA. This will also provide greater security.

APPENDIX A

CROSS REFERENCE - INTERVENTION

CONTENTIONS OF THE STATE OF ILLINOIS

AND RELATED DESCRIPTION IN THE MHB STUDY

The people of the State of Illinois represented by the Illinois Attorney General (IAG) entered the Morris Expansion licensing process with their Petition for Leave to Intervene and Request for Hearing, Docket No. 70-1308, September 16, 1977. In this petition, they address several arguments in support of a stay of the hearings and several technical contentions which may have an important impact on the helath and safety of the public in the vicinity of the Morris Operation (MO).

The following is an abreviated descrition of these arguments and contentions with a cross reference to the sections of the MHB Study, Technical Review of Risk due to Expansion of the Morris Operation Spent Nuclear Fuel Storage, where these subjects are discussed.

IAG ARGUMENTS/ISSUES:	MHB STUDY SECTIONS:
I. Argument for Stay:	
1. No national policy on spent fuel storage	5.1 - 5.5
2. NRC generic EIS incomplete	5.4
3. NRC regulations for spent nuclear fuel are incomplete	5.4
4. ERDA (now DOE) generic EIS on waste management incomplete	not addressed by MHB (hereinafter noted as N/A)

IAG ARGUMENTS/ISSUES:	MHB STUDY SECTIONS:
II. Request for an EIS on MO Expansion:	N/A
III. Contentions:	
A. Underestimated exposure due to:	
1. Future population increases	3.1
2. Storage of mixed oxide fuels	5.2
3. Cladding failures over lifetime	4.1.3
B. Accident analysed inadequate:	
1. Loss of basin cooling	3.2.3, 4.1.5, 4.1.10
2. Cask Accident/liner rupture	4.1.2, 4.1.5
3. Basket drop/criticality	4.1.2, 4.1.4
C. Accident Analyses not considered:	
1. Sabotage	4.1.5, App. B
2. Dresden/MO interaction	N/A
3. Cs Release in transportation	N/A
4. Tornado-related accidents	4.1.4
D. Loss of cladding integrity	4.1.2, 4.1.3, App. C
E. Ground water contamination	3.2.2, 3.2.7, 4.1.5
F. System/component weaknesses:	
1. Storage basins - new & old	3.2.2, 3.2.4, 4.1.1
2. Basin co[illegible]ng - new & old	3.2.3, 4.1.10
3. LAW vault	3.2.7, 4.1.5, 4.1.7
4. Waste lines - basin 3 to LAW vault	4.1.7
G. Security deficiencies in MO and transportation	4.1.1, 4.1.5
H. Emergency plans incomplete	N/A

IAG ARGUMENTS/ISSUES:	MHB STUDY SECTIONS:
I. Construction plan details incomplete:	
1. Site selection & existing basin integrity	3.1, 3.2.4., 4.1.1
2. Construction accidents inadequately evaluated	4.1.1
3. Public exposure due to construction	4.1.1
J. Occupational exposure and genetic defects of public inadequately covered	3.2.5, 4.1.8, 4.1.9
K. Health & safety impact of long term storage	5.3 - 5.5
L. Ultimate responsibility for perpetual care not established	6.0
M. Financial qualifications are not provided	N/A
N. Decommissioning plans inadequate re: transfer of non-decommisionable portions	6.1 - 6.6
O. Financial protection for public liability not defined	N/A
IV. Issues regarding the ER:	
A. ER data incomplete re expansion	N/A
B. ER understates expansion costs:	
1. Doesn't quantify health effects	N/A
2. Costs for economic and health charges not included for some accidents	N/A
3. Health costs for occupancy not included	N/A
4. Decommissioning costs not quantified	6.6
5. Economic cost of short term operation not quantified	N/A
6. Add in real cost of facilities	N/A

IAG ARGUMENTS/ISSUES:	MHB STUDY SECTIONS:
C. ER does not address the comparative cost on MO expansion for:	
1. No reprocessing	N/A
2. Policy favoring permanent disposal of spent fuel	N/A
D. ER has inadequate factual data to support need for expansion	N/A
E. Alternatives to expansion not addressed:	
1. Expansion of existing reactor pools	N/A
2. Dry storage	5.3
3. Reactor pools then dry storage	N/A
4. Reduced nuclear power generation	N/A
5. Storage in existing federal facility	N/A
6. ISFSF at other site	N/A
F. Environmental impact of added transportation due to expansion not covered	N/A

APPENDIX B

TOUR OF MORRIS OPERATION - MAY 11, 1978

OVERVIEW

On May 11, 1978, five people from the Illinois Attorney General's office and two from MHB Technical Associates were provided a tour and explanation of the Morris Operation. The tour lasted approximately four hours and covered the receiving operation, the pool storage, and some of the supporting system.

ATTENDEES:

I.A.G. Personnel	GE/Legal Reprsentatives	MHB
R. Eggerts	G. Voiland	D. Bridenbaugh
D. Hansell	R. Fine	G. Minor
S. Sekuler	G. Engles	
J. Cahan	R. Szwajkowski	
H. Chinn		

PRESENTATION

The initial presentation included films of the Sandia tests of spent fuel (rail and truck) casks. Slides were shown of the Morris facilities and the operation described. Following a brief question and answer period the tour was conducted.

The following sections deal with specific portions of the facility including description of the operation and identification of technical concerns where applicable.

CASK RECEIVING AREA

The cask receiving area is the point where the cask is transfered from the rail car or truck, raised upright, tested for leakage, flushed, and prepared for movement into the receiving basin. The number and types of casks are shown in Table 1. The Morris people stated that a rail cask took about 24 hours per receiving process and a truck cask required about 18 hours.

During the testing phase the cask must be smear tested within 3 hours of arrival to establish the presence and/or magnitude of any leakage. If this test shows too much leakage it must be reported. Next it is vented and the coolant tested for radiation. If it tests too high the cask would not be allowed to be received. However, the Morris people did not know where it would go in this eventuality.

