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Control of Heavy Loads at Nuclear Power Plants

Resolution of Generic Technical Activity A-36

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Office of Nuclear Reactor Regulation

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This report summarizes work performed by the NRC staff in the resolution of Generic Technical Activity A-36, "Control of Heavy Loads Near Spent Fuel". Generic Technical Activity A-36 is one of the generic technical subjects designated as "unresolved safety issues" pursuant to Section 210 of the Energy Reorganization Act of 1974.

The report describes the technical studies and evaluations performed by the NRC staff, the staff's guidelines based on these studies, and the staff's plans for implementation of its technical guidelines.

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<u>ABSTRACT</u>

In nuclear power plants heavy loads may be handled in several plant areas. If these loads were to drop in certain locations in the plant, they may impact spent fuel, fuel in the core, or equipment that may be required to achieve safe shutdown and continue decay heat removal. If sufficient spent fuel or fuel in the core were damaged and if the fuel is highly radioactive due to its irradiation history, the potential releases of radioactive material could result in offsite doses that exceed 10 CFR Part 100 limits. If the load damaged equipment associated with redundant safe shutdown paths, the capability to achieve safe shutdown may be defeated. Additionally, if fuel is of sufficient enrichment, the normal boron concentrations that are maintained may not be sufficient to prevent a load drop from causing the fuel configuration to be crushed and result in criticality.

Task A-36 was established to systematically examine staff licensing criteria and the adequacy of measures in effect at operating plants, and to recommend necessary changes to assure the safe handling of heavy loads. The task involved review of licensee information, evaluation of historical data, performance of accident analyses and criticality calculations, development of guidelines for operating plants, and review of licensing criteria. This report provides the results of the NRC staff's review of the handling of heavy loads and includes the NRC staff's recommendations on actions that should be taken to assure safe handling of heavy loads. These recommendations include: (1) a program should be initiated to review operating plants against the guidelines developed in Task A-36; (2) certain interim measures should be taken for operating plants until completion of this review program; (3) changes to certain Standard Review Plans and Regulatory Guides should be made to incorporate the guidelines in this report; (4) changes to technical specifications should be made after completion of the review; and (a task should be initated to establish guidelines for the control of small loads near spent fuel. The guidelines proposed include definition of safe load paths, use of load handling procedures, training of crane operators, guidelines on slings and special lifting devices, periodic inspection and maintenance for the crane, as well as various alternatives that include: use of a single failure proof handling system, use of mechanical stops or electrical interlocks to keep heavy loads away from fuel or safe shutdown equipment, or analyzing the consequences of postulated heavy load drops to show these are within acceptable limits.

This report completes Task A-36.

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ACKNOWLEDGMENT

The following NRC staff members contributed to the evaluation and resolution of Task A-36.

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CONTROL OF HEAVY LOADS
AT NUCLEAR POWER PLANTS

1. INTRODUCTION

1.1 Background

In nuclear plant operation, maintenance, and refueling activities, heavy loads may be handled in several plant areas. If these loads were to drop, they could impact on stored spent fuel, fuel in the core, or equipment that may be required to achieve safe shutdown or permit continued decay heat removal. If sufficient stored spent fuel or fuel in the core were damaged and if the fuel is highly radioactive due to its irradiation history, the potential releases of radioactive mat could result in offsite doses that exceed 10 CFR Part 100 limits. If the load damaged equipment associated with redundant or dual safe shutdown paths, the capability to achieve safe shutdown may be defeated. Additionally, if fuel is of sufficient enrichment, the normal boron concentrations that are maintained may not be adequate to prevent criticality if a load drop caused a crushing of the fuel assemblies.

In this task a heavy load is defined as a load whose weight is greater than the combined weight of a single spent fuel assembly and its handling tool. The handling of a single spent fuel assembly has been reviewed in the original licensing review or in the Generic Issue "Fuel Handling Accident Inside Containment."

In previous licensing reviews, the extent to which the potential for accidental load drops has been considered varies from plant to plant, with current licensing reviews being the most thorough and some older plants receiving little attention in this area. The review criteria for current licensing reviews are contained in various Regulatory Guides (RGs) and Standard Review Plans (SRPs).

Task A-36 was established to systematically examine staff licensing criteria and the adequacy of measures in effect at operating plants, and to recommend necessary changes to assure the safe handling of heavy loads once a plant becomes operational.

With the increased spent fuel storage capacities at many operating plants, largely in the form of increased density of fuel storage within the pool, the potential for a given load to damage a large number of fuel assemblies has increased. Additionally, when offsite waste repositories are established, there will be an increased frequency in the handling of spent fuel casks over the spent fuel pools and near spent fuel. Because of this the need to complete Task A-36 expeditiously was identified.

This report provides the results of the review of the handling of heavy loads and includes the task group's recommendations on actions that should be taken to assure safe handling of heavy loads. This report completes Task A-36.

1.2 Definitions

For the purposes of this review, the following definitions were used:

Handling system - All load bearing components used to lift the load, including the crane or hoist, the lifting device, and interfacing load lift points.

Heavy load - Any load, carried in a given area after a plant becomes operational, that weighs more than the combined weight of a single spent fuel assembly and its associated handling tool for the specific plant in question.

"Hot" fuel - Fuel that was at power sufficiently long such that, if the fuel were damaged, offsite doses due to release of gap activity could exceed 1/4 of 10 CFR Part 100 limits. (Sufficient decay times are calculated in Section 2.1 for worst case conditions assuming an entire core is damaged. Fuel that has not decayed for the necessary decay time is "hot" spent fuel.)

"Load hang-up" event - The act in which the load block and/or load is stopped by a fixed object during hoisting, thereby possibly overloading the hoisting system.

Safe load travel path - A path defined for transport of a heavy load that will minimize adverse effects, if the load is dropped, in terms of releases of radioactive material and damage to safety systems. This path should be administratively controlled by procedures and/or clearly outlined by markings on the floor where the load is to be handled (refer to Section 5.1.1(1)). It may also be enforced by mechanical stops and/or electrical interlocks.

Safe shutdown equipment - Safety related equipment and associated subsystems that would be required to bring the plant to cold shutdown conditions or provide continued decay heat removal following the dropping of heavy load. Safety functions that should be preserved are: to maintain reactor coolant pressure boundary; capability to reach and maintain subcriticality; removal of decay heat; and to maintain integrity of components whose failure could result in excessive offsite release.

Special lifting devices - A lifting device that is designed specifically for handling a certain load or loads, such as the lifting rigs for the reactor vessel head or vessel internals, or the lifting device for a spent fuel cask.

Spent Fuel - Fuel that has been critical in the core and is considered no longer sufficiently active to be of use in powering the reactor and therefore is soon to be, or already has been, removed from the reactor. It generally has an enrichment of less than 0.9 weight percent U-235.

"Two-blocking" event - The act of continued hoisting to the extent that the upper head block and the load block are brought into contact, and, unless additional measures are taken to prevent further movement of the load block, excessive loads will be created in the rope reeving system, with the potential for rope failure and dropping of the load.

1.3 Task A-36 Review Process

The initial step was to evaluate the adequacy of existing measures at operating facilities. To do this the Office of Inspection and Enforcement was requested to gather and provide information for six BWR's and six PWR's on the heavy load handling systems at those facilities. It was found that this information was insufficient for the purposes of Task A-36. Accordingly, a generic letter was prepared and sent to all licensees, with responses requested from non-SEP facilities. (SEP facilities are those older operating facilities under review in the Systematic Evaluation Program to determine adequacy of these facilities with respect to current criteria.) Responses were received by December 1978. The task group then initiated a survey of this information to determine what heavy loads are typically handled, measures employed by licensees to prevent or mitigate the consequences of a heavy load drop, and analyses performed by licensees to show that potential consequences are within acceptable limits. The results of this survey are summarized in "Survey of Licensee Information," Section 3, of this report.

To determine the potential consequences of dropping certain of these heavy loads, analyses were performed by the task group. These analyses were aimed at identifying potential offsite radiological consequences due to postulated load drops, and the potential for a load drop to cause criticality in the reactor core or in the spent fuel pool. The results of these analyses are summarized in "Potential Consequences of a Load Drop," Section 2, of this report.

Concurrent with the above analyses, the task group reviewed historical data available on load handling accidents, including load drop events. Data obtained and reviewed covered various crane applications, including nuclear facilities, naval shore and shipboard installations, as well as industrial facilities to the extent that reports are provided to OSHA. The review of the data was aimed at identifying the principal causes of load handling accidents, and estimating the probability of a load drop event. The results of this data review are provided in "Review of Historical Data on Crane Operations," Section 4, of this report.

Based on the review of the historical data, guidelines were developed by the task group that were aimed at the principal causes of load handling accidents to reduce the potential for such events. Additionally, these guidelines include further measures to assure that accidental load drops are extremely unlikely or that the consequences of such load drops are within acceptable limits, based on the analyses of Section 2, "Potential Consequences of a Load Drop." These guidelines are provided in "Recommended Guidelines," Section 5.1, of this report.

Certain of the alternative approaches suggested by the guidelines of Section 5.1 call for analyses of postulated load drops for the specific plant. These may include such things as an analysis of a spent fuel shipping cask drop or the drop of a reactor vessel head. Guidelines for performing such analyses are contained in "Analyses of Postulated Load Drops," Appendix A, of this report.

Section 5.2, "Bases for Guidelines," includes certain fault trees. Fault trees were developed for several of the alternatives suggested by the guidelines. Probabilities were then estimated or calculated for various faults or events, and used with the fault trees to determine the likelihood of obtaining unacceptable consequences with any of these alternatives.

Section 5.3 of this report is the staff "Safety Evaluation."

The existing criteria in Regulatory Guides and Standard Review Plans were then evaluated to determine the required changes to incorporate the guidelines of Section 5.1 that are appropriate for new plants.

Final recommendations of the task group were developed and are included as "Resolution of the Issue," Section 6, of this report. The recommendations listed in Section 6 are a summary of the recommendations contained in the various sections of this report.

1.4 Summary of Recommendations

Guidelines were developed that offer various alternatives to licensees to assure the safe handling of heavy loads. These "Recommended Guidelines" in Section 5.1 include general guidelines for all facilities to reduce the potential for the uncontrolled movement of a load or a load drop, such as by calling for: definition of safe load paths; development of load handling procedures, periodic inspection and testing of the crane; qualifications, training and specified conduct of the crane operator; and use of guidelines on rigging. Additionally, the guidelines define various acceptable alternative approaches for the containment building, refueling building and other safety related areas. These alternatives may include using a single-failure-proof handling system, analyzing the effects of a load drop, or using procedures and interlocks to keep loads away from spent fuel and safe shutdown equipment.

We have recommended a program to review operating plants against these guidelines. A draft generic letter has been prepared to obtain the required information and commitments. We have also recommended that: certain interim measures be taken for operating plants until completion of this program; changes be made to Standard Review Plans and Regulatory Guides; and changes to technical specifications be made after completion of the review.

2. POTENTIAL CONSEQUENCES OF A LOAD DROP

An accidental load drop could impact nuclear fuel or safety-related equipment with the potential for excessive offsite releases, inadvertent criticality, loss of water inventory in the reactor or spent fuel pool, or loss of safe shutdown equipment. The following sections discuss the potential for these adverse consequences to occur. Section 5 will provide recommended guidelines to prevent or mitigate these potential consequences.

2.1 Potential Offsite Releases Due To Heavy Load Drops On Spent Fuel

The analysis of the potential consequences of a heavy load drop onto spent fuel assemblies contained in this section is based primarily on the methods and assumptions used for fuel handling accidents as shown in the Standard Review Plan 15.7.4, "Radiological Consequences of Fuel Handling Accidents," NUREG-75/087, and Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

In a fuel handling accident analysis, we assume that a spent fuel assembly having the minimum decay time permitted (100 hrs or whatever value is used in the technical specifications) is being moved under water in the spent fuel storage pool. It is postulated that the fuel assembly drops from its maximum height in the pool and impacts upon the floor of the pool. This impact is assumed to rupture the cladding on the equivalent of all of the fuel rods in a fuel assembly causing a release of fission product gases which were contained in the space or gap between the fuel pellets and the cladding. The percent of inventory assumed to be released is based on guidelines in Regulatory Guide 1.25. The fission product gases released are about 10 standard cubic feet (0.3 cubic meters). The gas bubbles are released to the fuel pool water where they rise to the surface of the pool. The water is assumed to scrub out approximately 99% of the iodine fission products (Iodine 131-135) but is not assumed effective in reducing the quantity of noble gases released to the fuel building atmosphere. Once the radioactive gases reach the spent fuel building atmosphere, they are normally exhausted to the environment through a charcoal filter system which further reduces the quantity of airborne radioactive iodines. This filter is not effective in removing noble gases such as krypton and xenon which contribute to the whole body dose. A large release of radioactivity from the containment building can be prevented by rapid isolation of containment upon a high radiation signal; however, the size of the release will depend on the response time of such a system. The analyses in this section assume that the noble gases are not contained.

The postulated dose consequences of a heavy load drop on fuel assemblies in either the spent fuel pool area or in the reactor can be determined as a multiple of a single assembly fuel handling accident, once the total number of damaged fuel assemblies has been ascertained. Conversely, one may use the results of the analysis of damage to one assembly and determine the number of assemblies which must be damaged to reach certain limits on radioactive releases.

The exposure limits of 10 CFR Part 100 have been established for certain design basis accidents whose probability is sufficiently low that they "would result in potential hazards not exceeded by those from any accident considered credible" (10 CFR 100.11, footnote 1). For accidents of higher probability, the NRC staff has judged that lower dose acceptance criteria are appropriate.

The staff has for several years identified fuel handling and spent fuel cask drop accidents as two members of the class of limiting faults for which the radiological dose acceptace criteria are stated to be "well within" 10 CFR Part 100 guidelines. Other accidents in this class include control rod ejection (PWR)/control rod drop (BWR), waste gas system failures, and some steam line breaks and steam generator tube ruptures. The staff has, in all operating license and contruction permit reviews, interpreted this criterion as less than or equal to 25% of 10 CFR Part 10 values. This specific criterion has also been enunciated in position C.3 of Regulatory Guide 1.17, Rev. 1 for purposes of identifying systems requiring tornado protection. The staff therefore judges that the allowable exposures for a postulated heavy load drop onto irradiated fuel should be similar to that used for fuel handling accidents, and has therefore used one-fourth of the 10 CFR Part 100 values as an upper bound on the allowable exposure for such events.

Table 2.1-1 provides the results of analyses of postulated fuel handling accidents which damage a single assembly. To arrive at the results of Table 2.1-1, the assumptions used in the heavy load drop analyses are summarized in Table 2.1-2. Table 2.1-1 also lists the corresponding number of fuel assemblies that have to be damaged to yield doses of 75 rem thyroid or 6.25 rem whole body which are one-fourth of the 10 CFR Part 100 limits. Doses are provided for various decay times after going subcritical, when credit is taken for charcoal filters and when credit is not taken for filters. The latter calculations were done because certain existing operating plants remove wall sections or roof hatches when handling of the cask near spent fuel. From Table 2.1-1 it can be seen that for short decay times, exposure limits could be reached by damaging only a few assemblies.

If the results of Table 2.1-1 are plotted as shown in Figure 2.1-1 for PWRs and Figure 2.1-2 for BWR's, time after shutdown can be determined such that if a given number of fuel assemblies were damaged, it would not result in excessive offsite releases. These Figures may then be used to predict if radiological consequences would be within required limits if a given number of assemblies is damaged in a postulated load drop accident, based on the shutdown time. These Figures may also be used to show minimum required decay times if the postulated load drop could result in release of gap activity from an entire core. For a two unit facility, a core off-load of each unit could result in two cores being located in the same spent fuel pool. For such facilities minimum required decay times for worst case conditions should be based on potential damage to two cores.

As noted in Definitions, Section 1.2, spent fuel which has not decayed as long as the appropriate specified decay time is defined for the purposes of this report as "hot" spent fuel. The above decay times have been incorporated into certain alternatives in the guidelines of Section 5.1 of this report through the definition of "hot" spent fuel.

Applicability

The assumptions used in these analyses were selected so as to be bounding for nearly all plants, and thus lead to generic conclusions. To rely on these analyses for a specific plant, the licensee or applicant should verify that the assumptions used adequately scope the specifics of the plant.

If the assumptions are not conservative for the specific plant, or if a more accurate analysis is required for a specific plant, the results can be modified by a ratio of the plant power level or χ/Q values. Similarly, if other than 95% filter efficiency is provided in the spent fuel pool filters, the results can be obtained by a ratio of penetrations (i.e., 1.0 minus the efficiency) for both elemental and organic forms of radioactive iodine.

TABLE 2.1-1

SUMMARY OF LOAD

DROP ACCIDENT ANALYSES

No. of Days Subcritical	Exclusion Radius Dose[1/] Thyroid	Exclusion Radius Dose[1/] Whole Body	Low Population Zone[1] Thyroid	Low Population Zone[1] Whole Body	Minimum No. of Ass to Reach 1/4 of Part 100 Limits
PWR					
4 (no filters)	173.00	0.61	17.30	0.06	1
4 (w/filters)	8.63	0.5	0.86	0.06	9
40 (no filters)	7.74	0.01	0.77	0.00	12
40 (w/filters)	0.39	0.01	0.04	0.00	1.9×10^2
54 (no filters)	2.28	0.00	0.23	0.00	33
54 (w/filters)	0.11	0.00	0.01	0.00	5.8×10^2
90 (no filters)	0.10	0.00	0.01	0.00	7.5×10^2
90 (w/filters)	0.01	0.00	0.00	0.00	7.5×10^3
120 (no filters)	0.01	0.00	0.00	0.00	7.5×10^3
120 (w/filters)	0.00	0.00	0.00	0.00	7.5×10^3
BWR					
1 (no filters - SBGT)	92.26	0.70	9.23	0.07	1
1 (w/filters - SBGT)	4.61	0.65	0.46	0.07	17
40 (no filters)	2.67	0.00	0.27	0.00	28
40 (w/filters)	0.13	0.00	0.01	0.00	5.8×10^2
90 (no filters - SBGT)	0.04	0.00	0.00	0.00	1.9×10^3
90 (w/filters - SBGT)	0.00	0.00	0.00	0.00	1.6×10^4
120 (no filters)	0.00	0.00	0.00	0.00	1.6×10^4
120 (w/filters)	0.00	0.00	0.00	0.00	1.6×10^4

[1/] Dose per fuel assembly damaged (rems).

[2/] Number of assemblies that must be damaged to approach (1/4 of) Part 100 exposure limits, or 75 rem thyroid and 6.25 rem whole body (at exclusion area boundary).

TABLE 2.1-2

HEAVY LOAD DROP ACCIDENT ASSUMPTIONS

	Reactor Type	PWR and BWR
	Power Level (Mwt)	3,000
0-2 hour	X/Q (Exclusion area boundary), sec/M^3	1.0×10^{-3} 1/
0-2 hour	X/Q LPZ, sec/M^3	1.0×10^{-4} 1/
	Peaking Factor	1.2 2/
	No. of Assemblies in Core	193(PWR), 760(BWR)
	Pool Water Decontamination Factor	100 3/ (for radioactive iodines)
	Filter Efficiency %:	
	Elemental Iodine	95% 4/
	Organic Iodine	95%
	Cooling Time (hours)	100 or greater

1/ Based on 5% worst meteorological conditions.

2/ Value is 1.2 for greater than one damaged fuel assembly. For a single assembly the values are 1.65 and 1.5 for PWRs and BWRs, respectively.