Next the cask coolant is flushed, using basin water. The coolant is flushed to the LAW vault. Radiation monitoring is used to check the coolant for indication of fuel damage. Possible technical problems:

- Rupture of flush line.
- Accidental removal of head or shield water.
- Cask drop on lifting.
- Cask venting due to overpressure during handling.

UNLOADING BASIN

This very deep basin is used for submerging the entire cask while transfering spent fuel bundles to the baskets. Two cranes span this basin, a 125 ton crane for lifting the casks, and a smaller 7.5 ton fuel handling crane mounted under the larger crane. The crane that handles casks is radio-controlled. It has a potential for failure of various relav and interlock failures that could allow it to overtravel its allowable range.

The baskets are lifted or moved with a smaller crane using a rigid extension arm whose physical length is sized to prevent lifting the bundles/baskets out of the water or over another basket. Regular testing of the yokes and extension arms is used as preventive maintenance.

Once the baskets are in the storage pool, they are located on a spacer grid. This grid rides on the pool floor but is braced against the wall for seismic restraint. The baskets have catches at each corner designed to attach it to the grid once it is in place. These are designed to prevent tipping of a free-standing basket under the worst expected seismic event.

The patch where the 1972 cask tip accident in the unloading basin had occurred was still visible. The basket tip preventer (tipping a basket into the deep unloading pit) was described at length. However, nothing was mentioned about prevention of cask tip or basket tip in the basin.

BASIN 1 AND 2

The tour was advised not to spend much time next to Basin 2 because of the high radiation reading caused by the fin-fan cooler on the other side of the wall.

Basin 2 has an expansion gate built in to connect to the new basin once it is constructed. The gate appeared to be of concrete and steel construction. The pool was being tested for temperature rise and had had the cooling turned off for at least several hours. It was still fairly cool but the room was muggy and there were a few wisps of vapor visible on the pool surface, and considerable condensation was dripping from the walls and roof.

CONTROL ROOM

The control room was notable in that it had much instrumentation and controls equipment that was not really used for the fuel storage operation.

SUPPORT SYSTEMS

Because of the rain the tour spent a minimum of time at the LAW vault, sand filter and fin-fan cooler.

ITEMS WARRANTING FURTHER INVESTIGATION

1. Unloading basin and storage basin appear to be vulnerable to tilt and drop accidents. The presence of tilt-preventing device near the passageway to the storage pool serves to prevent the spill of baskets into the pit but not into the storage basin.

2. The basin leak detection system appears incapable of differentiating intrusion from extrusion water.

3. There were no protective devices to prevent foreign material (pens, cameras, tools, bombs, people) from being dropped into the pools.

4. The racking arrangement required moving many baskets to get at any centrally located basket.

5. The passage (gate) from the existing large pool to the expansion area is in place but it was not clear if any interconnecting re-bar, etc., had been implanted to link existing and new structures. It also appears to be a vulnerable place for large volume pool leakage or sabotage during construction.

6. Physical plant arrangement of pool and supportive equipment creates a potential radiation hazard. The radioactively

contaminated fin-fan heat exchanger located close to, but outside, the storage building was readily detectable next to Basin 2.

7. The control room is crowded with controls and indicators pertaining to the canyon (reprocessing area) which are of little or no value to the operation of the spent fuel facility and could cause confusion in an emergency situation.

8. A build-up of radiation levels in the canyon could complicate the maintenance of the few pieces of equipment in the area - mainly the evaporator.

9. The LAW vault decommissioning or emptying process was not discussed and could require transfer of a considerable amount of radioactive material.

10. It appears that there are no plans for unloading the pool in an emergency.

11. The physical security of the building itself is weak. Corrugated building siding is not an effective deterrent to a saboteur or to containing the radiation which could be released by an explosive charge.

One roll of 35mm slides was taken on the tour. The 21 pictures have been duplicated (2 copies each) and mailed for distribution. One set is for Illinois AG files; the other is to be given to MO in accordance with their verbal agreement.

APPENDIX C

BRIEF REVIEW OF CORROSION EFFECTS ON SPENT FUEL

C.1 INTRODUCTION

The following is a review of several recent papers on the subject of spent fuel storage and corrosion problems associated with spent fuel cladding. A summary statement is also provided.

C.2 SUMMARY

Twenty years experience with pool storage of spent fuel has demonstrated its viability as a short-term storage approach. However, the limit of storage time has not been defined. The longest stored fuel includes one low exposure Zircaloy-2 PWR bundle stored for 19 years and 60 stainless steel clad BWR bundles, also of low exposure, stored for approximately 13 years.(1) There have been a number of bundles of failed fuel stored in pools without complications.

There are, however, several identified mechanisms which could affect the fuel cladding integrity and the safety of storage.(2)

1. Time in storage
2. Pool temperature
3. Pool water chemistry
   - Chlorine concentration
   - Boron concentration (reactor pools - PWR)
   - pH control
   - Contaminants due to ion exchange resin degradation

4. Previous history of spent fuel (in reactor)
   - Interaction with fission products (e.g. iodine-induced SCC)[3]
   - High temperature operation
5. Galvanic and crevice corrosion
6. Stress corrosion cracking (SCC) caused by chloride in in coolant.
7. Cladding defects.

Most authors agree that the corrosion problems should not be a serious concern for pool storage of spent fuel over period of time in excess of 20 years storage,[1][2][4][5] provided the proper water quality and handling procedure are maintained. However, there is no clear evidence of an allowable upper limit of storage time and there is a need for continued monitoring and mechanical testing of stored spent fuel.

The potential requirement to minimize spent fuel storage and get the maximum practical energy from each bundle of fuel (due to a lack of reprocessing and unresolved waste disposal plans) could result in fuel having longer reactor residence time and higher exposure. This could also mean a longer exposure to corrosion, stress and temperature in the reactors and further justify long-range monitoring and testing of the spent fuel for possible slow degradation processes.

C.3 A. B. JOHNSON, UTILITY SPENT FUEL STORAGE EXPERIENCE, APRIL 1978

Johnson cites experience with pool storage over the last 19 years as the evidence that no long term corrosion problems will occur. The longest stored fuel includes 19 years on one low exposure PWR, Zircalloy-2 bundle and 12 years on 60 low exposure BWR, stainless clad bundles.

Visual inspection is the most common technique for evaluation of corrosion but there are also a few examples of metallurgical examination and a few cases where Canadian spent fuel performed satisfactorily when returned to a reactor after several years of storage (5-10 years).

Substantial corrosion has been experienced during pool storage of elements from plutonium production reactors and stainless clad fuel from gas-cooled reactors. Johnson believes these are sufficiently different in material and exposure history to rule out similar failures in LWR fuel.

The evidence cited covers at most two decades of experience. It can be extrapolated for a reasonable period, but as of now, no upper limit can be set for pool storage time.

The mechanisms for degradation of fuel bundle integrity identified in the literature include stress corrosion, galvanic action, hydriding, and handling accidents.

Johnson concludes that "the corrosion assessment leads to the conclusion that fuel bundle materials are corrosion-resistant and the pool storage environments are relatively benign. While some slow degradation cannot be ruled out, it appears to be unlikely."

There have been 3-4 fuel handling incidents per year (1974-76) but only two resulted in damage sufficient to cause gas release.

Future evaluation of spent fuel degradation effects should include both visual and mechanical inspection which can be linked back to earlier tests and correlated with fuel bundle history.

C.4 Z. A. Munir, AN ASSESSMENT OF THE LONG-TERM STORAGE OF ZIRCALLOY FUEL RODS IN WATER (UNDATED)

Munir's literature search disclosed that Zircalloy degradation during pool storage is a function of the following variables:

1. Pool temperature
2. Time
3. Water purity
   Cl concentration
   Boron concentration
   Exchange resin degradation products
4. Previous history
   Interaction with fission products at high temperature
   Temperature of zirc clad during operation
5. Galvanic corrosion and crevice corrosion
6. High stress and chloride concentration
7. Clad defects

Overall he believes that these problems are not of concern for storage of spent fuel in pools over extended periods of time (>10-20 years) but he does not cite an upper limit for time in pool storage. Munir also calls for continued testing and monitoring to catch any longer-term problems.

C.5 S. A. Mayman, CANADIAN EXPERIENCE WITH WET AND DRY STORAGE CONCEPTS, April 1978; presented to ANS.

Canada has planned to store its spent fuel for a longer period than called for in the original U.S. plan. Therefore they built in 5-10 years storage at CANDU reactors and have plans for extensive AFR construction. Spent fuel from CANDU reactors has very low concentrations of fissionable material and also a low heat generation rate (a 7500 MWD/MTHM bundle produces only 2 kw after one day cooling). This permits rather simple storage systems without concern for criticality and only nominal cooling requirements.

Present trends in Canada are toward air storage in a forest of concrete canisters, each about 16 feet tall and containing about 4½ tons of irradiated fuel. This is projected to require fuel management costs of only $6-7/kg.

C.6 J. R. Weeks, CORROSION OF MATERIALS IN SPENT FUEL STORAGE POOLS, July 1977.

Weeks discusses the corrosion resistant nature of the environment of spent fuel including the water chemistry of BWR, PWR and AFR pools.

He discusses some experience with stress corrosion cracking of materials under high temperature, acidic environment, and long-term exposure. Although concluding stress corrosion is unlikely under fuel pool conditions, Weeks acknowledges stress corrosion cannot be entirely ruled out.

Experience with galvanic corrosion has been very satisfactory but Weeks cautions against aluminum and stainless steel couples.

He cites the passivating oxide films on the materials in the pool, the water chemistry restriction on chlorine concentration (<1 ppm), low pool temperature and pH control as reasons for believing corrosion in fuel storage pools "should be negligible for periods upwards of 20 years."

C.7 J.T.A. Roberts Paper on PCI Failure Model from EPRI Journal, June 1978.

One of the main performance problems of Zircalloy fuel has been pellet clad interaction (PCI). Roberts claims 3-6% loss of capacity factor in BWR's and 1-2% in PWR's can be attributed to the operationg limits imposed by the vendors in order to prevent PCI.

A conflicting factor is the desire of utilities to run fuel to high burn-up rates and, thus, longer radiation exposures of the zirconium cladding. The more highly irradiated fuel shows decreasing resistance to iodine-caused stress corrosion cracking (SCC).

An effort is being made to identify specific improvements in fuel design to reduce PCI, SSC and corrosion. In the meantime, the implication for spent fuel pools is an increase in the quantity of non-improved high exposure fuel which may actually be weakened by the reactor environment it has experienced.

REFERENCE

1. A. B. Johnson Utility Spent Fuel Storage Experience, April 1978.
2. Z. A. Munir, An Assessment of the Long-Term Storage of Zircalloy Fuel Rods in Water. Final Report, 1978.
3. EPRI Journal, J.T.A. Roberts, Paper on PCI Failure Model, June 1978.
4. J. R. Weeks, Corrosion of Materials in Spent Fuel Pools, July 1977.
5. S.A. Mayman, Canadian Experience with Wet and Dry Storage Concepts, April 1978.

APPENDIX D

SPENT FUEL STUDY

BIBLIOGRAPHY - REPORTS

SPENT FUEL STUDY

BIBLIOGRAPHY - REPORTS

1. ERDA-76-43. Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle, Volumes 1 - 5, May 1976, Battelle Northwest for ERDA.

This is an extensive report started in 1975 and completed in 1976, which covers four major waste management functions: treatment, interim storage, transportation, and final storage or disposal. Section 17 of Volume 3 contains extensive information on the interim storage of spent fuel elements.