3/ See Reg. Guide 1.25

4/ See Reg. Guide 1.52

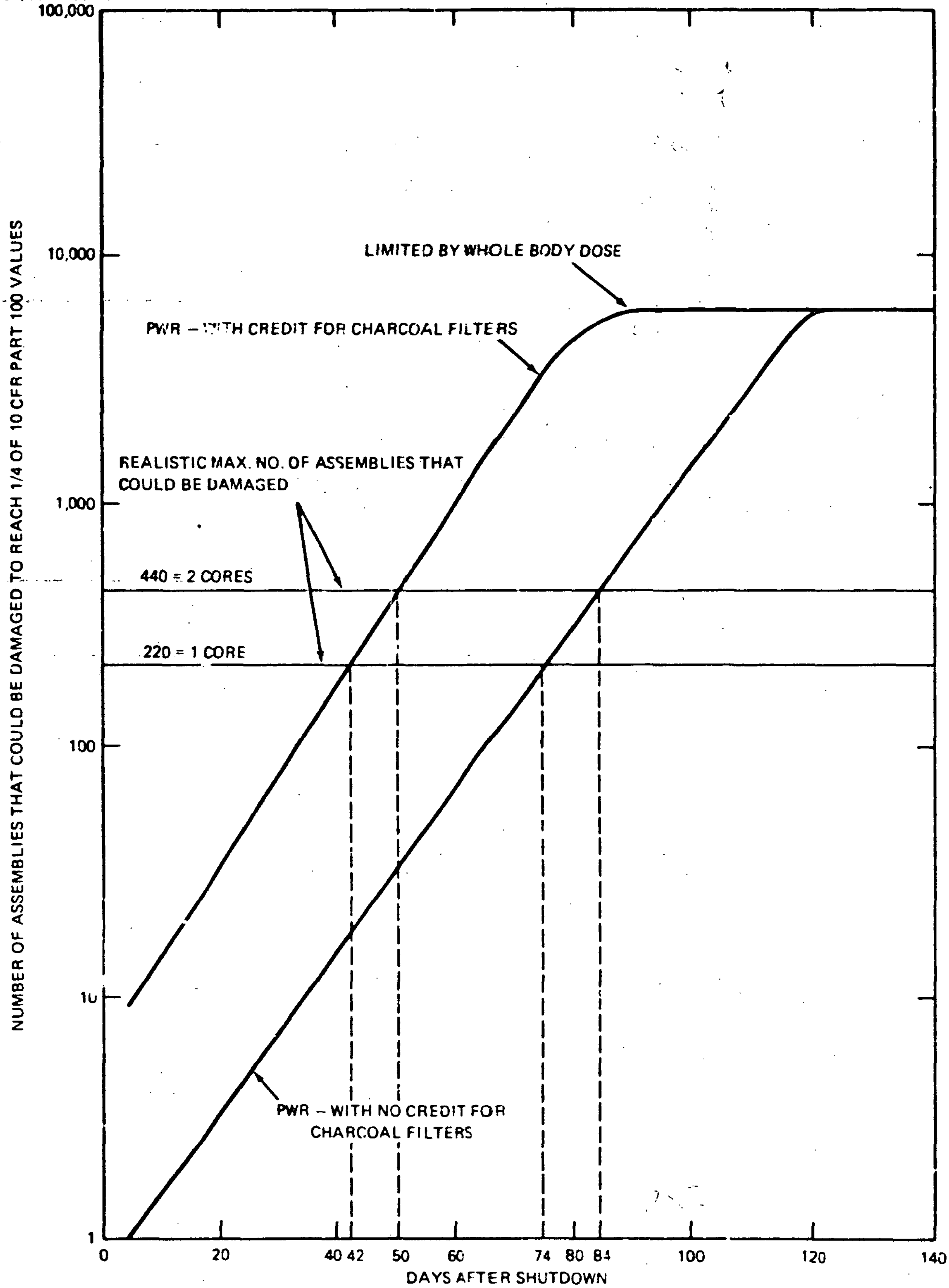

FIGURE 2.1-1 (PWR)
NUMBER OF FUEL ASSEMBLIES THAT COULD BE DAMAGED TO REACH 1/4 OF 10 CFR PART 100 LIMITS VS TIME AFTER REACTOR SHUTDOWN

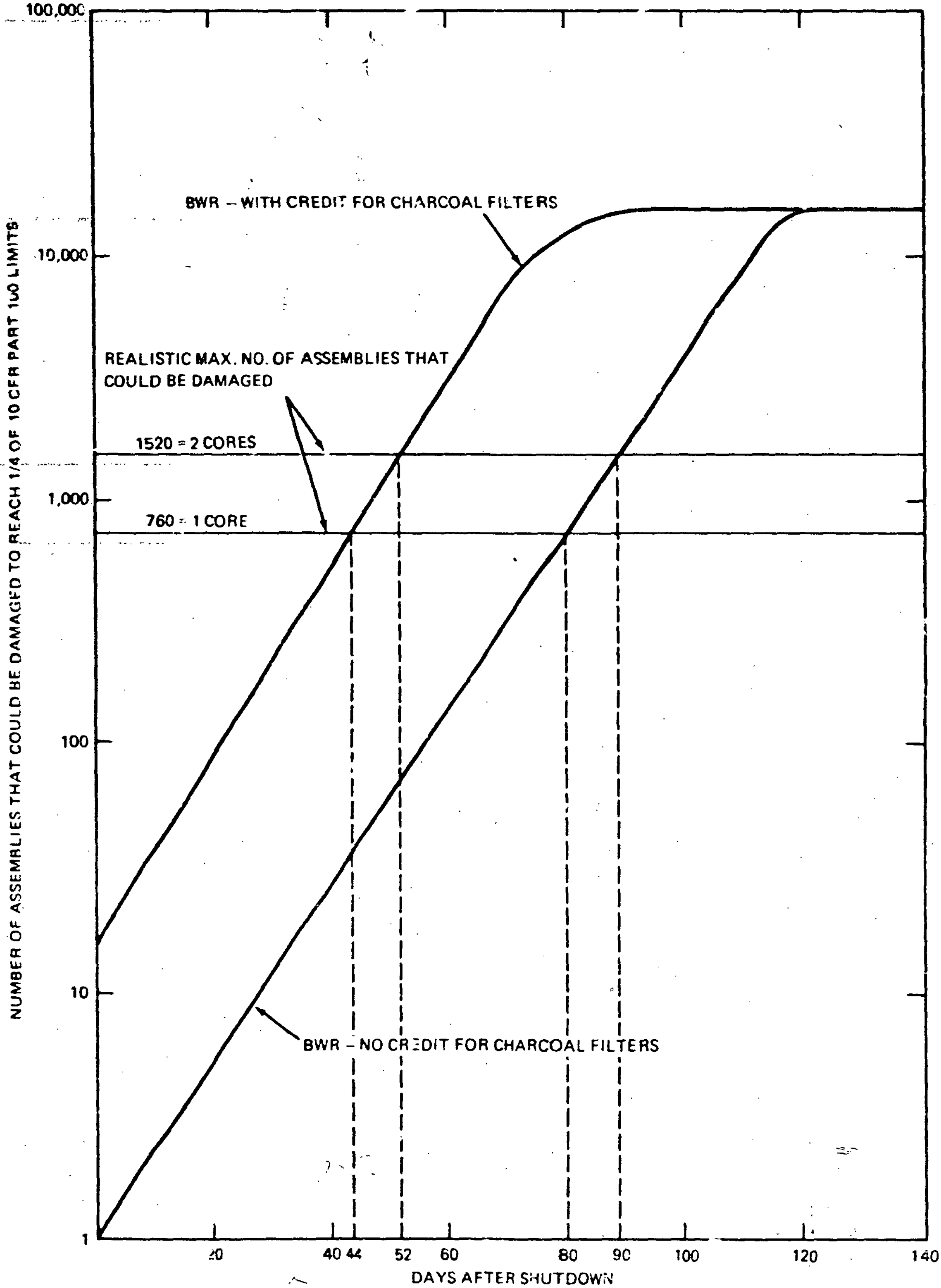

FIGURE 2.1-2 (BWR)
NUMBER OF FUEL ASSEMBLIES THAT COULD BE DAMAGED TO REACH 1/4 OF 10 CFR PART 100 LIMITS VS TIME AFTER REACTOR SHUTDOWN

2.2 Criticality Considerations

2.2.1 Introduction

In addition to the potential for a dropped load to cause a release of radioactive material due to rupture of the fuel clad, the dropped load potentially can change the spacing of the fuel lattice as well as the boron concentration. This effect could result in a critical mass of fuel in the reactor core or in the spent fuel pool. Due to design differences between vendors, most noticeably between PWR and BWR vendors, the potential for a load drop to result in fuel becoming supercritical varies. The following sections discuss this potential for BWR and PWR reactor cores and spent fuel pools.

2.2.2 Effect of Enrichment, Lattice Spacing and Boron Concentration

Figures 2.2-1, 2, 3, and 4, based on an infinite lattice study, illustrate the effect of enrichment, lattice spacing, and boron concentration on k_{eff}. While these figures are computed for Westinghouse 15 x 15 fuel, the trends shown hold for all commercial fuel designs. Typical refueling water concentrations are 2,000 ppm for PWR reactor cavities and storage pools, although this varies (see Section 2.2.5.1); and 0 ppm for BWR reactor cavities and storage pools. The following characteristics of commercial fuels are important to the present discussion:

(1) As built lattice spacings in the core for all fuel designs are chosen so that in pure water k_{eff} is near the maximum value that can be attained by adjusting lattice spacing. This can be seen in Figures 2.2-1, 2, 3, and 4, where the water/uranium ratio is an indicator of lattice spacing. PWR fuel is undermoderated in pure water (lattice spacing chosen to the left of the peak) throughout the cycle for both hot and cold conditions. At high boron concentrations, PWR fuel is highly overmoderated so that decreasing the lattice spacing (i.e., reducing the water/uranium ratio by crushing the fuel) increases k_{eff}.

(2) BWR fuel is undermoderated at hot conditions throughout the cycle; at cold conditions it is undermoderated at beginning-of-cycle and slightly overmoderated at end-of-cycle.

Because of this, decreasing the lattice spacing from its as-built value (i.e., reducing the water:uranium ratio by crushing the fuel) in pure water at end of cycle will slightly increase k_{eff} (but not above 0.95 due to the low enrichment at end-of-cycle), and at beginning of cycle will cause a decrease in k_{eff}.

Approximate levels of k_{eff} for fuels under different conditions are shown in Tables 2.2-1, 2, 3, and 4. Acutal numbers may vary somewhat from the numbers in these tables depending on fuel design; the numbers in these tables are meant only to serve as a guide to determine which fuel configurations have a potential for criticality resulting from a load drop.

The analyses of Figures 2.2-1, 2, 3, and 4 and Tables 2.2-1, 2, 3, and 4 are for an infinite array of fuel pins with no solid boral poison or steel or aluminum structural material, which are also neutron poisons. Because

Parameters are as follows. (The dimensions used are those of Westinghouse 15 x 15 fuel)

Fuel Pellet Diameter	0.3659"
Zirc Clad Inside Diameter	0.3734"
Zirc Clad Outside Diameter	0.4220"
As-Built W/U Ratio	1.647
Temperature	20 DEGC
Fuel Material	0.9 w/o U235

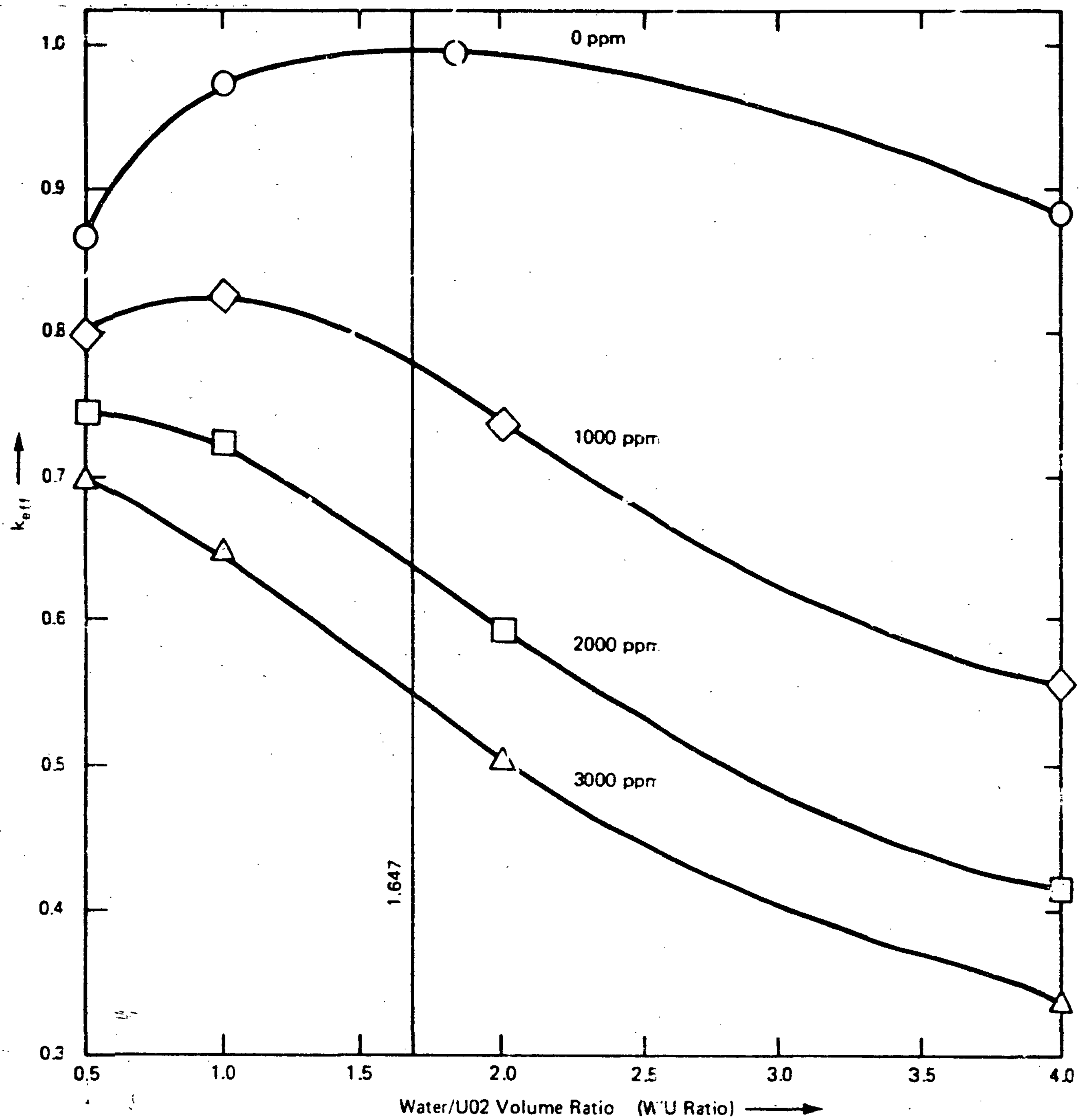

FIGURE 2.2-1
NEUTRON MULTIPLICATION FACTOR FOR INFINITE ARRAY OF FUEL RODS IN BORATED WATER

Parameters are as follows. (The dimensions used are those of Westinghouse 15 x 15 fuel)

Fuel Pellet Diameter	0.3659"
Zirc Clad Inside Diameter	0.3734"
Zirc Clad Outside Diameter	0.4220"
As-Built W/U Ratio	1.647
Temperature	20 DEGC
Fuel Material	2.0 w/o U235

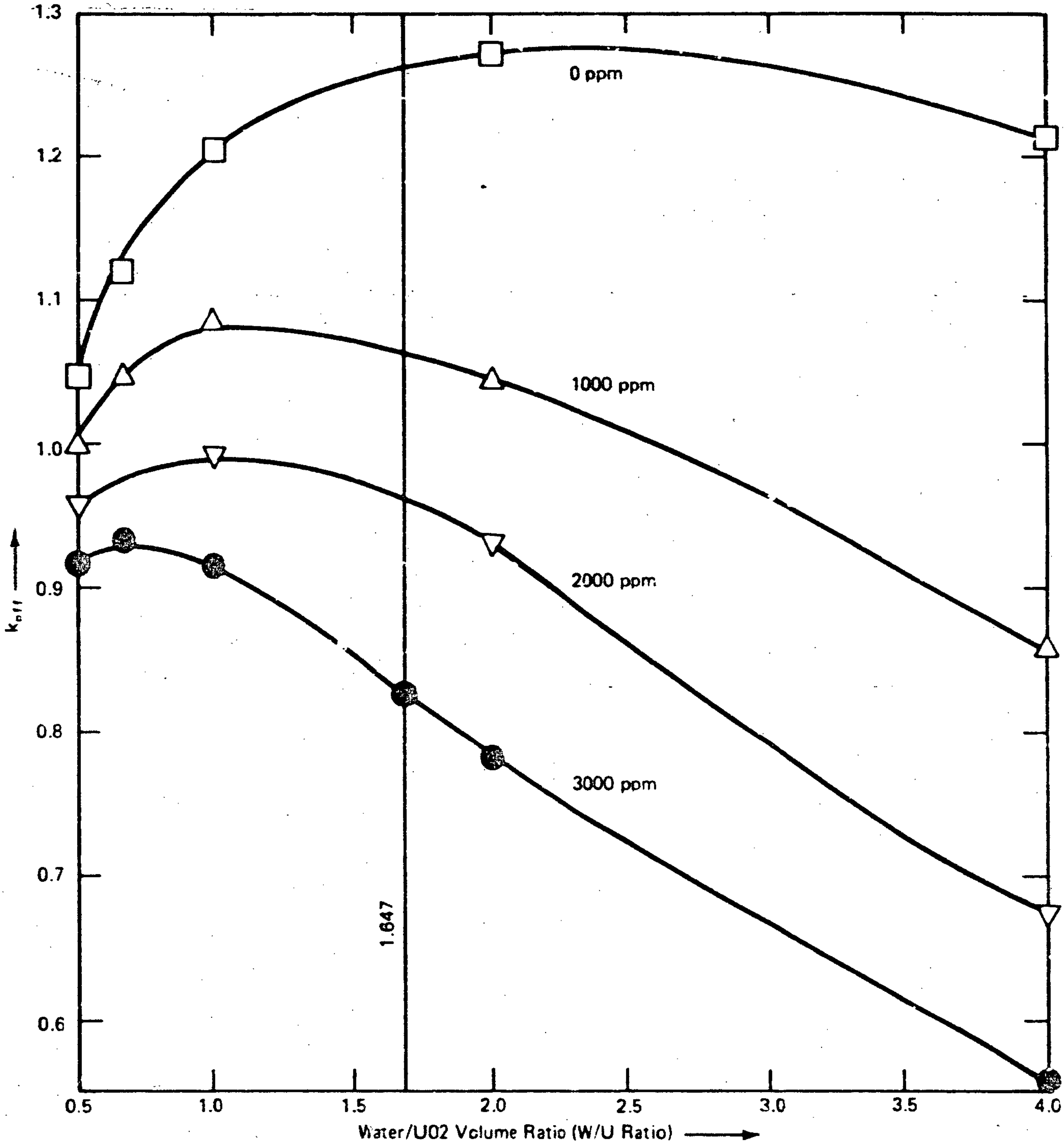

FIGURE 2.2-2
NEUTRON MULTIPLICATION FACTOR FOR INFINITE ARRAY OF FUEL RODS IN BORATED WATER

Parameters are as follows. (The dimensions used are those of Westinghouse 15 x 15 fuel)

Fuel Pellet Diameter	0.3659"
Zirc Clad Inside Diameter	0.3734"
Zirc Clad Outside Diameter	0.4220"
As-Built W/U Ratio	1.647
Temperature	20 DEGC
Fuel Material	3.5 w/o U235