MHB office copy + an additional copy of the summary and parts of Volume 3.

2. JPL Publication 77-69. An Analysis of the Technical Status of High Level Radioactive Waste and Spent Fuel Management Systems, December 1977, JPL for Calif. Energy Commission.

A good summary of the "state of the art" with an extensive section (with pictures) on spent fuel storage. Lots of numbers and quotes from the MFRP Consolidated Safety Analysis (NEDO-21326-11).

MHB office copy.

3. SAND-77-1816. Unlimited Release, Estimates of Relative Areas for the Disposal In Bedded Salt of LWR Wastes From Alternative Fuel Cycles, January 1978, Sandia, Lincoln-Larsen and Sisson for U.S. NRC and DOE.

Sandia report which discusses repository land areas expected to be required for various alternatives. Prepared in conjunction with the S-3 Hearings.

MHB office copy received from NRDC.

4. SAND-77-0274. WIPP Conceptual Design Report, Parts I, II, III, June 1977, Sandia for ERDA.

Detailed conceptual design with drawings of the demo salt bed disposal pilot plant in New Mexico.

Sent to MHB by Sandia at request of DOE.

5. BNWL-2256. Behavior of Spent Nuclear Fuel in Water Pool Storage. September 1977, by A.B. Johnson, Jr. for ERDA.

This report summarizes the current experience of irradiated fuel in water pool storage, discusses corrosion rates and mechanisms and factors affecting extended storage times.

Copy sent directly to MHB by DOE at request of NRDC.

6. ARH-2888REV. Retrievable Surface Storage Facility Alternative Concepts Engineering Studies, July 1974, by Atlantic Richfield Hanford Company and Kaiser Engineers prepared for AEC.

This report summarizes several alternatives considered by the AEC for retrievable surface storage. Considered in this report are both water and air-cooled storage concepts. Contains design descriptions and capital cost estimates.

Copy was transmitted direct to MHB by DOE at NRDC request.

7. Spent Fuel Storage Study 1976-1986. April 1977, Subcommittee on Spent Fuel Storage, Atomic Industrial Forum.

Summarizes the industry position on the problems of existing spent fuel storage capacity. Contains good projections of existing capacity and fuel production plus information on available shipping casks.

MHB copy in office.

8. JPL 77-59. An Analysis of the Back End of the Nuclear Fuel Cycle With Emphasis on High-Level Waste Management, Volumes I and II, August 12, 1977. Jet Propulsion Laboratory prepared for the Office of Science and Technology Policy.

Documents a study performed by a group from JPL and other organizations looking at high-level nuclear management. Identifies current programs and plans, implications of schedules, and lists missing elements in waste disposal plans. Emphasis is on policy and decision making rather than technology.

2 copies in MHB office.

9. Status of Nuclear Fuel Reprocessing Spent Fuel Storage and High-Level Waste Disposal, Overview and Summary and Draft Report, January 11, 1978, California Energy Commission.

Summarizes Energy Commission's one and a half-year review of fuel reprocessing and high-level waste in response to the California Legislature Bills. A good summary of the "state of the art."

MHB office copy.

10. NUREG-0116. Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle, October 1976, U.S.NRC.

This is a supplement to WASH-1248, Environmental Survey of Uranium Fuel Cycle. WASH-1248 purpose was to establish a technical basis for consideration of the environmental effects of uranium fuel cycle for environmental impact statements for individual LWRs. This supplement was prepared after the NRDC DC appeals court decision to establish a basis for identifying environmental impacts associated with fuel reprocessing and waste management activities. It provides background for the current S-3 Table.

MHB office copy.

11. NUREG-0216. Public Comments and Task Force Responses Regarding the Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle, March 1977, U.S.NRC.

Contains written comments received by the NRC on Report NUREG- 0116 above.

MHB office copy.

12. Spent Fuel Storage, A Review of the Technology of the Demonstration of Feasibility of Storing Unreprocessed Spent Fuel for Extended Time Periods, July 1977, by MHB for the California Energy Commission.

Summarizes spent fuel storage testimony presented in the California Energy Commission hearings in March 1977.

MHB office copy.

13. NUREG-0278. Technology, Safety, and Costs of Decommissioning a Reference Nuclear Fuel Reprocessing Plant, Volumes 1 & 2, October 1977. Battelle Northwest for U.S.NRC.

Describes decommissioning alternatives and evaluates the safety and costs associated with the alternatives for a reference fuel reprocessing plant. The reference plant is one with characteristics similar to the Barnwell Nuclear Plant.

MHB office copy.

14. Midwest Fuel Recovery Plant Technical Study Report, July 1974, General Electric Company.

Documents the results of the Reed Review of MFRP and reasons for not placing MFRP in operation. Recommends a new process flow sheet and plant configuration if plant is to be placed in operation.

MHB office copy.

15. ERDA 76-25. 1976-1985 LWR Spent Fuel Disposition Capabilities, 1976 Edition, May 1976, prepared by ERDA.

Provides a listing of spent fuel production and storage capabilities.

MHB office copy.

16. WASH-1503. Environmental Statement Radioactive Waste Repository Lyons, Kansas, June 1971, U.S.AEC.

Summarizes the initial measures taken as a part of the AEC's waste management policy and program for the permanent disposal of wastes. An early environmental impact report. Has interesting appendices with Congressional and other letters.

Loaned copy from NRDC.

17. NR-CONF-001. Proceedings of Nuclear Regulatory Commission Workshop on the Management of Radioactive Waste: Waste Partitioning as an Alternative, June 1976, Battelle Seattle Research Center for NRC.

Summarizes papers and discussion at a 3-day conference in 1976 discussing many aspects of the closure of the nuclear cycle. Interesting but not particularly authoritative.

MHB office copy.