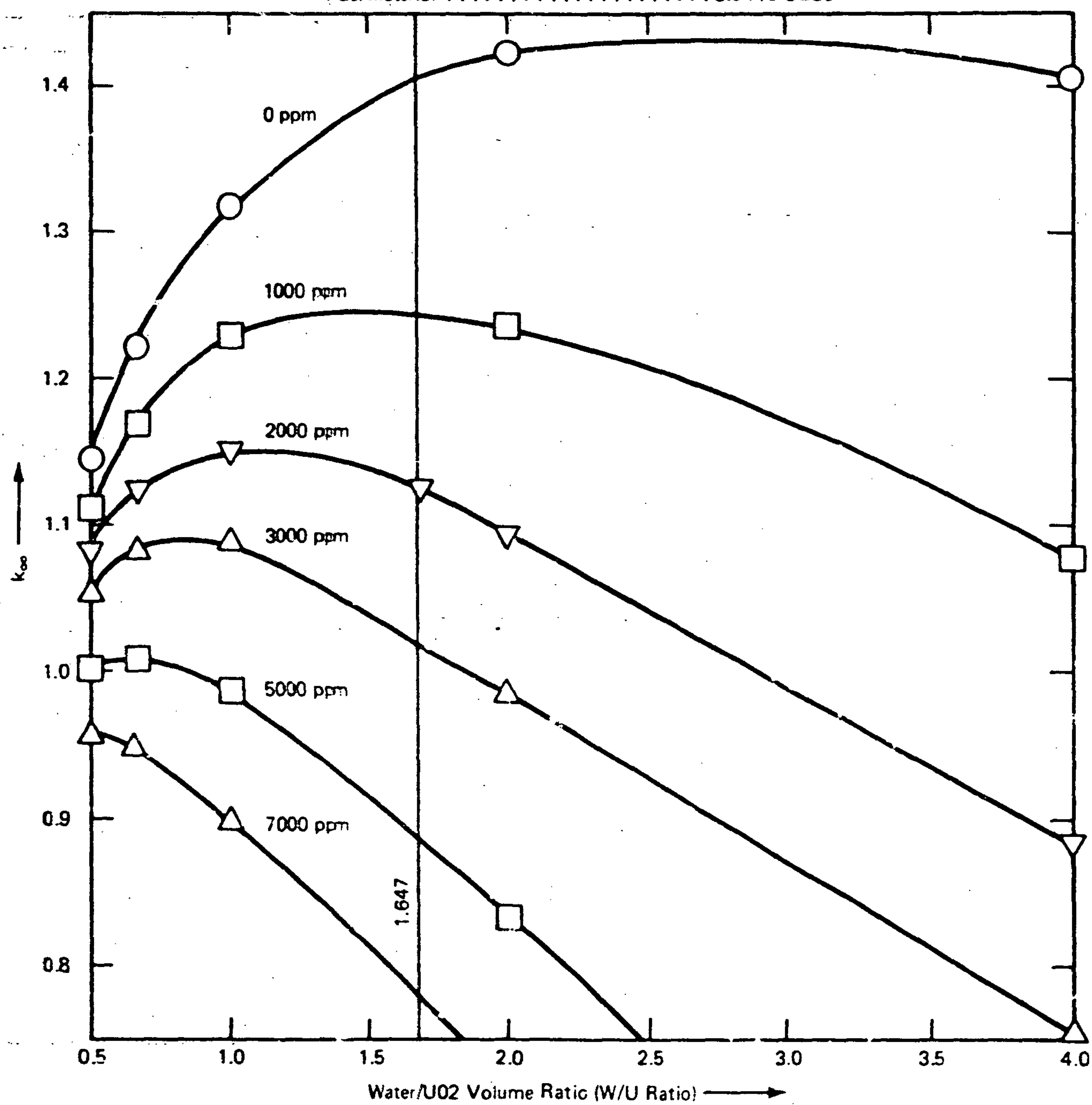

FIGURE 2.2-3
NEUTRON MULTIPLICATION FACTOR FOR INFINITE ARRAY OF FUEL RODS IN BORATED WATER

Parameters are as follows. (The dimensions used are those of Westinghouse 15 x 15 fuel)

Fuel Pellet Diameter	0.3659"
Zirc Clad Inside Diameter	0.3734"
Zirc Clad Outside Diameter	0.4220"
As-Built W/U Ratio	1.647
Temperature	20 DEGC
Fuel Material	5.0 w/o U235

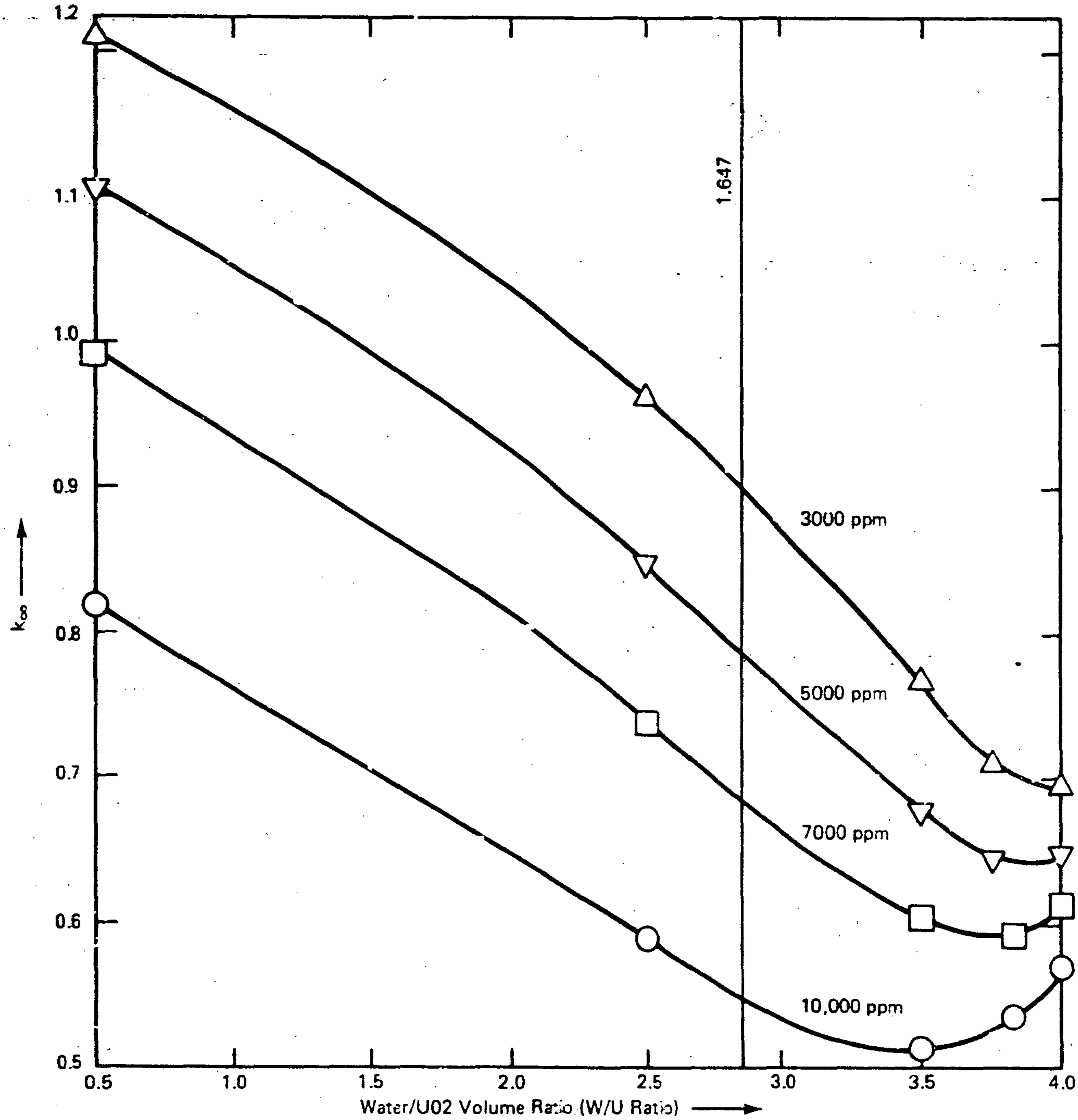

FIGURE 2.2-4
NEUTRON MULTIPLICATION FACTOR FOR INFINITE ARRAY OF FUEL RODS IN BORATED WATER

TABLE 2.2-1

APPROXIMATE K_{EFF} FOR 0.9 W/O U-235 FUEL UNDER DIFFERENT ACCIDENT CONDITIONS*

Condition of Fuel and Fuel Rack	Boron Concentration (ppm)					
	0	1000	2000	3000	4000	5000
1. Fuel and Fuel Rack Intact	0.40	0.36	0.30	0.20	--	--
2. Fuel Intact Rack Crushed So That Fuel Bundles Touch Or Alternatively Condition of Fuel in Cores	0.19	0.78	0.64	0.50	--	--
3. Rack and Fuel Crushed to Maximize k_{eff}	0.99	0.80	0.74	0.72	--	--

*0.9 w/o is typical enrichment for discharged fuel, and thus is representative of normal spent fuel pool conditions, but without boron poison plates.

TABLE 2.2-2

APPROXIMATE K_{EFF} FOR 2.0 W/O U-235 FUEL UNDER DIFFERENT ACCIDENT CONDITIONS*

Condition of Fuel and Fuel Rack	Boron Concentration (ppm)					
	0	1000	2000	3000	4000	5000
1. Fuel and Fuel Rack Intact	0.80	0.65	0.50	0.35	--	--
2. Fuel Intact Rack Crushed So That Fuel Bundles Touch Or Alternatively Condition of Fuel in Cores	1.26	1.07	0.97	0.83	--	--
3. Rack and Fuel Crushed to Maximize k_{eff}	1.28	1.08	0.99	0.93	--	--

*2.0 w/o U-235 is a typical core average enrichment for reload cores (after reload).

TABLE 2.2-3

APPROXIMATE K_{EFF} FOR 3.5 W/O U-235 FUEL UNDER DIFFERENT ACCIDENT CONDITIONS*

Condition of Fuel and Fuel Rack		Boron Concentration (ppm)					
		0	1000	2000	3000	5000	7000
1.	Fuel and Fuel Rack Intact	0.95	0.81	0.67	0.53	--	--
2.	Fuel Intact Rack Crushed So That Fuel Bundles Touch Or Alternatively Condition of Fuel in Cores	1.40	1.24	1.13	1.02	0.92	0.82
3.	Rack and Fuel Crushed to Maximize k_{∞}	1.43	1.24	1.15	1.09	1.01	0.96

*In the past 3.5 w/o U-235 has been the fuel rack design basis enrichment. Most fuel rack analyses have been performed using 3.5 w/o U-235. However, recently enrichments greater than 3.5 w/o have been used at some plants. Thus in this report we are considering enrichments up to 5.0 w/o U-235.

TABLE 2.2-4

APPROXIMATE K_{EFF} FOR 5.0 W/O U-235 FUEL UNDER DIFFERENT ACCIDENT CONDITIONS*

Condition of Fuel and Fuel Rack		Boron Concentration (ppm)			
		3000	5000	7000	10,000
1.	Fuel and Fuel Rack Intact	--	--	--	--
2.	Fuel Intact Rack Crushed So That Fuel Bundles Touch Or Alternatively Condition of Fuel in Cores	1.15	1.02	0.92	0.73
3.	Rack and Fuel Crushed to Maximize k_{∞}	1.19	1.11	1.06	1.01

*5.0 w/o is maximum enrichment for new fuel, and thus is representative of worst case spent fuel pool conditions for a limited number of assemblies in the pool.

of this the pin cell calculations and the results in lines 2 and 3 of these tables are at least slightly conservative for load drop conditions; for racks that contain a large amount of boral poison, these calculations and results are very conservative. Line 1 of these tables is based on the fuel rack design basis, and is applicable to all rack designs, whether or not they contain these neutron poisons.

The critical mass of PWR and BWR fuel in pure water is of interest in criticality estimates, and will be noted here. For PWRs, typically, 2 fresh adjacent fuel assemblies constitute a critical mass. For BWRs, typically, 14 to 20 fresh fuel assemblies, which always contain gadolinium poison, constitute a critical mass. During service, the gadolinium is depleted before the uranium, and hence, the reactivity of BWR assemblies increases during the first part of their service life. At maximum reactivity, typically 6 BWR assemblies constitute a critical mass.

2.2.3 Fuel Rack Design Basis

Currently, all fuel racks are designed to be subcritical under two conservative assumptions: (1) the rack and fuel are immersed in unborated water, (2) the rack is an infinite array in the x-y plane completely filled with fuel of the highest enrichment expected to be used at the plant. In review of spent fuel pool designs, the staff requires the licensee to demonstrate, by computation, that under these assumptions the k_{eff} for the spent fuel is equal to or less than 0.95. This computation must conservatively account for all uncertainties. Many, but not all, licensees have chosen to account for the uncertainties on a 95%/95% confidence/probability tolerance limit basis.

In the past the highest fuel enrichments encountered were as follows: (1) PWR fuel enriched to 3.5 w/o U235, and (2) BWR fuel enriched to give a k_{eff} of 1.35 for an array of adjacent fuel assemblies. However, with the introduction of new fuel management schemes, some PWR licensees are using fuel enrichments as high as 4.2 w/o U235, and in the future we may see even higher enrichments. For those licensees using the more highly enriched fuel, the fuel rack design basis should be changed to reflect the actual enrichments used.

The criticality calculations for rack designs are performed using combinations of diffusion codes, transport codes, and Monte Carlo codes. All calculational models are benchmarked against critical experiments.

2.2.3.1 Standard Rack Designs

The geometrical details of fuel rack designs in use as of this writing are illustrated in Figures 2.2-5-1 through 2.2-5-11. These are arbitrarily labeled Type 1 through Type 11 for the purposes of this report. The legend for these figures is given in Figure 2.2-5-12. Types 1 through 5 are for PWR fuel and Types 6 through 11 are for BWR fuel. Some rack types are one-of-a-kind or several-of-a-kind, and this is indicated in the figures. These figures are not drawn to scale, but rather the steel, aluminum, and boral are drawn thicker than as-built for ease in illustration.