18. Nuclear Technology: Waste Management Symposium, December 1974, Volume 24, No.3.

Contains approximately 20 papers selected from those presented at the Waste Management 1974 Symposium held in Tucson, Arizona, April 1974. Of particular interest, transportation of nuclear fuel and waste, geohydrologic considerations in the management of radioactive waste, and retrievable surface storage facility for commercial high-level waste.

MHB office copy.

19. ERDA 33. Nuclear Fuel Cycle: A Report by the Fuel Cycle Task Force, March 1975, ERDA.

A much-quoted report discussing the remaining questions in closing the fuel cycle. States the problems are political and societal rather than technical.

MHB office copy.

20. NUREG-0043. Alternative Processes for Managing Commerical High-Level Radioactive Wastes, April 1976, Battelle Northwest for ERDA.

Discusses a number of alternatives for managing high-level radioactive waste presently stored at West Valley (NFS). Basically applies ERDA 76-43 alternatives to West Valley.

MHB office copy.

21. Progress and Problems in Programs for Managing High-Level Radioactive Wastes, B-164052, January 1971, GAO report for the JCAE.

A historical report calling AEC decision in 1970 to develop salt mines for potential use as a federal repository a major milestone. Not much has changed.

MHB office copy.

22. Storage and Disposal of Radioactive Waste, Hearing Before the JCAE, November 19, 1975. Government Printing Office.

Alternatives, quantities, and more of the same.

MHB office copy.

23. BNWL-1940. A Program Plan for Comprehensive Characterization of Solidified High-Level Wastes, December 1975, Battelle Northwest for ERDA.

Description of a program to quantitatively measure the properties of various solidified waste forms, how they are affected by processing parameters and by thermal and radiation effects during storage and disposal.

MHB office copy.

24. Improvements Needed in the Land Disposal of Radioactive Wastes - A Problem of Centuries, January 1976, B-164105.

A GAO report discussing the problem of the other than high-level radioactive waste volumes and the unknowns associated with them. Recommends a comprehensive study of existing disposal sites and development of site selection criteria.

MHB office copy.

25. Nuclear Waste Disposal and Transportation, November 3, 1975, Prepared by Assembly Committee on Resources, Land Use and Energy Staff (Varanini, Simon, Praul).

An independent "state of the art" summary of waste disposal.

MHB office copy.

26. High-Level Radioactive Waste Management: Past Experience, Future Risks, and Present Decisions, April 1975, SAI prepared for Resources and Environment Division of the Ford Foundation.

Another broad summary of the same problem.

MHB office copy.

27. Memorandum of Points and Authorities in Support of Nuclear Regulatory Commission Licensing of the ERDA High-Level Waste Storage Facilities Under the Energy Reorganization Act of 1974, July 1975 by NRDC, Cotton and Lash.

Memo prepared to justify the need to license ERDA's proposed Hanford facilities.

MHB office copy.

28. BNWL-SA-5231 REV 1 A Rev w of High-Level Radioactive Waste Disposal Alternatives, Febr ary 1975 by Battelle, Gary Dau.

A rather broad and shallow look at most of the waste disposal alternatives, including extra-terrestrial disposal etc.

MHB office copy.

29. ORNL-TM-4481. Geochemical Behavior of Long-Lived Radioactive Wastes, July 1975, by Ferrucio Gerra, Oak Ridge National Laboratory.

A rather heavy report on the geochemical behavior of radioactive wastes. Short on conclusions.

MHB office copy.

30. Nuclear Fuel Reprocessing and High-Level Waste Disposal, July 1977, Interim California Energy Commission Report. Precursor to January 11, 1978 Energy Commission Report.

MHB office copy.

31. EPA 520/4-76-016. 40 CFR 190 Environmental Radiation Protection Requirements for Normal Operations of Activities in Uranium Fuel Cycle, final environmental statement, Volumes 1 & 2, November 1976, U.S. EPA.

Establishes radiation standards for normal operations of the uranium fuel cycle to assure protection for members of the public against radiation doses resulting from fuel cycle operations and to limit the environmental burden of long-lived radioactive materials. Documents the environmental affects of these standards.

MHB office copy.

32. NUREG-0217. NRC Task Force Report on Review of the Federal/State Program for Regulation of Commercial Low-Level Radioactive Waste Burial Grounds, March 1977, NRC.

One of the first looks at the overall implications of low-level waste disposal. Recommends accelleration of the development of NRC regulatory program.

MHB office copy.

33. NUREG-0002,ES. Final Generic Environmental Statement of Use of Recycled Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors (Executive Summary), August 1976, U.S.NRC.

A much shortened executive summary of the GESMO documentation. Since this has been terminated, it is academic but provides background.

MHB office copy.

34. Nuclear Plants, The More They Build, The More You Pay, 1976, by Ron Lahoue, Center for Study of Responsive Law.

An environmentalist's look at nuclear economics. A critical review of FPC and utility finances. Do-it-yourself economic evaluation techniques.

MHB office copy.

35. ERDA-1553-D. Draft Environmental Statement, Management of Intermediate Level Radioactive Waste, January 1977, Oak Ridge National Laboratory for ERDA.

A draft EIS prepared by ORNL to cover selection of a technique for the management of intermediate-level radioactive liquid waste at Oak Ridge.

MHB office copy.

36. The Nuclear Fuel Cycle: A Survey of the Public Health Environmental and National Security Effects of Nuclear Power, 1974, by Dan Ford, et al, UCS.

One of the first critical looks at high-level waste disposal alternatives and risks.

MHB office copy.

37. ERDA-77-25, 1977-1986 LWR Spent Fuel Disposition Capabilities, 1977 Edition, ERDA.

A summary of nuclear fuel storage capacity (update of ERDA-76-25).

Received from DOE under FOIA request.

38. U.S. and Non-U.S. Lightwater Reactor Spent Fuel Storage, July 1977, Nuclear Assurance Corp. for ERDA.

A detailed summary by reactor of spent fuel discharged and storage capacity.