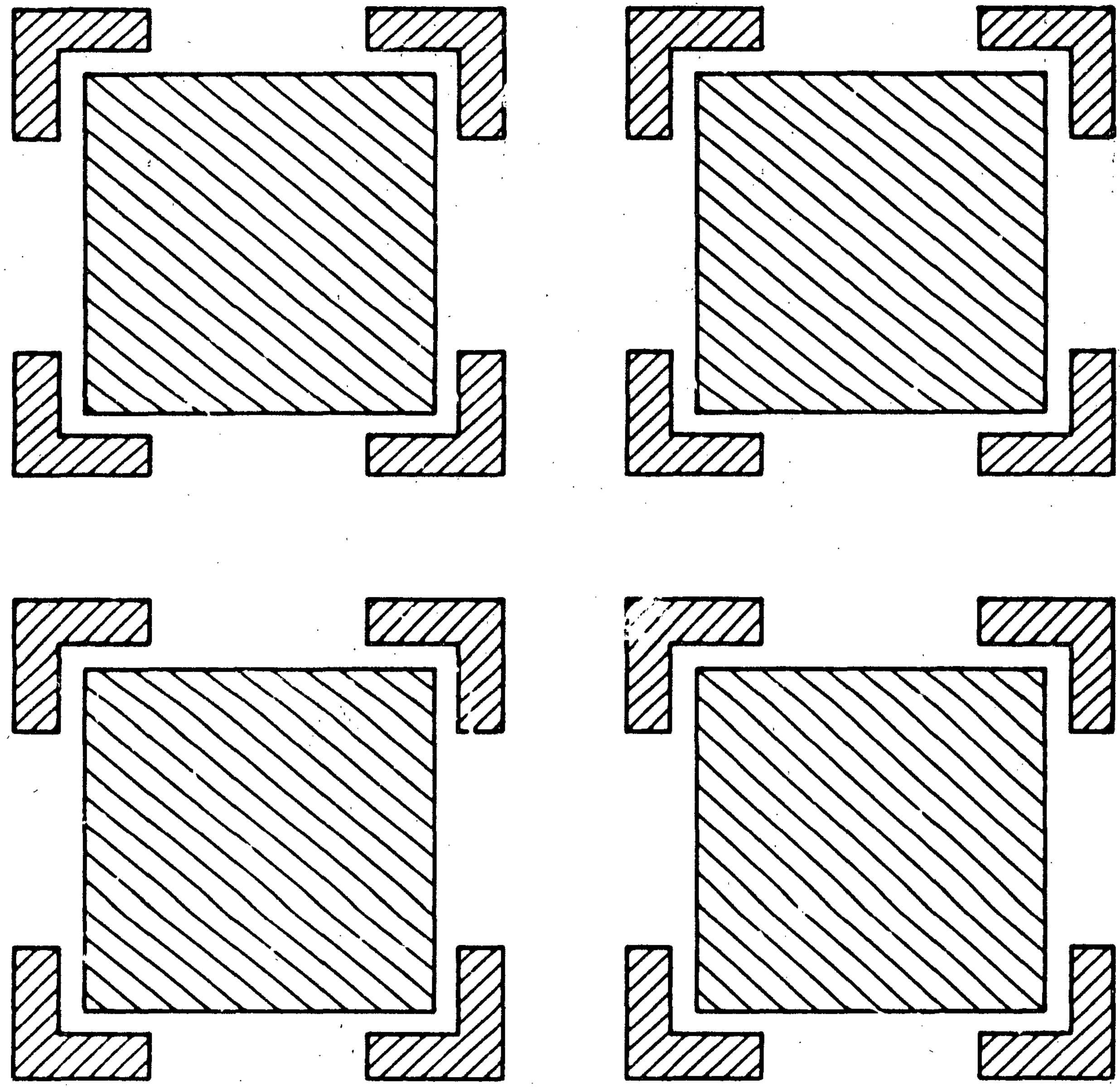

FIGURE 2.2-5-1 TYPE 1 SPENT FUEL RACK – PWR

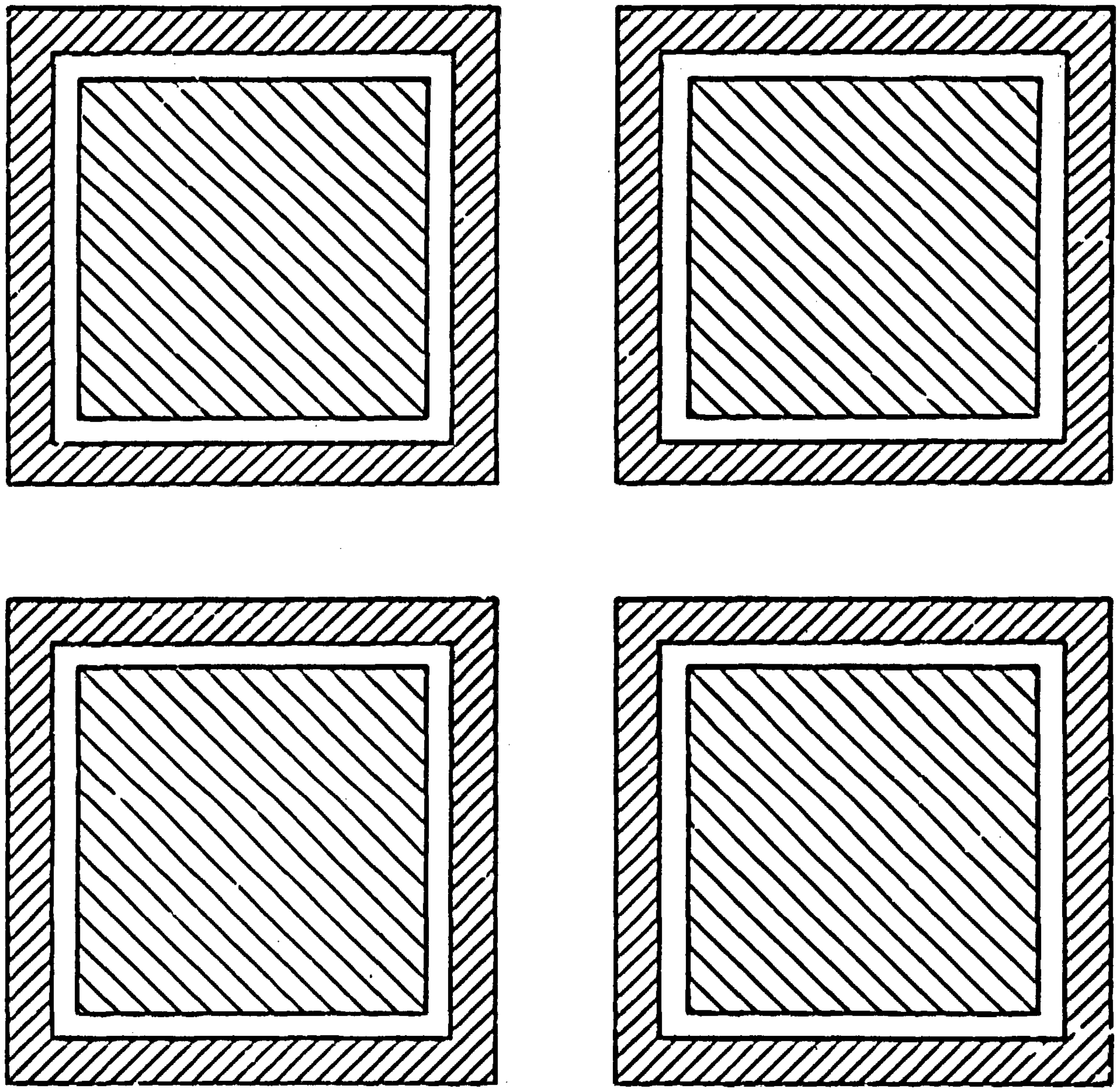

FIGURE 2.2-5-2 TYPE 2 SPENT FUEL RACK – PWR

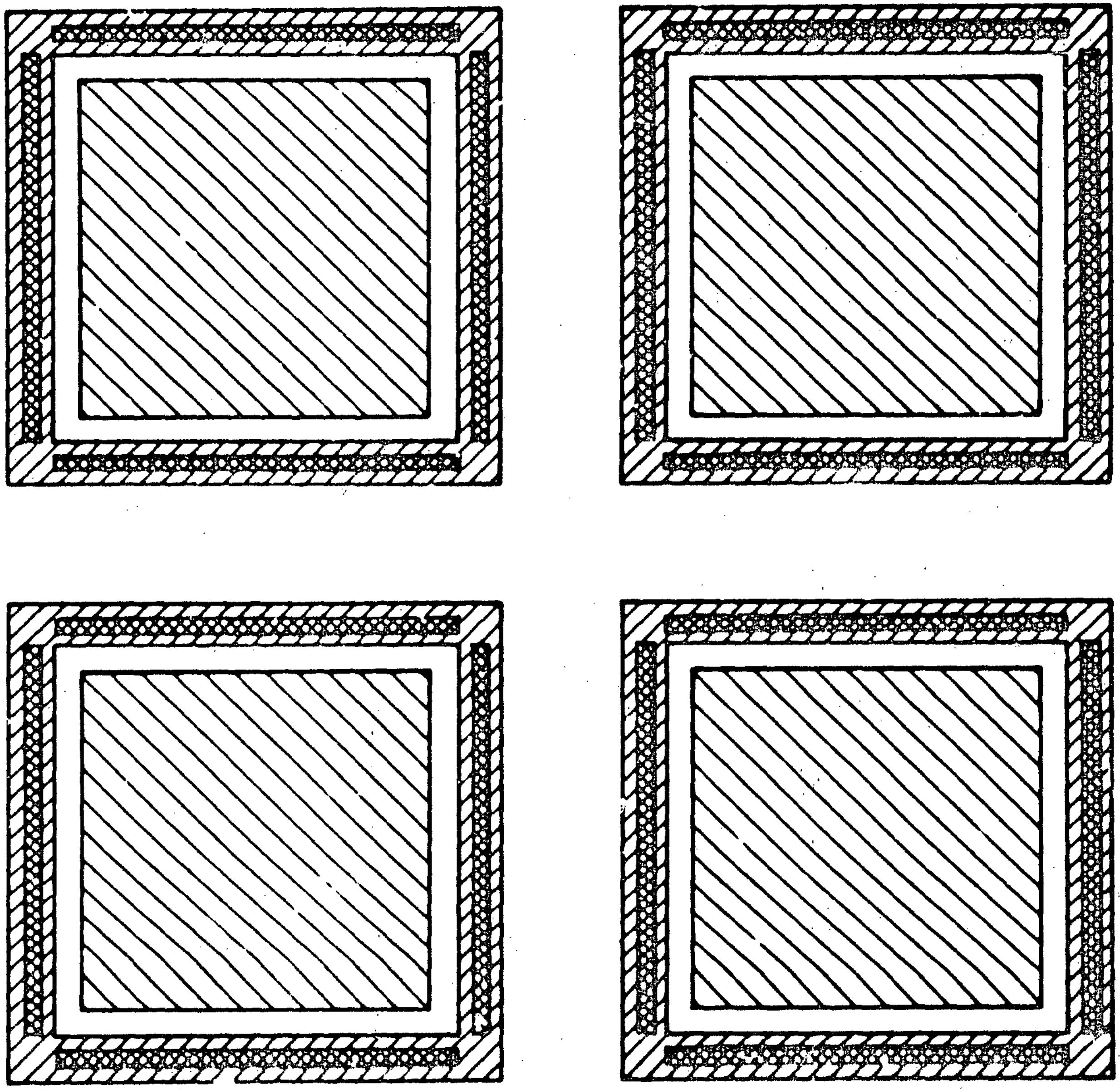

FIGURE 2.2-5-3 TYPE 3 SPENT FUEL RACK – PWR

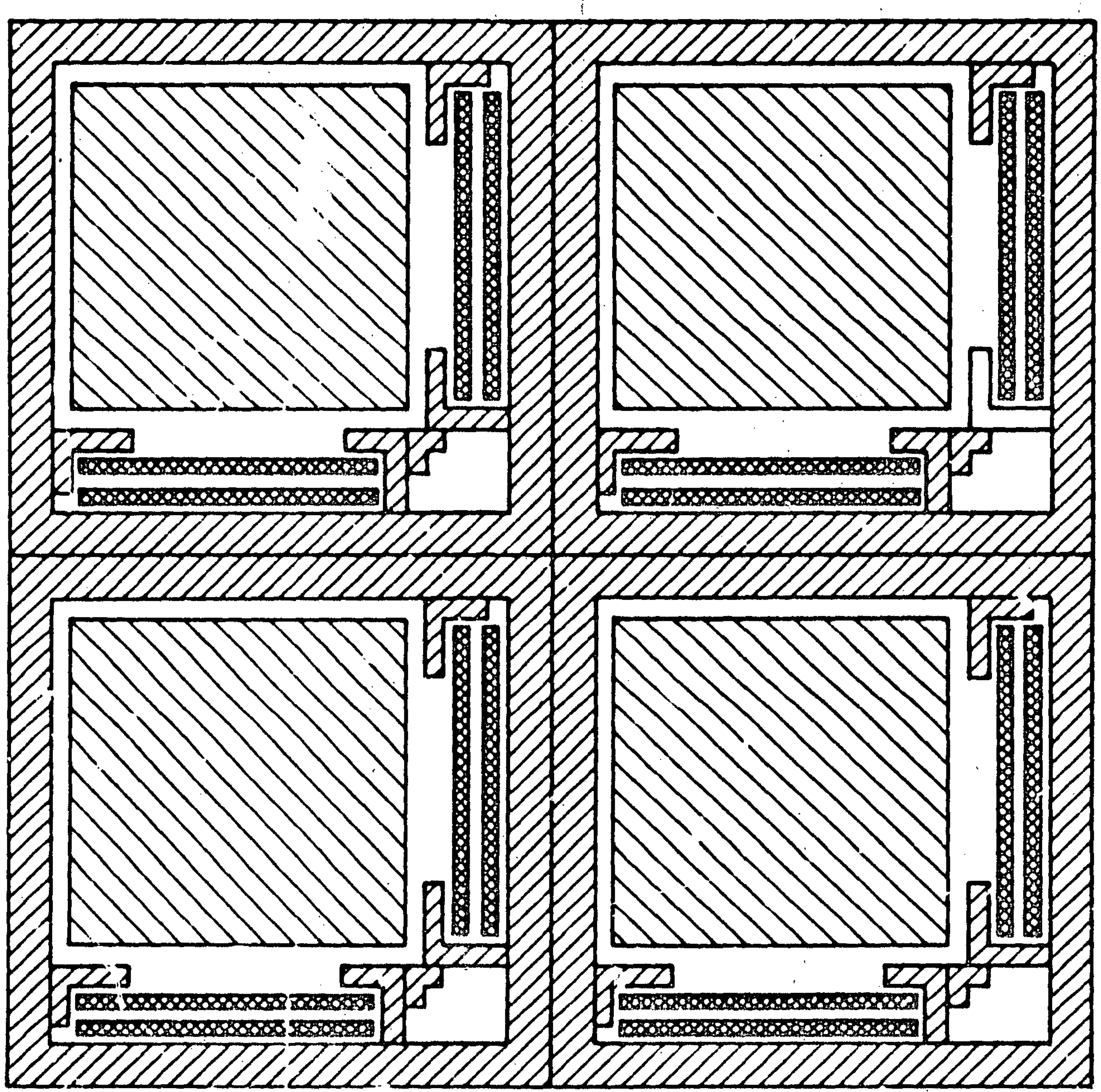

FIGURE 2.2-5-4 TYPE 4 SPENT FUEL RACK – PWR
THIS ONE-OF-A-KIND RACK IS INSTALLED AT POINT BEACH

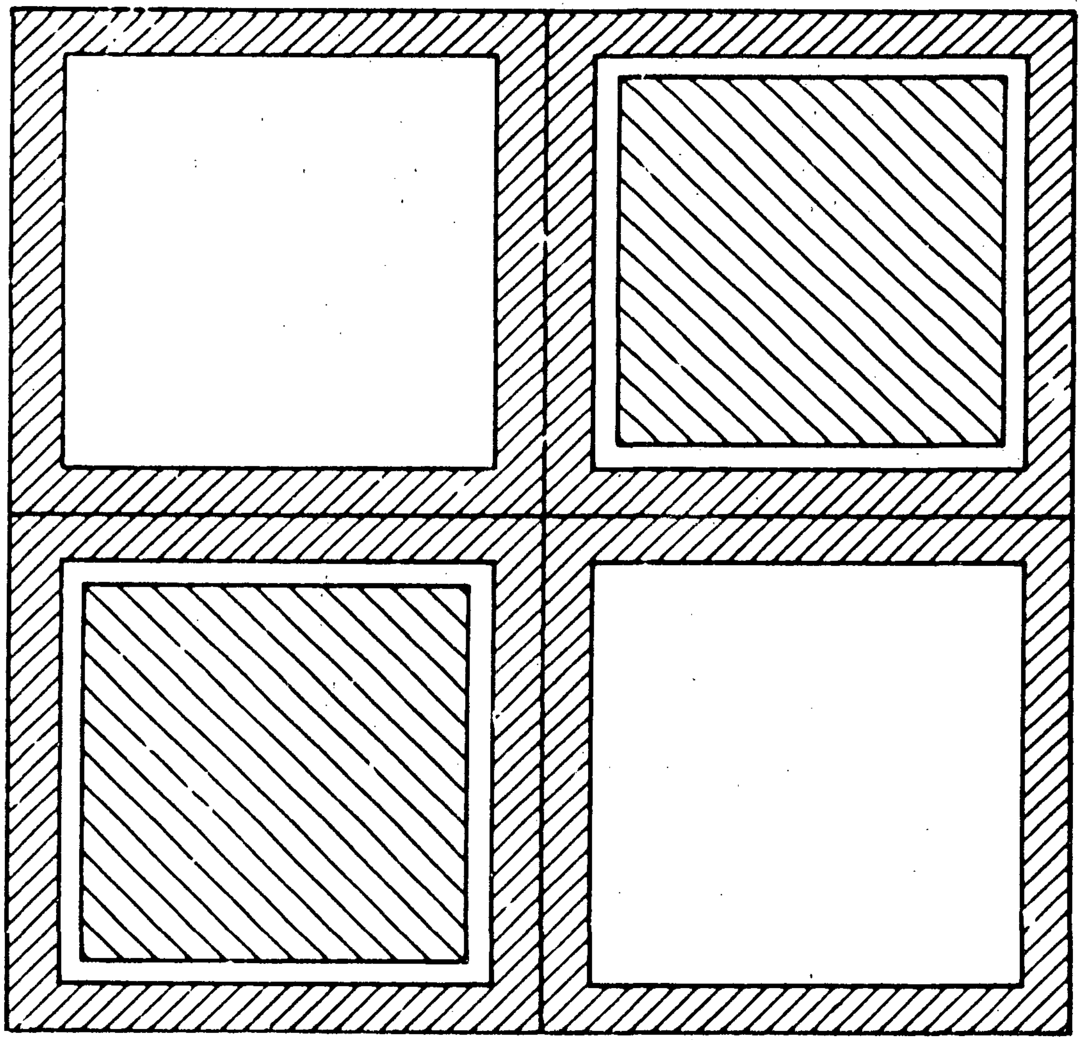

FIGURE 2.2-5-5 TYPE 5 SPENT FUEL RACK – PWR
THIS ONE-OF-A-KIND RACK IS INSTALLED AT GINNA

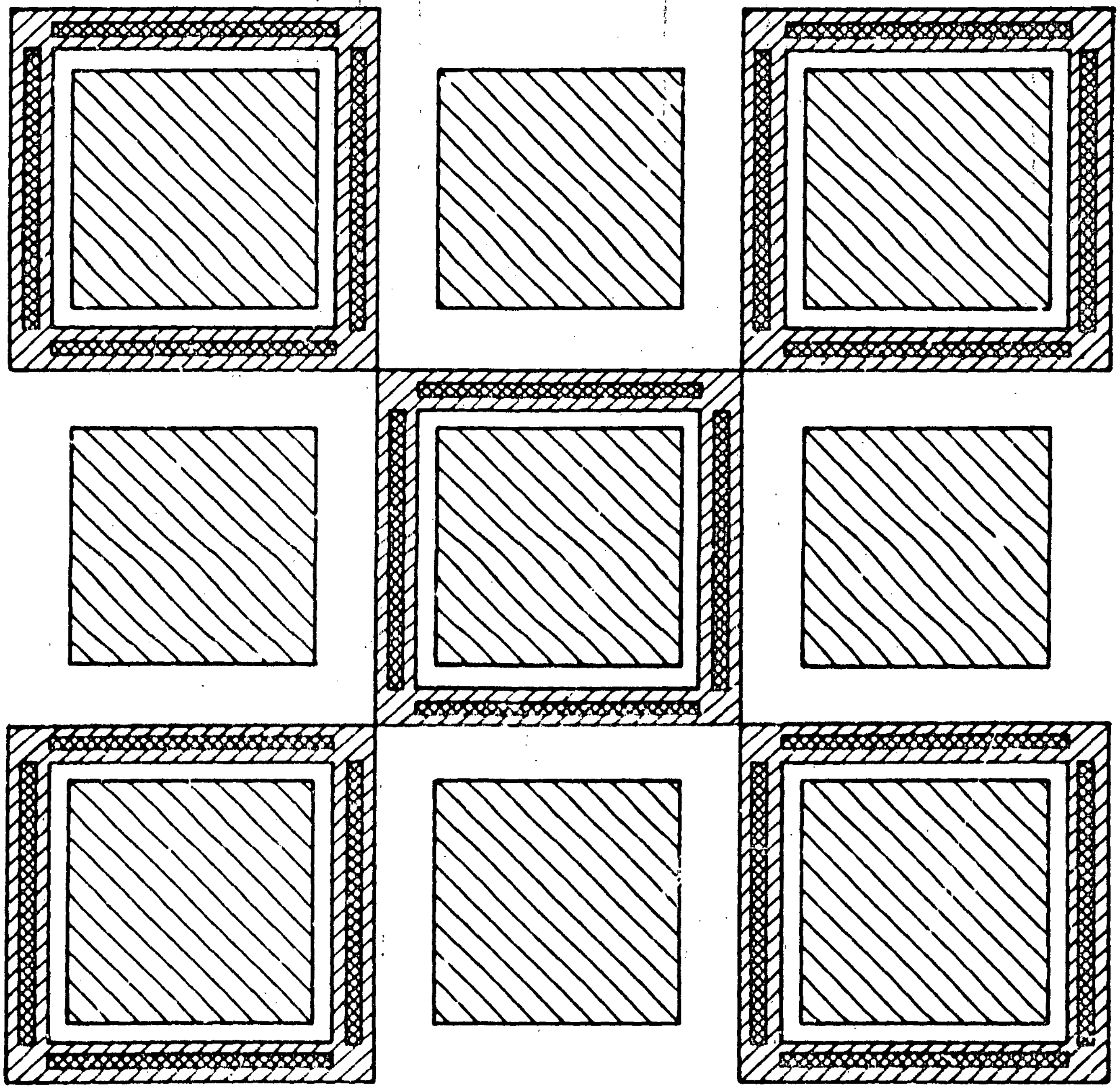

FIGURE 2 2-5-6 TYPE 6 SPENT FUEL RACK – BWR

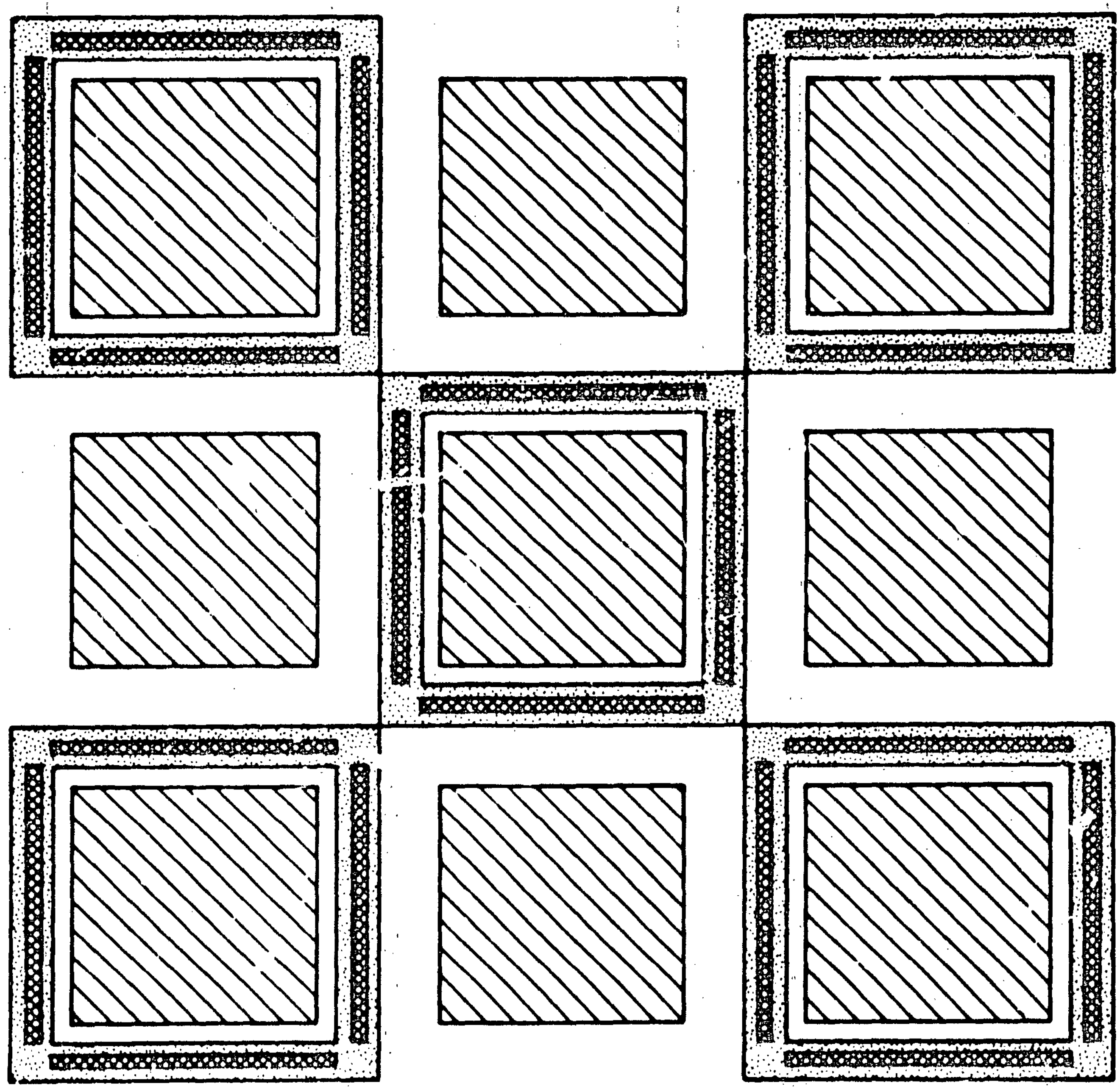

FIGURE 2.2-5-7 TYPE 7 SPENT FUEL RACK – BWR

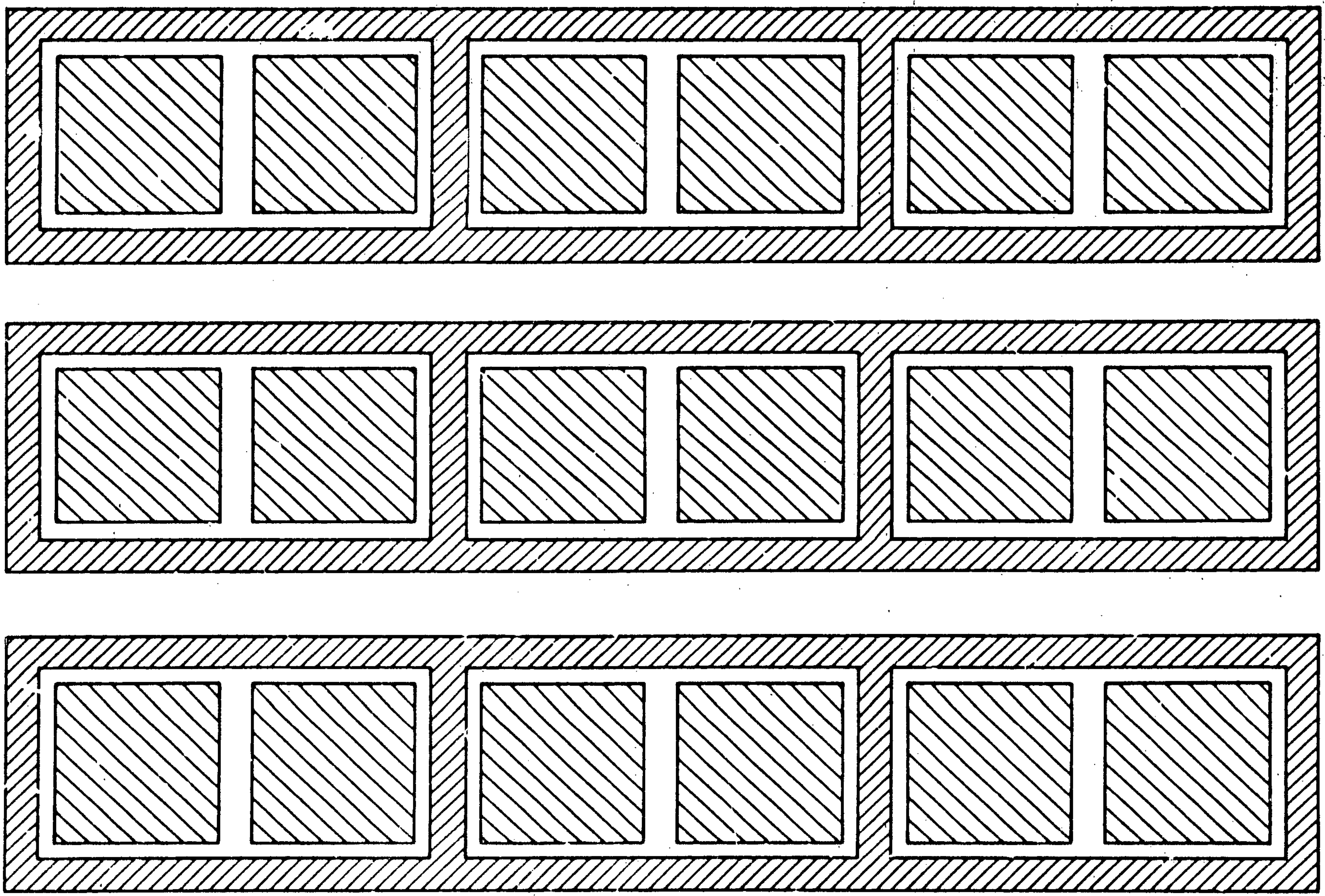

FIGURE 2.2-5-8 TYPE 8 SPENT FUEL RACK — BWR
THIS TWO-OF -A-KIND RACK IS INSTALLED AT OYSTER CREEK AND NINE MILE POINT

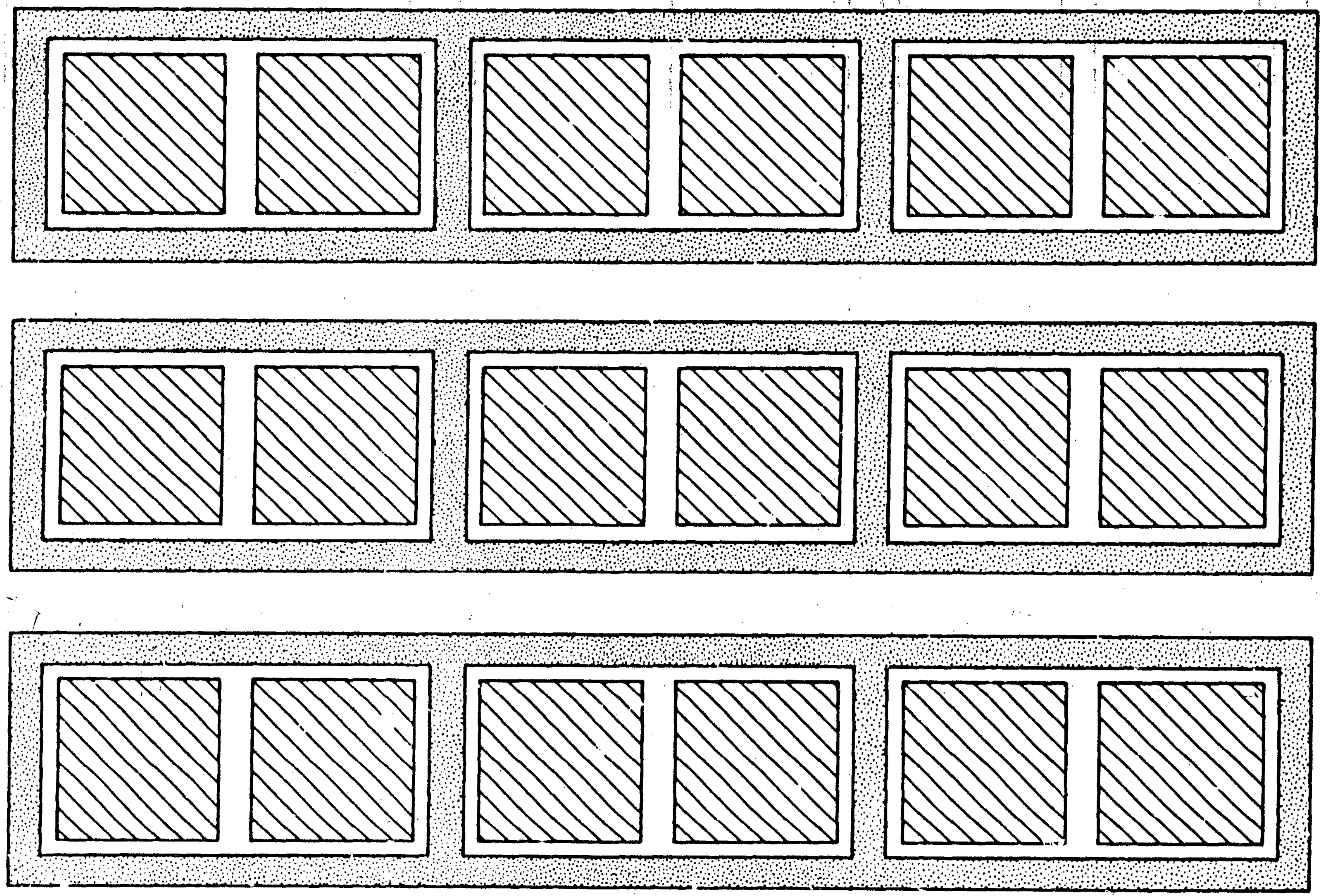

FIGURE 2.2-5-9 TYPE 9 SPENT FUEL RACK – BWR
THIS TWO-OF-A-KIND RACK IS INSTALLED AT DRESDEN 1 AND QUAD CITIES 1 & 2.

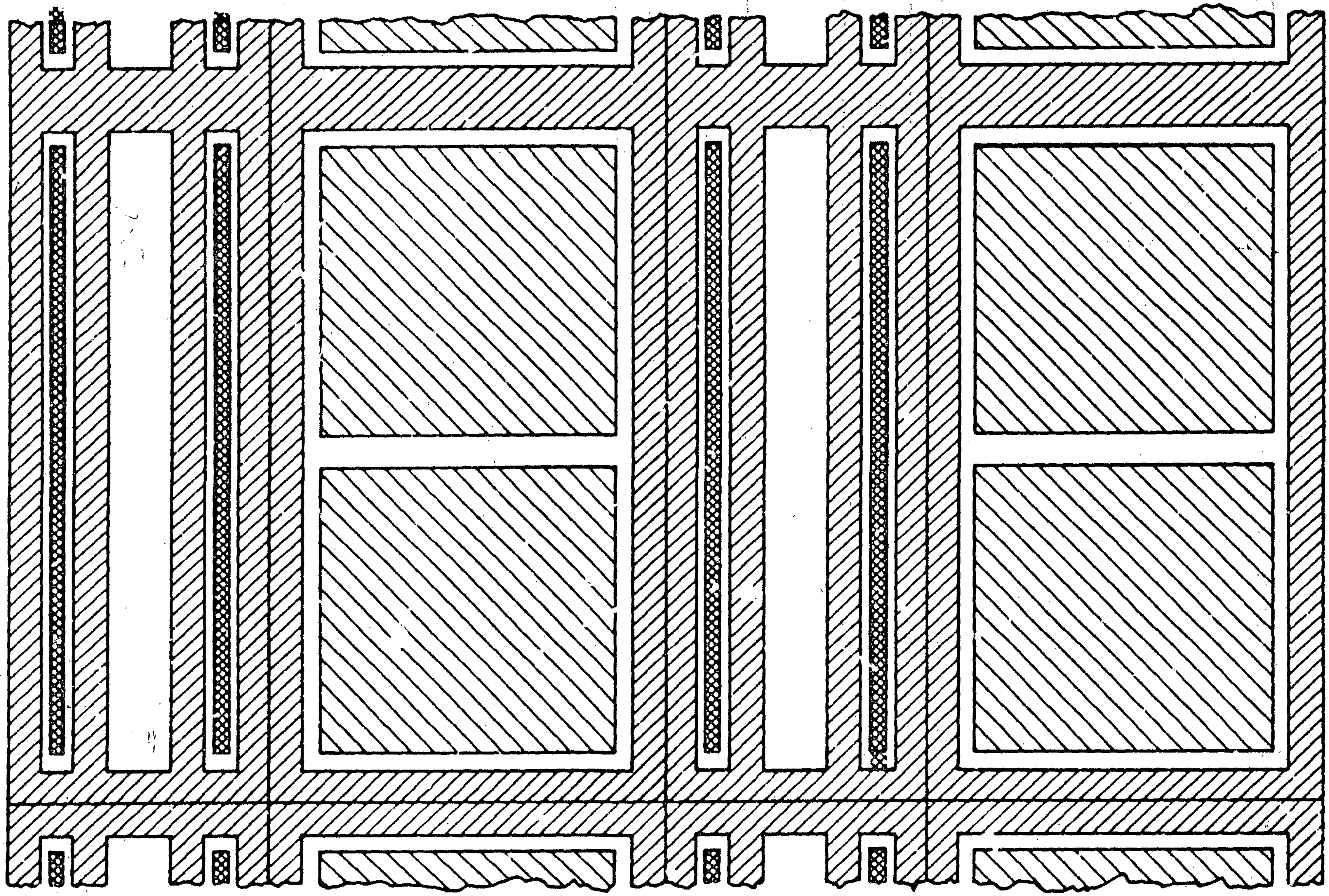

FIGURE 2.2-5-10 TYPE 10 SPENT FUEL RACK – BWR
THIS ONE-OF-A-KIND RACK IS INSTALLED AT NINE MILE POINT

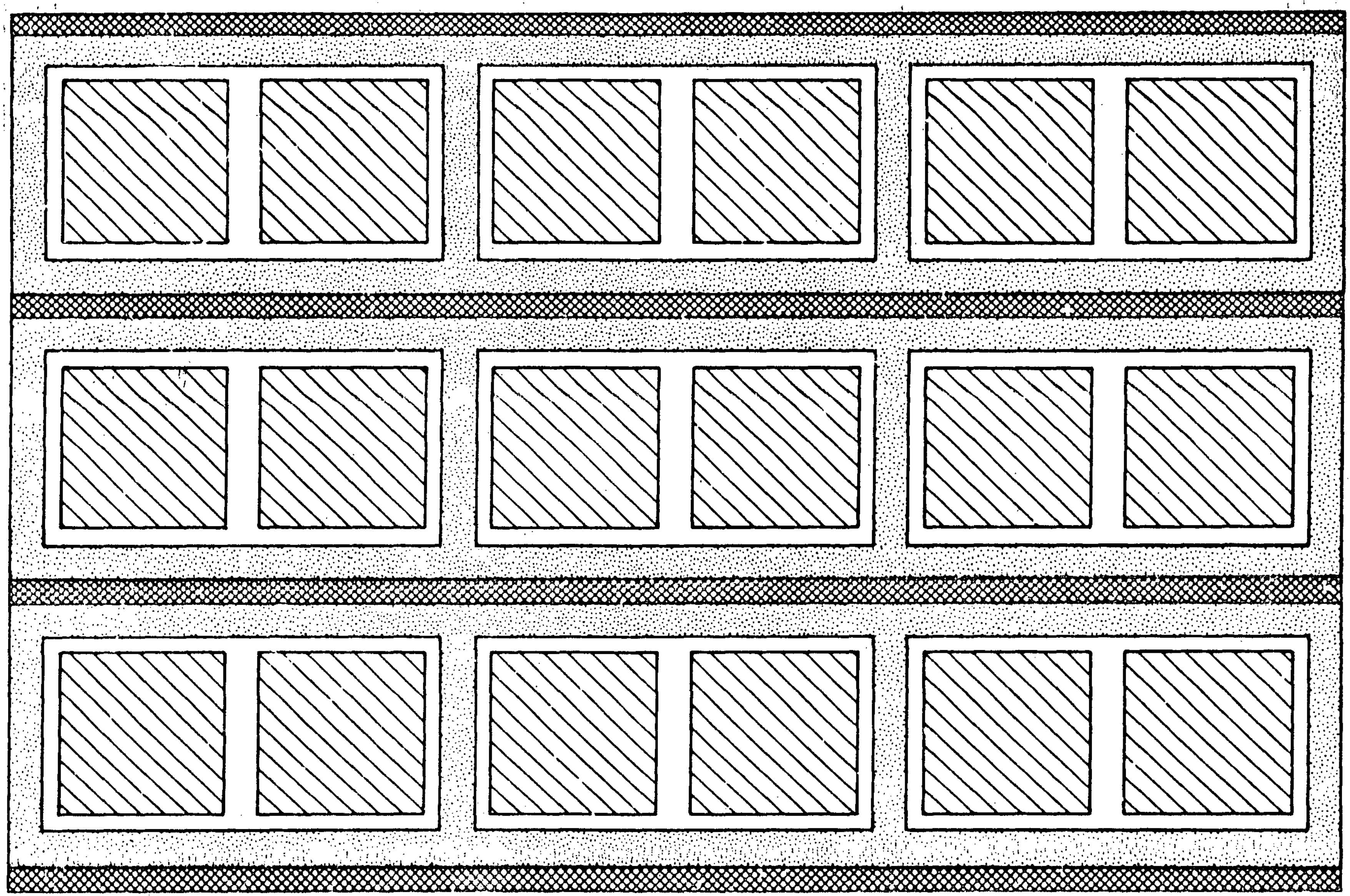

FIGURE 2.2-5-11 TYPE 11 SPENT FUEL RACK – BWR
THIS ONE-OF-A-KIND RACK IS INSTALLED AT COOPER

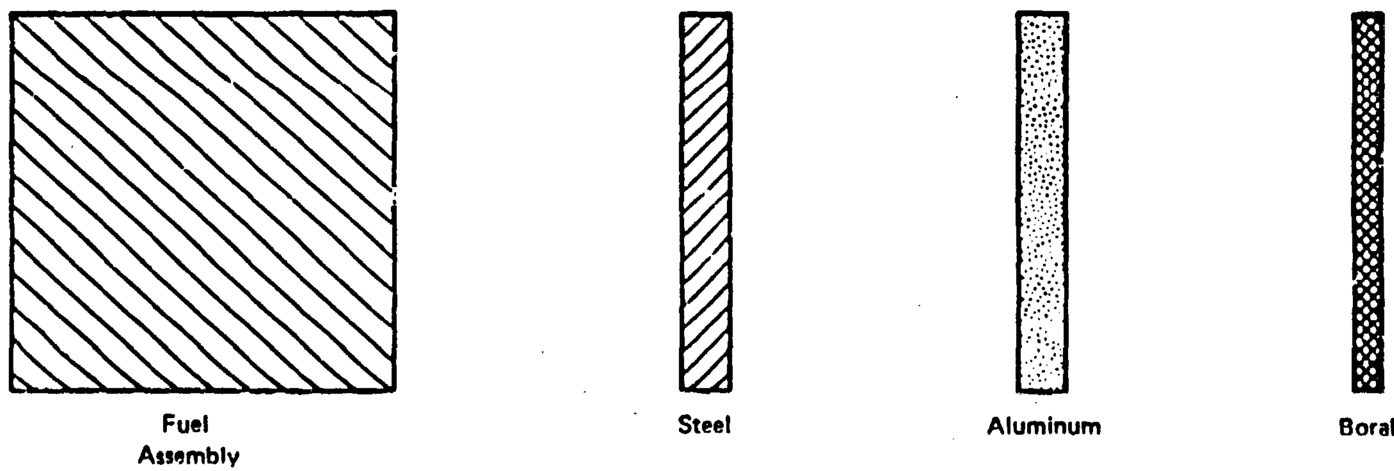

FIGURE 2.2-5-12 LEGEND FOR FIGURES 2.2-5-1 THRU 2.2-5-11

2.2.4 Potential for Criticality of BWR Fuel

2.2.4.1 BWR Spent Fuel Rack Design

The new high density BWR racks are composed of arrays of cans containing boral neutron poison. The spent fuel pools are filled with unborated water, rather than borated water, as is the case with the PWR spent fuel pools. Most BWR licensees demonstrate for these racks a k_{eff} of about 0.86 for fuel in the racks which would have a k_{eff} of 1.35 in pure water in the reactor core lattice with the control rods removed.

BWR spent fuel racks of type 8 and 9, both of which are two-of-a-kind racks, and type 10, which is a one-of-a-kind rack depend almost entirely on fuel separation to maintain the fuel subcritical, and contain no boral poison. For these racks, however, there is some neutron absorption in the steel or aluminium of the racks themselves, which aides somewhat in maintaining the fuel subcritical.

2.2.4.2 Potential for Criticality in A BWR Spent Fuel Pool

First we will discuss the case of racks with boral poison cans. As noted in Section 2.2.2(1), crushing the BWR fuel assemblies would not significantly increase the k_{eff} of the fuel. For racks with boral poison, it seems inconceivable that any load which might fall on the spent fuel pool would separate the fuel from the poison cans and subsequently push the assemblies together to form a critical mass. Therefore it appears that postulated load drop events would not cause a criticality in a BWR spent fuel pool that uses boron plate can type racks.

For those spent fuel pools which depend on fuel separation to prevent criticality, that is type 8, 9, and 10 spent fuel racks, the drop of a heavy load which crushes the fuel rack would substantially raise k_{eff}. If several highly enriched fuels were stored in the region of the pool where the load is dropped, a criticality could result. Additionally, these types of fuel racks have separation in only one direction, and hence crushing the rack by a heavy load drop is more likely to result in an optimum configuration for increasing k_{eff} than racks which have separation in two directions, such as certain PWR racks.

2.2.4.3 Potential Criticality of BWR Reactor Core Due to A Heavy Load Drop

At least three heavy loads are carried over the core during refueling, namely the steam dryer (20-40 tons), the moisture separator (20-75 tons), and the vessel head (45-96 tons). These are carried over the core before and after refueling. These "before" and "after" cases will be discussed separately.

BWR Technical Specifications typically require that during refueling, with the most reactive control rod out of the core, k_{eff} shall be no greater than 0.997. During refueling, single control rods must be withdrawn, so that a k_{eff} of 0.997 actually may occur. However, before and after refueling, when the heavy loads are carried over the core, all control rods are inserted, and k_{eff} is no greater than about 0.96. The k_{eff} of 0.96 would be attained only after the core is reloaded. Before reloading, k_{eff} would be significantly less than this, probably no greater than 0.90.