Received from DOE for NRDC FOIA request.

39. NRDC FOIA Documents, January 1978.

A listing of 27 letters and documents received from DOE in response to NRDC FOIA request. See enclosure A listing.

Received from NRDC.

40. An Assessment of the Long-Term Storage of Zircaloy Fuel Rods in Water, Final Report, Z. A. Munir, U.C. - Davis.

Literature search done by UC Prof. on long term storage performance to be expected of Zircalloy. Funded by Cal. Energy Commission.

MHB office copy provided by CEC.

41. NUREG-0404. Draft GEIS on Handling and Storage of Spent Light Water Power Reactor Fuel, Volumes 1 & 2, March 1978, NRC.

NRC's environmental statement prepared as noticed in the Federal Register to justify extended interim storage and on-site compact storage. Finds no significant additional impact. Compares primarily to added coal fired production.

MHB has loaned (by NRC) copy & has ordered one.

42. GAO Report - Nuclear Energy's Dilemma: Disposing of Hazardous Radioactive Waste Safely, EMD-77-41, September 9, 1977.

GAO's most recent status report. Finds: public/political opposition, gaps in laws, geological uncertainties, lack of criteria, etc. Recommends improved program (again).

MHB copy.

43. RHO-LD-77-4 SEP. Spent Unreprocessed Fuel Facility Monthly Progress Report, September 1977, Rockwell International.

One of a series of reports by Rockwell Hanford documenting work performed on the SURFF (spent unreprocessed fuel facility) study.

Received from NRDC from FOIA.

44. RHO-LD-77-4 OCT. Spent Unreprocessed Fuel Facility Monthly Progress Report, October 1977, Rockwell International.

See number 43 above.

45. RHO-LD-77-4 NOV. Spent Unreprocessed Fuel Facility Monthly Progress Report, November 1977, Rockwell International.

See number 43 above.

46. RHO-CD-136 Draft. Spent Unreprocessed Fuel Facility Program Plan. October 1977. Rockwell International.

See number 43 above.

47. RHO-LD-2 Informal Report. Spent Unreprocessed Fuel Facility Engineering Studies, February 1978. Rockwell International.

See number 43 above.

48. NUREG-0300, Proposed Goals for Radioactive Waste Management, May 1978. Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission.

49. A Review of the KBS Reports on Spent Nuclear Fuel Handling and High Level Waste Storage, June 1978, Nuclear Fuel Cycle Committee, California Energy Commission.

50. Reviews of Modern Physics, Vol. 50 Number 1 Part II, January 1978. Report to the APS by the Study Group on Nuclear Fuel Cycles and Waste Management. American Institute of Physics.

51. Staff Testimony on Economic Data to Support the Feasibility of the S-3 Model, Docket No. RM-50-3, U.S.A. Nuclear Regulatory Commission.

52. Economic Impacts of the Total Nuclear Waste Management Program Envisioned for the United States, L. Busch and A.J. Zielen, Argonne National Laboratory and S.J.S. Parry, U.S. Nuclear Regulatory Commission.

53. DOE/ER-0004/D, Findings of the "Deutch" Task Force on Nuclear Waste Management. February 1978. MHB Office Copy.

APPENDIX D

2/21/79
CCH

STATUS OF EXISTING LICENSE

Renewal Application to be submitted prior to March 1 in form of updated CSAR and updated Operating Experience Report.

Contingency Plan	- in review
Physical Security Plan	- OK
Security Qual. & Training	- in review
Decommissioning Plan	- in review
QA Plan	- OK
Emergency Plan	- OK

Features:

CSAR

Operating Experience Report

Operation Specifications - in review

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DEGREE OF COMPLIANCE WITH PART 72

Subpart E Siting Criteria (No Specific Criteria Given)

72.61 General a thru f: comply

72.62 Criteria for Design Basis External Natural Events a thru c: comply

72.63 Criteria for Design Basis external man-induced events a thru c: comply

72.64 Criteria for defining potential effects of the ISFSI on the region

a thru c: comply

72.65 Criteria for regional distribution of population

a: projections of future land and water uses incomplete

b thru f: comply

72.66 Criteria for defining acceptable seismic characteristics

a: site specific "g" value used - comply

b: comply

c: NA

72.67 Criteria for defining potential radiological consequences

a thru b: comply

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SUBPART F - GENERAL DESIGN CRITERIA

72.71 General Design Criteria

Overall Requirements

1 - Quality Standards: comply

2 - Protection Against Environmental Conditions and Natural Phenomena

i: comply

ii: site-specific value used-sec 72.66(a) - for seismic;other : comply

√ iii: Not in compliance (instrumentation)

iv: Comply or NA

3 - Protection Against Fires and Explosions: comply

4 - Sharing of structures, etc: NA

5 - Proximity of sites: comply

6 - Testing and Maintenance: comply

7 - Emergency Capability: comply

8 - Confinement Barriers and Systems

i: comply

ii: comply

iii: comply

9 - Instrumentation and Control System : comply

10 - Control Room or Control Areas : comply

11 - Utility Services

i: comply

ii: comply

iii: comply

<u>Nuclear Criticality Safety</u>

12 - Design for Criticality Safety : comply

13 - Acceptable Methods of Control

i: comply

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Radiological Protection

14 - Exposure Control: comply

i: comply

ii: comply

iii: comply

iv: comply

v: comply

15 - Radiation Alarm System: comply

16 - Effluent Monitoring: no means for measuring flows of air

No Kr^{85} routine measurement

17 - Effluent Control: Comply

Spent Fuel and Radioactive Waste Storage and Handling

18 - Spent fuel and radioactive waste storage and handling systems: comply

i: comply

ii: comply

iii: comply

iv: comply

v: Marginal compliance (object to requirement)

19 - Waste Treatment: no proven method for the LAW vault material disposition

Decommissioning

20 - Decommissioning: marginal compliance

Subpart G - Quality Assurance

72.75 Quality Assurance program; Records

a: comply

b: NA

c: comply

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Subpart H - Plant Protection

72.81 Physical Security Plan

a thru c: comply

Subpart I - Training and Certification of ISFSI Personnel

72.91 Scope of Training Program: comply (no identification of safety related manipulations and controls have been made)

72.92 Responsibility for Training Program: OK

72.93 Physical Requirements: comply

72.94 NA

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Exceptions to Compliance with Regulatory Guide 3.44

2.1.3 Population Distribution and Trends

Calls for 4 decades

2.6.2.5 Design Earthquake

The "present staff position" will likely be modified as a result of comments on the proposed rule (10CFR72). MO is designed to 0.2g, not 0.25g.