After Reload: For this case, k_{eff} is no greater than about 0.96. Since the core is undermoderated, crushing the core will decrease k_{eff}. Thus, it appears there is no possibility of driving the core critical by crushing it in the after-reload case.

Before Reload: For this case, k_{eff} is no greater than about 0.90. The core is overmoderated, but crushing can only increase k_{eff} by a fraction of one percent. Thus it is not possible to drive the core critical by crushing it for the before-reload case.

The reactor is kept subcritical during refueling by the presence of cruciform control rods which are inserted from the bottom of the core. If a heavy load were to fall on the core and drive these rods out of the core (either before, during, or after reload), the core would immediately become supercritical. The potential for the load to drive the rods out of the core is small due to: the absorption of energy by deformation of fuel and control rods, more likely control rod failure modes than driving rods out of the core, and the catcher assembly below the control rod drives. However, information available to the staff was not sufficient to rule out this failure mechanism as a credible event. Guidelines contained in Section 5.1.4 and Appendix A require consideration of this mechanism.

2.2.5 Potential for Criticality of PWR Fuel

2.2.5.1 PWR Fuel Rack Designs

PWR fuel racks are maintained subcritical by employing three mechanisms: separation, steel neutron poison, and boron neutron poison plates. While the design analysis is performed assuming the spent fuel pool is filled with unborated water, in actuality the spent fuel water is borated to about 2000 ppm.

While the spent fuel pool boron concentration is typically not specified by a licensing requirement, the refueling water boron concentrations are delineated in all PWR Technical Specifications, and in all cases the spent fuel pool boron concentration will be very nearly or exactly the same as that of the refueling water. No credit is taken for the boron in demonstrating the subcriticality of the spent fuel pool under normal storage conditions. Required refueling water concentrations range from 1700 ppm to 2300 ppm boron concentration, depending on the plant. One notable exception to this is San Onofre Unit No. 1, for which the required boron concentration is 3750 ppm to 4300 ppm boron concentration.

2.2.5.2 Potential for Criticality in A PWR Spent Fuel Pool

It is apparent from Tables 2.2-2, 3, and 4 that under conditions where an accidental load drop crushes fuel from an offload core, it may be necessary to take credit for the borated spent fuel pool water to demonstrate subcriticality.

However, it appears from Tables 2.2-2, 3, and 4 that the 1700 ppm to 2300 ppm boron concentration normally maintained in the storage pools may not be adequate to guarantee subcriticality of a large array of fresh or partially burned fuel under load drop accident conditions. Subcriticality could be maintained by providing an increased boron concentration. It should be noted that the results shown in Tables 2.2-2, 3, and 4 were computed without taking credit for the neutron absorbing effects of the boral poison in the fuel racks or the structural aluminum or steel. Some racks rely heavily on the presence of these absorbers to maintain the fuel in a subcritical condition, while other racks rely principally on fuel separation to maintain the fuel subcritical. For those racks which rely principally on separation the values in Tables 2.2-2, 3, and 4 could be reasonably conservative for load drop accident conditions. For racks which contain a large amount of solid neutron absorber, under load drop accident conditions we can expect k_{eff} to be significantly lower than is indicated by the tables.

It should be noted that the above conclusions may be somewhat conservative because they are based on criticality calculations which assume an infinite array of highly enriched fuel. Generally, criticality calculations based on more realistic amounts of highly enriched fuel predict substantially lower required boron concentrations.

Additionally, it is possible that the drop of a heavy load could puncture the spent fuel pool liner. Normally, for PWR spent fuel pools, the borated refueling water can be pumped directly into the spent fuel pool to make up for leakage. If the leakage is so great that the reserve of borated water is exhausted, it would be necessary to fill the pool with unborated water from whatever source might be available. If such an accident also crushed the fuel rack to bring a large amount of highly enriched fuel together, then a criticality would ensue due to boron dilution. The potential for damage to the pool liner due to an accidental load drop is further discussed in Section 2.3.

2.2.5.3 Potential Criticality of A PWR Reactor Core Due to Heavy Load Drop

There are two load drop mechanisms which could cause a criticality of the core during refueling.

(1) The reactor vessel could be damaged and the borated refueling water backup exhausted, resulting in criticality due to boron dilution when makeup is supplied from an unborated source.

(2) The fuel could be compacted to a critical configuration in the 2000-ppm refueling water.

The potential for damage to the reactor vessel due to a load drop is discussed further in Section 2.3.

With respect to the potential for fuel to be crushed to a compact configuration, the data from Table 2.2-2 at 2.0 w/o U235 may be used to represent effects or a reload core. At 2000 ppm, the worst case analyzed here gives a whole core k_{eff} of 0.99. The control rods, which are in the core during refueling, and are not considered in Table 2.2-2, have a reactivity worth of about 10%. This would bring the k_{eff} down to about 0.89. Reload core average enrichments range from about 2.0 w/o U235 to 2.4 w/o U235. The difference of 0.4 w/o U235 has a reactivity worth of, at most, 0.06. This would bring K_{eff} up to approximately 0.95. Thus, assuming about a 5% uncertainty on the above analysis, the maximum k_{eff} of a reactor core under load drop conditions ranges from about

0.85 to 1.00. These figures do not rule out the possibility that a PWR core could become critical under the worst postulated load drop conditions. From Table 2.2-2 it seems that for the worst postulated load drop conditions, the core could be maintained subcritcal if boron concentration is maintained above 2500 ppm.

The above discussed the potential for criticality using an average core enrichment value. However, some higher enrichment assemblies are located in the core. It appears unlikely that crushing the more highly enriched assemblies located around the perimeter of the core (i.e., 3.5 w/o) could result in a localized criticality. The cores are always designed to produce a flat power distrbution, with the highly enriched assemblies placed in positions where the neutron leakage is high. In fact, because of this leakage, the highly enriched fresh fuel assemblies are normally the low power assemblies in the core. In Tables 2.2-2 and 2.2-3, it can be seen that for both the core average assemblies and the 3.5 w/o U-235 assemblies, optimal crushing increases the reactivity by about 2%. Thus, crushing is not expected to drive the local k_{eff} in the highly enriched assemblies significantly above the k_{eff} for the whole core.

2.2.6 Synopsis of Potential Criticality Situations

In the preceding paragraphs we have discussed the potential for criticality in the event of a heavy load drop in some detail. We will here give a summary of these findings.

For the following cases there appears to be no potential for criticality due to a heavy load drop:

(1) BWR spent fuel racks made up of a compact array of cans with boron plates;
(2) A BWR core, if it is postulated that the drop of a heavy load will not drive the control rods out of the core.
(3) A BWR or PWR spent fuel pool which contains only totally spent fuel.

A low potential for criticality exists in the following cases:

(1) A PWR reactor core if crushed due to a heavy load drop.
(2) PWR and BWR spent fuel racks which contain some neutron poison, but still depend on fuel separation to maintain the fuel subcritical. These would only be a problem if crushed and if they contained non-spent fuel. (Note: All PWR racks depend to some extent on separation, BWR racks may or may not depend on separation).

A high potential for criticality exists in the following cases:

(1) A BWR core if the heavy load were to drive the control rods out of the core, although the probability of this failure mode is considered small.
(2) PWR and BWR spent fuel racks which have no boral poison, but depend on fuel separation to maintain the fuel subcritical. This would only be a problem if they were crushed and if they contained non-spent fuel such as an off-load core.

2.3 Safe Shutdown Equipment

Loads may be carried in the area of safe shutdown equipment when the plant is operating. If these loads experienced uncontrolled movement or were dropped on safe shutdown equipment, the equipment may be unable to perform its function. The safe shutdown equipment includes items such as cabling, pumps, instrument racks, the control room, switchgear, and piping required to attain and maintain a safe shutdown. The loads could include various plant equipment, such as motors, pumps, valves, heat exchangers, switchgear, turbine equipment, and shielded shipping casks.

An example of the above is the handling of the shielded spent fuel cask in a BWR. The cask may be carried into the reactor building on a rail transporter or truck flatbed, and then unloaded and hoisted from grade elevation vertically 90 feet (27 m) to the refueling floor level. If a "two-blocking" event were to occur during this lift, a load drop in excess of 90 feet (27 m) could result. At some BWR's, this cask lift takes place over the suppression pool or a corner room which may contain residual heat removal (RHR), core spray, or reactor core isolation cooling (RCIC) pumps and equipment. It is generally acknowledged the intervening floor can not withstand a cask drop from such an elevation. The exact equipment that may be damaged in such a postulated event will depended on the specific plant layout and the location(s) where the drop is postulated to occur. If equipment from only one safety division, or safe shutdown path, is damaged, safe shutdown could generally be effected using equipment from the alternate or redundant shutdown path.

The potential for load drops to damage equipment from both safe shutdown paths will depend on plant layout and potential load paths. However, for most plants, redundant or dual equipment is already well separated due to other safety concerns such as protection against flooding, missiles, pipe whip, electrical faulting, and fire protection. Despite measures taken for these concerns, areas may still exist, particularly at older facilities, where redundant safe shutdown equipment are located in the same area or in separate areas but still within the path of a falling load.

Pool and Vessel Water Inventory

The reactor vessel head may weigh 55-165 tons (50,000-150,000 kg) and may be hoisted 20-50 feet (6-15 m) above the vessel flange. During the refueling operations, a drop of the reactor vessel head could impact the vessel flange. Since PWR vessels are typically supported by the vessel nozzles and refueling takes place when the vessel is cold and possibly below the NDTT (nil-ductility transition temperature; i.e., part of the vessel is in the brittle fracture range), a load drop having sufficient kinetic energy may potentially result in damage to the vessel nozzles or piping and cause loss of water inventory. Damage to only the nozzles or to piping would not in itself uncover the core; however, the possible lack of makeup water along with the boil-off due to decay heat could lead to uncovering of the core and subsequent fuel damage and release of fission products. Additionally, it appears that a postulated load drop of the vessel head could potentially damage the vessel itself in either BWRs or PWRs, and lead to uncovering the fuel if sufficient leakage resulted beyond water makeup capability.

Similarly, a load drop of the spent fuel cask in the spent fuel storage area could potentially result in damage to the spent fuel pool liner and structure, causing leakage of inventory. Excessive leakage beyond water makeup capability could lead to uncovering of the fuel with subsequent fuel damage and release of fission products.

3. SURVEY OF LICENSEE INFORMATION

In response to the staff's generic letter of June 12, 1978, licensees submitted various details related to load handling operations at their facilities. This information included:

(1) Identification of heavy loads and frequency of movement over or near spent fuel in the storage pool or fuel in the reactor;
(2) Identification of load paths normally followed in handling these heavy loads;
(3) Description of procedures developed relative to handling of heavy loads;
(4) Identification of certain analyses performed relative to a heavy load drop, such as cask drop analyses;
(5) Identification of certain design features which preclude a heavy load drop, such as a single-failure-proof crane;
(6) Identification of certain safety systems over which heavy loads may be handled; and
(7) Identification of conformance to Regulatory Guide 1.13, namely whether a single failure proof crane is provided, the spent fuel pool is designed to withstand a cask drop, or loads are precluded from being brought over the spent fuel pool by crane design.

The following sections provide a summary of the information submitted, indicating the types of heavy loads that are handled and the measures already in effect which prevent, or mitigate the consequences of, accidental load drops.

In general, information in this Section does not include plants in the Systematic Evaluation Program (SEP) because at the time this generic letter was sent to licensees, the staff was planning to have the SEP Program resolve this issue for SEP plants. The staff has since decided that implementation of guidelines contained in this report will be carried out for all operating plants, including SEP plants.

3.1 Heavy Loads

Information submitted by licensees was reviewed to identify the types of heavy loads that are handled, and their frequency of movements, over or in proximity to spent fuel or safe shutdown equipment. Table 3.1-1 provides a summary of typical loads handled, frequency of movement, and range of weight of the loads. The following are significant points to be noted from Table 3.1-1:

(1) PWR - Refueling Building

(a) There are a large number of heavy loads that may be carried in proximity to spent fuel, but for many plants, heavy loads need not be brought over spent fuel in the pool. This means that measures, such as the installation of mechanical stops or electrical interlocks, can be taken to preclude loads from being brought over spent fuel.
(b) Despite this, some loads such as the spent fuel shipping cask and pool gates may have to be brought over or near the spent fuel pool, although not over spent fuel.
(c) Certain plants may have to bring heavy loads over or in proximity to spent fuel.

TABLE 3.1-1

SURVEY OF HEAVY LOADS

	Area		Loads Handled	Over (O) or Only Proximity (P) to Fuel	Approx. Weight 1/	Frequency Handled
1.	PWR - Refueling Building	1.	Spent Fuel Shipping Cask	(P)	15-110 Tons (13-100,000 kg)	2/
		2.	Pool Divider Gates (some plants)	(P)	2 Tons (1800 kg)	2-4 x's (per refueling)
		3.	Fuel Transfer Canal Door	(P)	2 Tons (1800 kg)	2-4 x's (per refueling)
		4.	Missile Shields	(P)	4-20 Tons (4-19,000 kg)	2 x's (per refueling)
		5.	Irradiated Specimen Shipping Cask	(P)	3.5-12 Tons (3-11,000 kg)	Once per year to once per 10 years
		6.	Plant Equipment (some plants) (e.g., pumps, motors, valves, heat exchangers, etc.)	(O)	2-4 Tons (1800-3600 kg)	As required for modification or replacement
		7.	Spent resin, filter, or other radioactive material shipping casks	(P)	5-37 Tons (4500-33,000 kg)	~ 5 x's per year
		8.	New fuel shipping containers with fuel (usually 4 assemblies)	(P)	3-4 Tons (2700-3300 kg)	3/
		9.	Failed Fuel Container	(O)	1 Ton (900 kg)	Less than once per refueling

TABLE 3.1-1 (Continued)

Area		Loads Handled	Over (O) or Only Proximity (P) to Fuel	Approx. Weight 1/	Frequency Handled
1. (cont.)	10.	Fuel transfer carriage	(O) or (P)	1.5 Tons (1300 kg)	Only for maintenance or repair (~ once per 10 years)
	11.	Crane Load Block	(O)	4-10 Tons (4-9,000 kg)	1/
2. PWR - Containment Building	1.	Reactor Vessel Head	(O)	55-165 Tons (50-150,000 kg)	2 x's (per refueling)
	2.	Upper Internals	(O)	25-65 Tons (23-33,000 kg)	2 x's (per refueling)
	3.	In-Service Inspection Tool	(O)	4.5 Tons (4,000 kg)	Used at least once every three years
	4.	Reactor Coolant Pump	(P)	30-40 Tons (27-36,000 kg)	4-10 x's over life of plant
	5.	Missile Shields	(P)	10-20 Tons (9-18,000 kg)	2 x's (per refueling)
	6.	Crane Load Block	(O)	4-10 Tons (4-9,000 kg)	1/
3. BWR-- Reactor Building.	1.	Missile or Shield Plugs (6-12)	(P)	15-125 Tons (13-112,000 kg)	2 x's (per refueling)
	2.	Drywell Head	(P)	45-85 Tons (40-77,000 kg)	2 x's (per refueling)

TABLE 3.1-1 (Continued)

Area		Loads Handled	Over (O) or Only Proximity (P) to Fuel	Approx. Weight 1/	Frequency Handled
3. (cont.)	3.	Reactor Vessel Head	(O) (Over reactor)	45-96 Tons (40-86,000 kg)	2 x's (per refueling)
	4.	Steam Dryers 5/	(O) (Over reactor)	20-40 Tons (18-36,000 kg)	2 x's (per refueling)
	5.	Moisture Separators 5/	(O) (Over reactor)	20-75 Tons (18-68,000 kg)	2 x's (per refueling)
	6.	Spent Fuel Pool Gates	(O) (Over spent fuel pool)	2-6 Tons (1800-5,000 kg)	2 x's (per refueling)
	7.	Dryer/Separator Storage Pit Shield Plugs (some plants)	(P)	75 Tons (68,000 kg)	2 x's (per refueling)
	8.	Refueling Slot Plugs	(O) (Over spent fuel pool)	2-6 Tons (1800-5400 kg)	2 x's (per refueling)
	9.	Spent Fuel Shipping Cask	(O) (Over spent fuel pool)	15-110 Tons (14-99,000 kg)	4/
	10.	Vessel Service Platform	(O)	1-5 Tons (900-4500 kg)	5-10 x's (per refueling)
	11.	Waste and Debris Shipping Casks	(O) (Over reactor and/or spent fuel pool)	8-30 Tons (7-27,000 kg)	1-3 x's (per year)

TABLE 3.1-1 (Continued)

Area		Loads Handled	Over (O) or Only Proximity (P) to Fuel	Approx. Weight 1/	Frequency Handled
3. (cont.)	12.	Vessel Head Insulation	(P)	4-6 Tons (4-5,000 kg)	2 x's (per refueling)
	13.	Replacement Fuel Storage Racks for Spent Fuel	(O) (Over spent fuel)	8 Tons (7,000 kg)	On installation
	14.	Crane Load Block	(O)	4-10 Tons (4-9,000 kg)	1/
	15.	Plant Equipment	(O) (Over safety equip.)	1 Ton (900 kg)	
4. Other Plant Areas	1.	Spent Fuel Shipping Casks (some plants)	(O) (Over safety equipment)	15-110 Tons (14-99,000 kg)	2/, 4/
	2.	Turbine or other equipment in turbine building (some plants)	(O) (Over) safety equipment)	2-150 Tons (2-135,000 kg)	As required for equipment overhaul and replacement
	3.	Other plant equipment (pumps, motors, valves, heat exchangers, etc.)	(O) (Over safety equipment)	1-30 Tons (1-27,000 kg)	As required for equipment overhaul and replacement

TABLE 3.1-1

FOOTNOTES

1/ Listed weight for loads does not include weight of load block except where listed separately. The load block may add 4-10 tons (4,000 - 9,000 kg) to the weight of the dropped load. Because of this, the load block should be considered a heavy load even if it is not carrying a load, or is being used with a lighter load.

2/ These are presently not being used at most plants. However, once offsite waste repositories are established, casks will be used frequently for shipping spent fuel offsite. For a typical 1,000 MWe pressurized water reactor, spent fuel casks must be shipped offsite from 7 to 65 times per year depending on the size cask used. This is based on casks currently licensed for use in the United States.

3/ A typical 1,000 MWe power plant would usually require 16 or 17 new fuel containers (four fuel assemblies each) per year.

4/ These are presently not being used at most plants. However, once offsite waste repositories are established, casks will be used frequently for shipping spent fuel offsite. For a typical 1,000 MWe boiling water reactor, spent fuel casks must be shipped offsite from 12 to 125 times per year depending on the size cask used. This is based on casks currently licensed for use in the United States.

5/ Due to certain dimensional restrictions, for most BWR's it would not be possible to drop the dryers or moisture separators onto fuel in the reactor core.

(2) PWR - Containment Building

(a) All plants have to carry the reactor vessel head and vessel internals over the reactor vessel and core. Further, periodically other inspection or maintenance equipment will be handled over the reactor vessel.
(b) Certain other loads would normally be carried in proximity to the reactor, and if properly controlled, would not be brought over the reactor vessel or core.

(3) BWR - Reactor Building

(a) As in the PWR Containment Building, there are a number of heavy loads such as the vessel head, steam dryers, and moisture separators that would have to be moved over or in close proximity to the reactor vessel and core. Further, other inspection and maintenance equipment will be periodically handled over the reactor vessel.
(b) Certain heavy loads at most plants would have to be brought over the spent fuel pool but not over spent fuel if properly handled, such as refueling slot plugs, spent fuel pool gates, spent fuel shipping cask, and shielded radioactive waste and debris shipping casks.
(c) There are a number of loads that would normally be carried in proximity to the reactor vessel and spent fuel pool. If properly handled, these would not be moved over or in close proximity to fuel in the core or in the spent fuel pool.
(d) The reactor building contains equipment for safe shutdown systems. Heavy loads, such as the spent fuel shipping cask or plant equipment, may be carried over safe shutdown equipment in the reactor building.

(4) Other Plant Areas

There are a number of heavy loads which, if not properly controlled, could be brought over safe shutdown equipment.