7.2 Radiation Sources

The MO CSAR includes only irradiated fuel and contaminated basin water as radiation sources. Other tanks and pipes should be included.

10. Operating Controls and Limits

Such controls and limits have been submitted to amend the MO License.

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Proposed Strategies

A. ¶72.36(a) Transfer License*

(b)(1) Application shall include: identification, financial, and technical qualifications as for a new application

... And any additional information requested, e.g. ...

(2) Radiation Protection information and qualification

(3) Consent of existing licensee

(c) Interested persons will be notified and hearings <u>may</u> result

Transfer will be approved if it is determined:

(1) The transferee is qualified

(2) The transferee is within laws, regulations and commission orders

* Also covered in:
¶70.36 inalienability of licenses

B. New License

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§72.14 Contents of Application

(a) General I.D.

(e) Financial

72.5 Technical Info

(a) SAR

(c) QA Plan

(d) Physical Security Plan

(e) Prop Testing

(f) Decommissioning Plan

72.19 Emergency Plan

72.20 Environmental Report * * *

APPENDIX E

§ 72.18 Decommissioning plan, including financing.

(a) Each application under this part shall include a proposed decommissioning plan that contains sufficient information on proposed practices and procedures for the decontamination of the site and facilities and for disposal of residual radioactive materials after all spent fuel has been removed, in order to provide reasonable assurance that the decontamination and decommissioning of the ISFSI at the end of its useful life will provide adequate protection to the health and safety of the public. This plan shall identify and discuss those design features of the ISFSI that facilitate its decontamination and decommissioning at the end of its useful life.

(b) The decommissioning plan shall include the financial arrangements made by the applicant to provide reasonable assurance that the planned decontamination and decommissioning of the ISFSI will be carried out.

APPENDIX F

GENERAL ELECTRIC

NUCLEAR FUEL AND SERVICES DIVISION

GENERAL ELECTRIC COMPANY, 175 CURTNER AVE., SAN JOSE, CALIFORNIA 95125

Mail Code 861

SPENT FUEL SERVICES OPERATION

DMD-547

Docket No. 72-1
Docket No. 70-1308
License No. SNM-1265

May 15, 1981

Office of Nuclear Material Safety & Safeguards
Attn: R E. Cunningham, Director
Division of Fuel Cycle & Material Safety
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: RESPONSE TO NRC REQUEST FOR INFORMATION re EMERGENCY TRAINING

Gentlemen:

On May 8, 1981, Dr. Tom Clark of your staff called to request that we provide a discussion of emergency training plans and activities carried on at Morris Operation as reflected in the *Radiological Emergency Plan for Morris Operation*, NEDO-21894, in relation to the content of 10CFR50, Appendix E, Section IV-F, "Training."* The following discussion is in response to his request:

Emergency Consequences and Response Required

In general, Appendix E is concerned with an emergency at a reactor requiring the involvement of many segments of an emergency structure, including public agencies at Federal, State and local level who might be involved in evacuation or other action within the Emergency Protection Zone (EPZ). This concern is in stark contrast to the emergency spectrum at Morris Operation (REP 4.2) where there is no off-site impact for any credible accident or other emergency. Although emergencies at Morris Operation would not require the complex response nor the large emergency force contemplated by Appendix E, each element of "Contents of Emergency Plan" of Appendix E is addressed in General Electric's license application as required by 10CFR72.19.

* References to emergency plan sections are noted "REP X.X.X".

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There are two elements that constitute the principal basis for implementing radiological emergency training at Morris Operation. These elements are the limited nature of consequences from credible emergencies (REP 4.2 through 4.2.8) and the small staff required to safely and efficiently operate an ISFSI (REP 5.2, 5.2.3 and 5.2.4). The nature of the consequences from credible accidents or other emergencies at Morris Operation limits the scope of specialized emergency training required for operating and management personnel as well as for off-site support personnel. The small staff requirements of Morris Operation precludes the use of specialized emergency teams, as listed in Appendix E,and the limited consequences make such specialization unnecessary.

Specialized Training Activity and Categories of Emergency Personnel

All essential functions of the categories of emergency personnel contained in Appendix E, IV-F are performed at Morris Operation.*

The Emergency Brigade (REP 5.2.3) performs the functions of radiological monitoring (c.), fire fighting (d), damage control and repair (e.), first aid and rescue teams (f). Emergency Brigade training is an integral part of operator training as described in Attachment F to the applicant's amended application for license renewal under 10CFR72 dated January 12, 1981. This training is on-going and includes drills and exercises as well as classroom work.

Personnel responsible for emergency assessment (b.) include shift supervisors (REP 6.2) and the Emergency Task Force (REP Chapter 2 and §5.2.4). Shift supervisors are provided with special instructions regarding emergencies. These instructions, located in the Control Room, include lists of telephone numbers, copies of emergency plans, copies of Morris Operating Instructions, etc. Shift supervisors are fully qualified as operators and are the most experienced of operations personnel. They are well qualified to undertake the Emergency Coordinator responsibilities in an emergency (a.).