Additionally, once offsite waste repositories are established, there will be frequent handling of spent fuel shipping casks. The frequency will depend on the size of the plant and the size shipping cask to be used. Because of this, the frequency of movement could vary from only five to over 100 shipments per year. It should be noted that if one of the larger casks were used by a certain facility, this would mean fewer offsite shipments; however, due to the larger size of the cask, the destructive forces developed by a postulated load drop may result in more damage to fuel assemblies as well as to safe shutdown equipment. The size of cask that may be used will also be limited by crane capacity, rail capacity serving the facility, and physical space available for movement of the cask.

3.2 Present Protection

The types of measures presently provided at operating plants to prevent or mitigate the consequences of accidental load drops varies considerably. The following sections describe the results of our survey of licensee information to identify such measures.

3.2.1 Technical Specifications

Most plants have technical specification requirements that pertain to the handling of heavy loads. Additionally, Standard Technical Specifications include a specification that prohibits travel of loads in excess of the nominal weight of a single fuel assembly over fuel assemblies in the storage pool. Twenty-seven plants (Table 3.2-1) do not have such a specification. However, fourteen of these plants include design features such as interlocks or single-failure-proof crane design to preclude a heavy load from dropping on spent fuel. Thus heavy loads could be carried over fuel assemblies in the storage pools of fourteen of these plants. This table includes plants in the Systematic Evaluation Program (SEP).

Several plants have a technical specification that prohibits movement of the spent fuel cask over the spent fuel pool. However, as noted in Table 3.2-1, such a specification does not prohibit other heavy loads from being carried over the spent fuel pool such as spent fuel pool gates, refueling slot plugs, waste and debris shipping casks, plant equipment, fuel transfer carriage, or just the crane load block without a load. Therefore, a specification that restricts movement of only the cask is not adequate to restrict other heavy loads from being carried over fuel assemblies in the storage pool.

3.2.2 Load Handling Procedures

Several plants have procedures related to the handling of heavy loads as shown in Table 3.2-2, for activities such as crane operation, refueling, handling of reactor components, or cask handling. However, a large number of plants apparently do not have such procedures.

Additionally, very few plants (3 out of 54) have procedures related to training of crane operators.

3.2.3 Crane Design

The survey included 54 non-SEP operating reactors comprised of 36 PWRs and 18 BWRs. Each of the 36 PWRs had its own individual polar crane to serve the reactor vessel within containment. By sharing the rectilinear cranes between two reactors at 10 PWR sites, the total number of PWR spent fuel cask handling rectilinear cranes was 26. Therefore, a total of 62 polar and rectilinear cranes were installed at the 36 PWR reactors for handling heavy loads over or near the reactor vessel or storage pool. Due to the difference in plant layout at the 18 BWR reactors, one rectilinear crane is capable of servicing both the reactor vessel and associated spent fuel pool. Further, at one BWR site, one rectilinear crane was able to meet the load handling requirements of three reactors and their associated spent fuel pools. At another BWR site the single rectilinear crane serves two reactors and associated spent fuel pools. Therefore, the 18 BWR reactors included in the review required a total of only 15 rectilinear cranes. At two of the reviewed BWR sites, the rectilinear crane also served other reactors not included in the survey, i.e., an SEP reactor and a reactor which was not included in the survey because it had not become operational at the time the questionnaire was sent out. Consequently, the 15 rectilinear cranes actually served the load handling requirements of 20 reactors and associated pools.

TABLE 3.2-1

TECHNICAL SPECIFICATIONS PROHIBITING HEAVY LOADS OVER STORAGE POOL

Plants that do not have a Technical Specification prohibiting handling of heavy loads over spent fuel (i.e., greater than a fuel assembly plus handling tool):

	NOTES
Big Rock Point	
Browns Ferry 1 - 3	single-failure-proof crane
Cooper	Limit switches to prevent travel over spent fuel
Dresden 1	
Dresden 2 and 3	single-failure-proof crane[1/]
Duane Arnold	
Ft. Calhoun	electric interlocks to prevent travel over spent fuel
FitzPatrick	
H. B. Robinson	single-failure-proof crane
Hatch 1	single-failure-proof crane
Haddam Neck	[1/]
Indian Point 2	[1/]
Millstone 1	[1/]
Monticello	single-failure-proof crane
Nine Mile Point	single-failure-proof crane
Oyster Creek	
Palisades	
Pilgrim	
Maine Yankee	
Quad Cities 1 and 2	single-failure-proof crane[1/]
Turkey Point 3 and 4	
Vermont Yankee	single-failure-proof crane[1/]

[1/] These facilities have technical specifications that prohibit handling of the spent fuel cask over the spent fuel pool, but do not prohibit heavy loads other than the cask from being brought over spent fuel.

TABLE 3.2-2

SURVEY OF PROCEDURES IN EFFECT RELATED TO CONTROL OF HEAVY LOADS

		PLANTS WHICH HAVE SUCH PROCEDURES[1]	PLANTS WHICH APPARENTLY DO NOT HAVE SUCH PROCEDURES[2]
1.	Procedures on crane operation	34	20
2.	Refueling procedures	30	24
3.	Movement of reactor components during or prior to refueling	27	27
4.	Cask handling operations	22	32
5.	Crane operator training	3	51

[1] In some cases procedures were not submitted, but were referenced by title and/or description.

[2] Information provided by licensees did not indicate that such procedures were in use.

Our review shows, as indicated in Table 3.2-3, that the intent of NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," was:

(1) not met by any of the 36 PWR polar cranes,
(2) not met in 24 of the 26 PWR rectilinear cranes (however, two licensees have committed to upgrade a total of two cranes serving three plants to meet single-failure-proof criteria),
(3) not met by 5 of the 15 BWR rectilinear cranes. (None of these 5 cranes served more than one reactor); and
(4) apparently met by the remaining 10 BWR rectilinear cranes. (At one site where the rectilinear crane served 3 reactors, the utility made crane modifications in order to make it single-failure-proof, however the staff has not evaluated these modifications.) Further there are three other sites where the crane serves more than one reactor and associated spent fuel pool, i.e., one case where two reactors in the survey share one crane, one case where one of the two reactors was not included in the survey because it was not operational when the questionnaire was sent out, and one case where the crane is shared between an SEP plant and a reactor included in the survey. Thus, the 10 single-failure-proof BWR cranes serve 13 reactors included in the survey, or 15 reactors counting the SEP plant and the plant which became operational recently.

3.2.4 Other Design Features

In addition to the use of a single-failure-proof crane, various other design features are used in operating plants as shown in Table 3.2-4. For example, the spent fuel pool areas of 51 of the 54 operating plants included in the survey are enclosed, exhausted through charcoal filters, and have ventilation systems that maintain the area at a lower pressure than the outside area so that leakage is into the area. Such a feature will reduce the quantity of airborne radioactive iodines released to the environment, as discussed in Section 2.1. These filters, however, are not effective in removing noble gases such as kryptons and xenons, which contribute to the whole body dose.

Approximately one-half of the operating plants have spent fuel pools that are designed for their assumed cask drop, so that leakage that may result from a cask drop is not sufficient to cause uncovering of spent fuel. Some of the spent fuel pools for the remaining operating plants may be able to withstand a cask drop even though they were not originally designed to have this capability.

Fewer than one-half of the operating PWR plants have rapid containment isolation on a high radiation signal (16 of 36 PWRs). Such a feature reduces offsite dose that may result from dropping of a heavy load on spent fuel although the size of the release will depend on the response time of this system. As a result of the staff's review of the containment purge system; the balance of the PWRs will have this capability, although it may not be automatic during the refueling mode. (See "Containment Building-PWR" Guidelines, Section 5.1.3.)

According to licensee responses, 39 of 54 operating plants have a spent fuel pool that is apparently designed to comply with Regulatory Guide 1.13. This means that such plants either have a single failure proof crane, a pool designed to withstand a cask drop without experienceing excessive leakage, electrical interlocks to prevent heavy loads from being carried over the spent fuel pool,

TABLE 3.2-3

CRANES SATISFYING INTENT OF NUREG 0554-SINGLE FAILURE PROOF CRANE[1/]

		Satisfy NUREG-0554		Do Not Satisfy NUREG-0554	
		# Plants	# Cranes	# Plants	# Cranes
PWR:	Containment Polar Crane	0	0	36	36
	Refueling Building Crane	2	2	34[2/]	24[2/]
BWR:	Reactor Building Crane	13	10	5[3/]	5[3/]

1/ Fifty-four (54) reactors were covered in the survey. Not included were eleven (11) SEP plants, two (2) plants indefinitely out of service, two (2) plants which were recently licensed, and Ft. St. Vrain which is a Gas-Cooled Reactor and the survey was limited to water reactors (there were 70 plants licensed to operate as of September 1979).

2/ However, licensees of three (3) of these plants have committed to upgrade the cranes used to handle the cask (affects two (2) cranes), although no date for completion of this upgrading has been established.

3/ However, one BWR licensee has committed to upgrade the reactor building crane.

TABLE 3.2-4

SURVEY OF DESIGN FEATURES RELATED TO CONTROL OF HEAVY LOADS[1/]

	PLANTS THAT HAVE THIS DESIGN FEATURE	PLANTS WITHOUT THIS DESIGN FEATURE
Containment (Spent Fuel Pool Area)	51	3
Charcoal Filters (Spent Fuel Pool Area)	51	3
Pool Designed for Their Assumed Cask Drop	26	28
Containment Isolation on High Radiation (PWR)	16[2/]	20
Compliance with Regulatory Guide 1.13	39	15

[1/] Exclusive of crane features covered by Table 3.2-3.

[2/] PWR plants that have rapid containment isolation on high radiation:

- D.C. Cook 1 & 2
- Ft. Calhoun
- Farley
- Maine Yankee
- North Anna 1
- Point Beach 1 & 2
- Prairie Island 1 & 2
- Rancho Seco
- Salem 1
- Turkey Point 3 & 4
- Zion 1 & 2

or cranes whose range of travel is such that heavy loads could not be brought over the spent fuel pool.

3.3 Load Drop Analyses

In addition to the design features described in Section 3.2, for some postulated load drops analyses have been performed to show that potential consequences are not unacceptable. All operating plants included in the survey have performed analyses related to a postulated fuel handling accident. However, few plants have performed other analyses, as summarized in Table 3.3-1.

Only five (5) plants have analyzed the potential consequences of a cask drop on spent fuel, in terms of offsite releases. Three (3) plants have performed analyses of the potential for a cask drop to cause criticality. Six (6) plants have analyzed the consequences of a reactor vessel internals or reactor head drop in terms of potential damage to the reactor vessel or to fuel in the core. In addition to these analyses twenty-six (26) plants have spent fuel pools that are designed and analyzed to withstand an assumed cask drop as listed in Table 3.2-4. In some cases the assumed cask weighs significantly less than the load rating of the overhead crane handling system.

TABLE 3.3-1

SURVEY OF LICENSEE ANALYSES RELATED TO CONTROL OF HEAVY LOADS

	PLANTS THAT HAVE PERFORMED THESE ANALYSES	PLANTS THAT HAVE NOT PERFORMED THESE ANALYSES
Cask drop damage to fuel	5	49[2/]
Fuel handling accident	54	0
Potential for drop to cause criticality	3[1/]	51
Plenum assembly or reactor head drop	6	18

[1/] These analyses only considered potential for a drop to cause criticality in the spent fuel pool, but not in the reactor.

[2/] However, some of these plants have separate cask loading areas and would not require carrying of the cask over the spent fuel pool.

4. REVIEW OF HISTORICAL DATA ON CRANE OPERATIONS

A variety of industrial type cranes and hoists have been in widespread use for many years to handle loads of greater than one ton (900 kg). They include chain falls, cable hoists (motor and mechanical - ratchet type), gantry cranes, cantilever gantry cranes, boom cranes (fixed and portable), rectilinear or overhead traveling cranes, cantilever wall cranes and polar cranes. As such, there is a broad base of experience with cranes and hoists, and a continual improvement in equipment to reduce the frequency of accidents. However, despite this broad base of experience, there is no single data bank available that can provide an accurate prediction of crane reliability against a load drop (i.e., probability of crane operation without dropping the load, per lift).

Typically, crane events that result in significant property damage or personnel injury are reported to insurance companies. However, not all events are reported and thus the completeness of such data is uncertain. Additionally, these statistics do not generally identify cause categories. Nonetheless, data is available from other sources that may be used to estimate bounds on the probability of a load drop, and to identify the principal causes of crane accidents. Useful data was obtained from the Occupational Safety and Health Administration, the Department of the Navy, and the NRC Licensee Event Report (LER) data file.

4.1 OSHA

The Occupational Safety and Health Administration (OSHA) collects some data on crane events (formerly through the Bureau of Labor Statistics). This data involves only those events reported to OSHA or obtained from insurance company records and therefore is not complete. This data only lists statistics on cause categories and does not include reports or descriptions of the events for further analysis. It does, however, present an interesting picture of the major causes of significant crane failures based on a large sample of industrial crane events. Table 4-1 provides a summary of the data collected by OSHA. A review of this data indicates that:

(1) The greatest contributor to crane accidents are crane operator errors (Categories B, C, and F) which accounted for 42% of all accidents. Improved operator training and qualification and use of operating procedures would reduce the frequency of operator errors.

(2) Improper rigging or inappropriate slings was the next greatest cause of crane accidents (Category A), with 34% of all accidents. Improved understanding of guidance on handling loads with slings and periodic inspection of the lifting devices would reduce the frequency of this type of failure. This would also reduce frequency of Category E events (overloading).

(3) Crane component failures (14% - 16%, if we consider a portion of Category G events) also led to many of the accidents. Improved maintenance and inspection would reduce the occurrence of Category D and G events (equipment failures, inadequate inspection and maintenance, and other various causes).

TABLE 4-1

DATA ON CRANE ACCIDENTS

FROM OSHA[1/] RECORDS

The following is a statistical summary of major crane accident causes based on an analysis of over 1,000 crane accidents involving damage to equipment:

	CAUSE CATEGORY	PERCENTAGE
A.	Loss of load due to poor rigging or slings	34
B.	Performing minor maintenance, inspection, or unrelated work while load is in motion	22
C.	Operating crane without authorization or proper signals	18
D.	Failure of defective boom, cable, or sheaves	14
E.	Failure due to overloading	4
F.	Handling load too near stationary equipment	2
G.	Other causes (including failures of control systems and inadequate inspection or maintenance)	6

[1/] Occupational Safety and Health Administration

4.2 Navy

A large number of cranes and hoists are used by the U.S. Navy in applications ranging from large shipyard cranes and on-board cranes used for cargo or weapons handling, to smaller cranes, hoists, and chainfalls used for miscellaneous load handling. The number of cranes and hoists in active use in any one year is approximately 2500 to 3000. The Navy personnel contacted during this study did not have access to an exact accounting of the number of cranes or hoists in use; however, based on the experience of these individuals in shipyard crane operations and shipboard cranes and hoists, it is believed that these are accurate limits on the number of cranes and hoists in use.

Similarly, an exact accounting of the number of lifts per year made by each crane was not available. The frequency of usage varies greatly; where a few cranes may be used only 5 or 10 times per year, others may be used almost 4 or 5 times per day (or approximately 1,000 to 1,250 times per year, excluding weekends and holidays). It is believed that an average number of lifts per crane is probably between 2 and 10 times per week (or approximately 100 to 500 lifts per year).

Using the above, we estimate that there are between 2.5×10^5 and 1.5×10^6 lifts per year by Navy cranes.

The Department of the Navy maintains several reporting systems that record crane events that involve material damage or personnel injury involving Navy cranes. The data system records causes of events, consequences and sequence of actions leading up to the event. The task group received computer print-out summaries of 466 crane events covering a period from February 1974 to October 1977. Most of these events involved minor personnel injuries. Of the 466 events, 75 events were ones that resulted in equipment damage, and of these 45 events were identified as load drops or potential load drops. Table 4-2 provides a summary of the principal causes of these 45 events.

The 45 load drop or potential load drop events occurred between February 1974 and October 1977. However, 31 of these events took place from January 1977 to October 1977 (10 months). (Only 14 of the 45 events occurred in 1974 through 1976. This is due to changes in the number of facilities and vessels covered in the reporting system.) The 31 events over 10 months is equivalent to approximately 37 events per year.

If we assume that somewhere between all and 1/2 of all events are being reported, then load drop or potential load drop events are occurring at a rate of between 37 and 74 events per year. If we then combine this event rate with the estimated number of lifts per year, we can obtain a conservative estimate of probability of load drop per lift:

$$P\ (\text{load drop}) = \frac{\text{No. of load drops per year}}{\text{No. of lifts per year}} = \frac{D}{L}$$

Where $37 \leq D \leq 74$, with a midpoint of 55.5; and $2.5 \times 10^5 \leq L \leq 1.5 \times 10^6$, with a midpoint of 8.75×10^5.

Therefore, $2.5 \times 10^5 \leq P\ (\text{load drop}) \leq 3 \times 10^4$ with a midpoint of 2.7×10^5.

TABLE 4-2

CAUSES OF CRANE ACCIDENTS

DEPARTMENT OF THE NAVY

(February 1974 - October 1977)

CAUSE CATEGORY	NO. OF LOAD DROP OR POTENTIAL LOAD DROP EVENTS	% OF TOTAL	TOTAL NUMBER OF CRANE EVENTS RESULTING IN EQUIPMENT DAMAGE	% OF TOTAL
1. Crane Failure	10	23%	17	23%
(Design Error)	(1)	(2.3%)	(2)	(3%)
(Maintenance Personnel)	(2)	(4.6%)	(2)	(3%)
(Crane Component Failure)	(7)	(16.3%)	(13)	(17%)
2. Crane Operator Error	30 1/	70%	54	73%
(Distraction/ Inattention)	(11)	(26%)	(24)	(32%)
(Inadequate Training)	(8)	(18%)	(13)	(19%)
(Failed To Follow Proper Precautions/ Procedures)	(11)	(26%)	(17)	(23%)
3. Rigging	3	7%	3	4%
(Rigger)	(3)	(7%)	(3)	(4%)
(Rigging)	(0)	(0)	(0)	(0)
Totals:	43		74	

1/ 15 (50%) of these events occurred when the crane or hoist was left in the "raise" mode or inadvertently raised to limit.

A review of the Navy data for identification of principal causes of load drops or potential load drops was performed, and is summarized in Table 4-2. This summary shows that:

(1) Operator errors are by far the greatest contributors to load drop events; more thorough operator training and operating procedures would reduce the frequency of operator error.
(2) Of all the events directly caused by operator error, 15 of the 30, or 50% were the result of inattention by leaving the hoist or crane in the "raise" mode or inadvertently raising the lower load block up to or near the upper load block ("two-blocking" or nearly "two-blocking"). Application of single-failure-proof features as well as improved operator training and procedures would greatly reduce the frequency of these events.
(3) The next greatest contributor was due to random material failures. Closer adherences to the prescribed inspection frequency and more thorough inspections as well as application of single-failure-proof features would greatly reduce the frequency of random material failures.

In terms of applicability to nuclear facilities, there are four areas that can be compared: operator training and qualification, procedural controls, complexity of equipment operation, and design of equipment. Navy crane operators receive some initial training and are provided manuals on proper crane operation. It does not appear that the training required or the procedures used by the Navy are as detailed as what is called for by the guidelines contained in Section 5.1 of this report. Many of the cranes in use by the Navy are similar in method of operation and design as cranes used at nuclear power plants, i.e., overhead gantry or rectilinear cranes. However, the Navy also uses a large number of boom type cranes. Boom type cranes are more susceptible to failure due to operator action in moving the boom, or positioning the boom properly without overextending it. It is therefore expected that actual failure rates of cranes in nuclear facilities would be lower than the estimates arrived at above using data from the Navy, once the guidelines of Section 5.1 are implemented.

4.3 Licensee Event Reports (LERs)

During this period a total of 34 crane incidents were reported. Two of these incidents occurred during the plant construction period and the remainder occurred during normal plant operating periods, including refueling periods.

The incidents can be broken down as shown in Table 4-3.

These events involved a partial drop of the reactor vessel head without impacting any object (15 inches); a 3 inch drop of the reactor vessel head on the vessel flange; drop of a core barrel and internals (6 feet); damage to fuel during refueling; damage to nearby equipment by crane hook; dropping of a polar crane hook; crane overload; damage to new fuel storage racks; and damage to a control room roof deck.