The Emergency Coordinator and Emergency Brigade are supported by the Emergency Task Force. Members of the task force are specialists or managers of specific activities and their emergency duties parallel their normal duties. For example, the Senior Engineer - Licensing & Radiological Safety (REP 5.2.4.6) provides radiological expertise and analyses

* In the following discussion each category of emergency personnel contained in IV-F are identified by letter reference to IV-F, notated (a.), (b.), etc.

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during an emergency.

The Manager - Morris Operation may act as Emergency Coordinator (REP 5.2.1) or delegate these duties to another. In either case, he is advised by the Emergency Task Force and directs overall operation of the facility during an emergency. He participates in emergency drills and exercises at Morris Operation. He is actively involved in on-going coordination with the local Emergency Services and Disaster Agency, local law enforcement agencies and other aspects of emergency planning, as well as having attended classes in radiation safety and other emergency-oriented training.

Medical support personnel and security personnel have received specialized radiological training and other training as described in our response to questions dated March 18, 1981; see response to question 6. Support personnel from the Division's headquarters in San Jose would be working within their speciality and do not require special training or instruction other than a situation briefing upon arrival at the site.

In summary, the initial training required for emergency response at Morris Operation is covered by existing training programs and is integrated in the training and certification program required by 10CFR72, including periodic retraining requirements.

Training Available to Local Service Personnel

Training is offered and provided to local service and law enforcement personnel as noted in our response to questions dated March 18, 1981; see response to question 6. Local news media personnel have frequently visited the site and such visits are encouraged.

Exercise and Drills

The "full scale" exercises discussed in Appendix E, IV-F, 1, 2 and 3 are applicable to nuclear power plants. We know of no reason to apply these full scale exercise requirements to an ISFSI. The very limited consequences and the small controlled area [illegible] make it unnecessary to do so. However, liaison is maintained among local and State emergency agencies.

Adequate emergency exercises and drills are conducted as discussed in REP 8.1. Communication links are tested daily.

Maintaining Emergency Preparedness and Recovery

The plan for maintaining emergency preparedness is described in REP-Chapter 8. A recovery plan is in effect (REP-Appendix 3).

Distances to Specific Support Services

Approximate mileage between Morris Operation and off-site support services are as follows:*

Glenwood Medical Group	... 20 miles
St. Joseph's Hospital	... 20 miles
University of Chicago Hospital	... 55 miles
Coal City (fire and rescue)	... 8 miles
Murray & Trettel	... 60 miles

Please call H. Rogers (408*925-6496) or C. Herrington (408*925-6385) of this office if there are questions regarding this response or other aspects of emergency planning at Morris Operation.

Respectfully,

GENERAL ELECTRIC COMPANY

D M Dawson

D.M. Dawson, Manager
Licensing & Transportation

DMD:HAR:bn

* Per phone conversation between Dr. K.J. Eger (GE) and Dr. A.T. Clark (NRC) May 12, 1981.

NOTICE OF DISTRIBUTION

to

SERVICE LIST - DOCKET NO. 70-1308 & 72-1

In the matter of General Electric's application for renewal of Materials License No. SNM-1265, copies of the documents discussed in the attached letter have been forwarded to the law firm of Mayer, Brown and Platt, 231 South LaSalle, Chicago, IL. 60604, counsel for General Electric Company, for transmittal to the service list as shown below:

Andrew C. Goodhope, Esq., Chairman
Atomic Safety and Licensing Board
3320 Estelle Terrace
Wheaton, Maryland 20906

Dr. Linda W. Little
Atomic Safety and Licensing Board
5000 Hermitage Drive
Raleigh, North Carolina 27612

Dr. Forrest J. Remick
Atomic Safety and Licensing Board
305 East Hamilton Avenue
State College, Pennsylvania 16801

Atomic Safety and Licensing Appeal Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Bridget Little Rorem
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Susan N. Sekuler, Esq.
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Atomic Safety and Licensing
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Docketing and Service Section
Office of the Secretary
U.S. Nuclear Regulatory Commission
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List 3/28/80 A

APPENDIX G

§ 72.35 Changes, tests and experiments.

(a)(1) The holder of a license issued under this Part may, without prior Commission approval unless the proposed change, test or experiment involves a change in the license conditions incorporated in the license, an unreviewed safety question, significant increase in occupational exposure or a significant unreviewed environmental impact: (i) make changes in the ISFSI described in the Safety Analysis Report, (ii) make changes in the procedures described in the Safety Analysis Report, or (iii) conduct tests or experiments not described in the Safety Analysis Report.

(2) A proposed change, test, or experiment shall be deemed to involve an unreviewed safety question (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report may be increased; (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

(b)(1) The licensee shall maintain records of changes in the ISFSI and of changes in procedures made pursuant to this section if such changes constitute changes in the ISFSI or procedures described in the Safety Analysis Report. The licensee shall also maintain records of tests and experiments carried out pursuant to paragraph (a) of this section. These records shall include a written safety evaluation that provides the bases for the determination that the change, test, or experiment does not involve an unreviewed safety question. The records of changes in the ISFSI and of changes in procedures and records of tests shall be maintained for the lifetime of the ISFSI.

(2) Annually, or at such shorter interval as may be specified in the license, the licensee shall furnish to the appropriate regional office, specified in Appendix D of Part 20 of this chapter, with a copy to the Director, Office of Nuclear Material and Safeguards, a report containing a brief description of such changes, tests, and experiments, including a summary of the safety evaluation of each. Any report submitted by a licensee pursuant to this paragraph will be made a part of the public record pertaining to this license.

(c) The holder of a license issued under this Part who desires (1) to change the license conditions, (2) to change the ISFSI or the procedures described in the Safety Analysis Report, or (3) to conduct tests or experiments not described in the Safety Analysis Report that involve an unreviewed safety question, a significant increase in occupational exposure, or significant unreviewed environmental impact, shall submit an application for amendment of the license, pursuant to § 72.29 of this Part.