It should be noted that no personnel injuries were reported in connection with or as a result of these incidents. It should also be noted that no release of radioactivity occurred as a result of any of the incidents, including those involving damage to fuel elements. In one case two fuel rods were bent without breach of the cladding; other cases involved damage to fuel element couplings.

TABLE 4-3

SURVEY OF CRANE

LER EVENTS

(July 1969 - July 1979)

Cause Category	No. of Events	Percentage
A. Failure during plant construction phase	2	6
B. Failure due to design or fabrication errors	9	26
C. Failure due to lack of adequate inspection	2	6
D. Failure due to operator error or lack of training	8	24
E. Failure due to random mechanical component failures	5	15
F. Failure due to random failures of control system components	3	9
G. Events due to lack of operating procedures	4	12
H. Events due to crane overloading (including load hangup)	1	3
Total	34	

Load drops that resulted in damage to concrete structures occurred only in two Category A incidents. Another heavy load drop occurred when an upper hoist travel limit switch failed on a polar crane resulting in the wire rope rubbing against a beam and "two-blocking" with subsequent wire rope failure and dropping of the empty load block and hook to the floor (part of Category F). A test load drop occurred as a result of improper hook selection; no damage incurred.

The greatest number of incidents were Category B events which included malfunctioning of components due to improper selection or installation and improper fabrication procedures such as questionable welding of structural members.

Operator errors, Category D, also occurred quite frequently. Improved training may reduce this frequency; however, influence of the human element in this case may continue to be high because operators for these cranes only operate this equipment occasionally and, therefore, may not obtain the intimate familiarity with the crane operation that in most of the cases could have prevented the incident from happening.

The likelihood of the incidents in Categories B, C, E, F and H (19 of 34 incidents) may have been reduced if the crane had been designed, constructed and tested in accordance with the provisions of NUREG-0554. Many operator errors, including overloading (12 of 34 incidents) may be reduced by directing more attention to administrative controls and operator training.

The Category H (overloading) incident did not result in any adverse condition; an overload sensing device stopped the hoisting motion before the ultimate strength of the wire rope had been exceeded.

5. GUIDELINES FOR CONTROL OF HEAVY LOADS

Our evaluation of the information provided by licensees indicates that existing measures at operating plants to control the handling of heavy loads cover certain of the potential problem areas, but do not adequately cover the major causes of load handling accidents. These major causes include operator errors, rigging failures, lack of adequate inspection and inadequate procedures. The measures in effect vary from plant to plant, with some having detailed procedures while others do not, some have performed analyses of certain postulated load drops, certain plants have single-failure-proof cranes, some PWR's have rapid containment isolation on high radiation, and many plants have technical specifications that prohibit handling of heavy loads or a spent fuel cask over the spent fuel pool. To provide adequate measures that minimize the occurrence of the principal causes of load handling accidents and to provide an adequate level of defense-in-depth for handling of heavy loads near spent fuel and safe shutdown systems, the measures in effect should be upgraded.

5.1 Recommended Guidelines

The following sections describe various alternative approaches which provide acceptable measures for the control of heavy loads. The <u>objectives</u> of these guidelines are to assure that either (1) the potential for a load drop is extremely small, or (2) for each area addressed, the following evaluation criteria are satisfied:

I. Releases of radioactive material that may result from damage to spent fuel based on calculations involving accidental dropping of a postulated heavy load produce doses that are well within 10 CFR Part 100 limits of 300 rem thyroid, 25 rem whole body (analyses should show that doses are equal to or less than 1/4 of Part 100 limits);

II. Damage to fuel and fuel storage racks based on calculations involving accidental dropping of a postulated heavy load does not result in a configuration of the fuel such that k_{eff} is larger than 0.95;

III. Damage to the reactor vessel or the spent fuel pool based on calculations of damage following accidental dropping of a postulated heavy load is limited so as not to result in water leakage that could uncover the fuel, (makeup water provided to overcome leakage should be from a borated source of adequate concentration if the water being lost is borated); and

IV. Damage to equipment in redundant or dual safe shutdown paths, based on calculations assuming the accidental dropping of a postulated heavy load, will be limited so as not to result in loss of required safe shutdown functions.

After reviewing the historical data available on crane operations, identifying the principal causes of load drops, and considering the type and frequency of load handling operations at nuclear power plants, the NRC staff has developed an overall philosophy that provides a defense-in-depth approach for controlling the handling of heavy loads. This philosophy encompasses an intent to prevent as well as mitigate the consequences of postulated accidental load drops. The following summarizes this defense-in-depth approach:

(1) Provide sufficient operator training, handling system design, load handling instructions, and equipment inspection to assure reliable operation of the handling system; and

(2) Define safe load travel paths through procedures and operator training so that to the extent practical heavy loads avoid being carried over or near irradiated fuel or safe shutdown equipment; and

(3) Provide mechanical stops or electrical interlocks to prevent movement of heavy loads over irradiated fuel or in proximity to equipment associated with redundant shutdown paths.

Certain alternative measures may be taken to compensate for deficiencies in (2) and (3) above, such as the inability to prevent a particular heavy load from being brought over spent fuel (e.g., reactor vessel head). These alternative measures can include: increasing crane reliability by providing dual load paths for certain components, increased safety factors, and increased inspection as discussed in Section 5.1.6 of this report; restricting crane operations in the spent fuel pool area (PWRs) until fuel has decayed so that off-site releases would be sufficiently low if fuel were damaged; or analyzing the effects of postulated load drops to show that consequences are within acceptable limits. Even if one of these alternative measures is selected, (1) and (2) above should still be satisfied to provide maximum practical defense-in-depth.

The following sections provide guidelines on how the above defense-in-depth approach may be satisfied for various plant areas. Fault trees and associated probabilities were developed and used as described in Bases for Guidelines, Section 5.2 of this report, to evaluate the adequacy of these guidelines and to assure a consistent level of protection for the various areas.

5.1.1 General

All plants have overhead handling systems that are used to handle heavy loads in the area of the reactor vessel or spent fuel in the spent fuel pool. Additionally, loads may be handled in other areas where their accidental drop may damage safe shutdown systems. Accordingly, all plants should satisfy each of the following for handling heavy loads that could be brought in proximity to or over safe shutdown equipment or irradiated fuel in the spent fuel pool area and in containment (PWRs), in the reactor building (BWRs), and in other plant areas.

(1) Safe load paths should be defined for the movement of heavy loads to minimize the potential for heavy loads, if dropped, to impact irradiated fuel in the reactor vessel and in the spent fuel pool, or to impact safe shutdown equipment. The path should follow, to the extent practical, structural floor members, beams, etc., such that if the load is dropped, the structure is more likely to withstand the impact. These load paths should be defined in procedures, shown on equipment layout drawings, and clearly marked on the floor in the area where the load is to be handled. Deviations from defined load paths should require written alternative procedures approved by the plant safety review committee.

(2) Procedures should be developed to cover load handling operations for heavy loads that are or could be handled over or in proximity to irradiated fuel or safe shutdown equipment. At a minimum, procedures should cover handling of those loads listed in Table 3-1 of this report. These procedures should include: identification of required equipment; inspections and acceptance criteria required before movement of load; the steps and proper sequence to be followed in handling the load; defining the safe load path; and other special precautions.

(3) Crane operators should be trained, qualified and conduct themselves in accordance with Chapter 2-3 of ANSI B30.2-1976, "Overhead and Gantry Cranes."

(4) Special lifting devices should satisfy the guidelines of ANSI N14.6-1978, "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 pounds (4500 kg) or More for Nuclear Materials." This standard should apply to all special lifting devices which carry heavy loads in areas as defined above. For operating plants certain inspections and load tests may be accepted in lieu of certain material requirements in the standard. In addition, the stress design factor stated in Section 3.2.1.1 of ANSI N14.6 should be based on the combined maximum static and dynamic loads that could be imparted on the handling device based on characteristics of the crane which will be used.* This is in lieu of the guideline in Section 3.2.1.1 of ANSI N14.6 which bases the stress design factor on only the weight (static load) of the load and of the intervening components of the special handling device.

(5) Lifting devices that are not specially designed should be installed and used in accordance with the guidelines of ANSI B30.9-1971, "Slings." However, in selecting the proper sling, the load used should be the sum of the static and maximum dynamic load.* The rating identified on the sling should be in terms of the "static load" which produces the maximum static and dynamic load. Where this restricts slings to use on only certain cranes, the slings should be clearly marked as to the cranes with which they may be used.

(6) The crane should be inspected, tested, and maintained in accordance with Chapter 2-2 of ANSI B30.2-1976, "Overhead and Gantry Cranes," with the exception that tests and inspections should be performed prior to use where it is not practical to meet the frequencies of ANSI B30.2 for periodic inspection and test, or where frequency of crane use is less than the specified inspection and test frequency (e.g., the polar crane inside a PWR containment may only be used every 12 to 18 months during refueling operations, and is generally not accessible during power operation. ANSI B30.2, however, calls for certain inspections to be performed daily or monthly. For such cranes having limited usage, the inspections, tests, and maintenance should be performed prior to their use.)

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*For the purpose of selecting the proper sling, loads imposed by the SSE need not be included in the dynamic loads imposed on the sling or lifting device.

(7) The crane should be designed to meet the applicable criteria and guidelines of Chapter 2-1 of ANSI B30.2-1976, "Overhead and Gantry Cranes" and of CMAA-70, "Specifications for Electric Overhead Travelling Cranes." An alternative to a specification in ANSI B30.2 or CMAA-70 may be accepted in lieu of specific compliance if the intent of the specification is satisfied.

5.1.2 Spent Fuel Pool Area - PWR

Many PWR's require that the spent fuel shipping cask be placed in the spent fuel pool for loading. Additionally, other heavy loads may be carried over or near the spent fuel pool using the overhead crane, including plant equipment, rad-waste shipping casks, the damaged fuel container and replacement fuel storage racks. Additionally, certain crane failures could cause the crane lower load block to be dropped, and therefore this should also be considered as a heavy load. The fuel handling crane is used for moving fuel and is generally not used for handling of heavy loads. To provide assurance that the evaluation criteria of Section 5.1 are met for load handling operations in the spent fuel pool area, in addition to satisfying the general guidelines of Section 5.1.1, one of the following should be satisfied:

(1) The overhead crane and associated lifting devices used for handling heavy loads in the spent fuel pool area should satisfy the single-failure-proof guidelines of Section 5.1.6 of this report.

<u>OR</u>

(2) Each of the following is provided:

(a) Mechanical stops or electrical interlocks should be provided that prevent movement of the overhead crane load block over or within 15 feet horizontal (4.5 meters) of the spent fuel pool. These mechanical stops or electrical interlocks should not be bypassed when the pool contains "hot" spent fuel, and should not be bypassed without approval from the shift supervisor (or other designated plant management personnel). The mechanical stops and electrical interlocks should be verified to be in place and operational prior to placing "hot" spent fuel in the pool.

(b) The mechanical stops or electrical interlocks of 5.1.2(2)(a) above should also not be bypassed unless an analysis has demonstrated that damage due to postulated load drops would not result in criticality or cause leakage that could uncover the fuel.

(c) To preclude rolling if dropped, the cask should not be carried at a height higher than necessary and in no case more than six (6) inches (15 cm) above the operating floor level of the refueling building or other components and structures along the path of travel.

(d) Mechanical stops or electrical interlocks should be provided to preclude crane travel from areas where a postulated load drop could damage equipment from redundant or alternate safe shutdown paths.

(e) Analyses should conform to the guidelines of Appendix A.

<u>OR</u>

(3) Each of the following are provided (Note: This alternative is simlar to (a) above, except it allows movement of a heavy load, such as a cask, into the pool while it contains "hot" spent fuel if the pool is large enough to maintain wide separation between the load and the "hot" spent fuel.):

(a) "Hot" spent fuel should be concentrated in one location in the spent fuel pool that is separated as much as possible from load paths.

(b) Mechanical stops or electrical interlocks should be provided to prevent movement of the overhead crane load block over or within 25 feet horizontal (7.5 m) of the "hot" spent fuel. To the extent practical, loads should be moved over load paths that avoid the spent fuel pool and kept at least 25 feet (7.5 m) from the "hot" spent fuel unless necessary. When it is necessary to bring loads within 25 feet of the restricted region, these mechanical stops or electrical interlocks should not be bypassed unless the spent fuel has decayed sufficiently as shown in Table 2.1-1 and 2.1-2, or unless the total inventory of gap activity for fuel within the protected area would result in offsite doses less than ¼ of 10 CFR Part 100 if released, and such bypassing should require the approval from the shift supervisor (or other designated plant management individual). The mechanical stops or electrical interlocks should be verified to be in place and operational prior to placing "hot" spent fuel in the pool.

(c) Mechanical stops or electrical interlocks should be provided to restrict crane travel from areas where a postulated load drop could damage equipment from redundant or alternate safe shutdown paths. Analyses have demonstrated that a postulated load drop in any location not restricted by electrical interlocks or mechanical stops would not cause damage that could result in criticality, cause leakage that could uncover the fuel, or cause loss of safe shutdown equipment.

(d) To preclude rolling, if dropped, the cask should not be carried at a height higher than necessary and in no case more than six (6) inches (15 cm) above the operating floor level of the refueling building or other components and structures along the path of travel.

(e) Analyses should conform to the guidelines of Appendix A.

<u>OR</u>

(4) The effects of drops of heavy loads should be analyzed and shown to satisfy the evaluation criteria of Section 5.1 of this report. These analyses should conform to the guidelines of Appendix A.

5.1.3 Containment Building - PWR

PWR containment buildings contain a polar crane that is used for removing and reinstalling shield plugs, the reactor vessel head, upper vessel internals, and on occasion, other heavy equipment such as the reactor coolant pump, the reactor vessel inspection platform, and the cask used for damaged fuel. Additionally the crane load block may be moved over fuel in the reactor when handling smaller loads or no load at all. Due to the weight of the load block alone, this should also be considered as a heavy load. To provide assurance that the criteria of Section 5.1 are met for load handling operations in the containment building, in addition to satisfying the general guidelines of Section 5.1.1, one of the following should be satisfied:

(1) The crane and associated lifting devices used for handling heavy loads in the containment building should satisfy the single-failure-proof guidelines of Section 5.1.6 of this report.

<u>OR</u>

(2) Rapid containment isolation is provided with prompt automatic actuation on high radiation so that postulated releases are within limits of evaluation Criterion I of Section 5.1 taking into account delay times in detection and actuation; and analyses have been performed to show that evaluation criteria II, III, and IV of Section 5.1 are satisfied for postulated load drops in this area. These analyses should conform to the guidelines of Appendix A.

OR

(3) The effects of drops of heavy loads should be analyzed and shown to satisfy the evaluation criteria of Section 5.1. Loads analyzed should include the following: reactor vessel head; upper vessel internals; vessel inspection platform; cask for damaged fuel; irradiated sample cask; reactor coolant pump; crane load block; and any other heavy loads brought over or near the reactor vessel or other equipment required for continued decay heat removal and maintaining shutdown. In this analysis, credit may be taken for containment isolation if such is provided; however analyses should establish adequate detection and isolation time. Additionally, the analysis should conform to the guidelines of Appendix A.

5.1.4 Reactor Building - BWR

The reactor building in BWRs typically contains the reactor vessel and spent fuel pool, as well as various safety-related equipment.

The reactor building overhead crane may be used in many day-to-day operations such as moving various shielded shipping casks or handling plant equipment related to maintenance or modification activities. The crane is also used during refueling operations for removal and reinstallation of shield plugs, drywell head, reactor vessel head, steam dryers and separators, and refueling canal plugs and gates. The crane would also be used subsequent to refueling for handling of the spent fuel shipping cask. This cask may be lifted as high as 100 feet (30 m) above the grade elevation at which the cask is brought into the reactor building. Additionally the overhead crane's load block may be moved over fuel in the reactor or over the spent fuel pool when handling smaller loads or no load at all. Due to the weight of the load block alone, this should also be considered as a heavy load.

To assure that the evaluation criteria of Section 5.1 are satisfied one of the following should be met in addition to satisfying the general guidelines of Section 5.1.1:

(1) The reactor building crane, and associated lifting devices used for handling the above heavy loads, should satisfy the single-failure-proof guidelines of Section 5.1.6 of this report.

OR

(2) The effects of heavy load drops in the reactor building should be analyzed to show that the evaluation criteria of Section 5.1 are satisfied. The loads analyzed should include: shield plugs, drywell head, reactor vessel head; steam dryers and separators; refueling canal plugs and gates; shielded spent fuel shipping casks; vessel inspection platform; and any other heavy loads that may be brought over or near safe shutdown equipment as well as fuel in the reactor vessel or the spent fuel pool. Credit may be taken in this analysis for operation of the Standby Gas

Treatment System if facility technical specifications require its operation during periods when the load being analyzed would be handled. The analysis should also conform to the guidelines of Appendix A.

5.1.5 Other Areas

In other plant areas, loads may be handled which, if dropped in a certain location, may damage safe shutdown equipment. Although this is not a concern at all plants, loads that may damage safe shutdown equipment at some plants include the spent fuel shipping cask, turbine generator parts in the turbine building, and plant equipment such as pumps, motors, valves, heat exchangers, and switchgear. Some of these loads may be less than the weight of a fuel assembly with its handling tool, but may be sufficient to damage safe shutdown equipment.

(1) If safe shutdown equipment are beneath or directly adjacent to a potential travel load path of overhead handling systems, (i.e., a path not restricted by limits of crane travel or by mechanical stops or electrical interlocks) one of the following should be satisfied in addition to satisfying the general guidelines of Section 5.1.1:

(a) The crane and associated lifting devices should conform to the single-failure-proof guidelines of Section 5.1.6 of this report;

OR

(b) If the load drop could impair the operation of equipment or cabling associated with redundant or dual safe shutdown paths, mechanical stops or electrical interlocks should be provided to prevent movement of loads in proximity to these redundant or dual safe shutdown equipment (In this case credit should not be taken for intervening floors unless justified by analysis).

OR

(c) The effects of load drops have been analyzed and the results indicate that damage to safe shutdown equipment would not preclude operation of sufficient equipment to achieve safe shutdown. Analyses should conform to the guidelines of Appendix A, as applicable.

(2) Where the safe shutdown equipment has a ceiling separating it from an overhead handling system, an alternative to Section 5.1.5(1) above would be to show by analysis that the largest postulated load handled by the handling system would not penetrate the ceiling or cause spalling that could cause failure of the safe shutdown equipment.

5.1.6 Single-Failure-Proof Handling Systems

For certain areas, to meet the guidelines of Sections 5.1.2, 5.1.3, 5.1.4, or 5.1.5, the alternative of upgrading the crane and lifting devices may be chosen. The purpose of the upgrading is to improve the reliability of the handling system through increased factors of safety and through redundancy or duality in certain active components. NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," provides guidance for design, fabrication, installation, and testing of new cranes that are of a high reliability design. For operating plants, Appendix C to this report, "Modification of Existing Cranes," provides guidelines on implementation of NUREG-0554 for operating plants and plants under construction.

Section 5.1.1 of this report provides certain guidance on slings and special handling devices. Where the alternative is chosen of upgrading the handling system to be "single-failure-proof", then steps beyond the general guidelines of Section 5.1.1 should be taken.

Therefore, the following additional guidelines should be met where the alternative of upgrading handling system reliability is chosen:

(1) Lifting Devices:

(a) Special lifting devices that are used for heavy loads in the area where the crane is to be upgraded should meet ANSI N14.6 1978, "Standard For Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More For Nuclear Materials," as specified in Section 5.1.1(4) of this report except that the handling device should also comply with Section 6 of ANSI N14.6-1978. If only a single lifting device is provided instead of dual devices, the special lifting device should have twice the design safety factor as required to satisfy the guidelines of Section 5.1.1(4). However, loads that have been evaluated and shown to satisfy the evaluation criteria of Section 5.1 need not have lifting devices that also comply with Section 6 of ANSI N14.6.

(b) Lifting devices that are not specially designed and that are used for handling heavy loads in the area where the crane is to be upgraded should meet ANSI B30.9 - 1971, "Slings" as specified in Section 5.1.1(5) of this report, except that one of the following should also be satisfied unless the effects of a drop of the particular load have been analyzed and shown to satisfy the evaluation criteria of Section 5.1:

(i) Provide dual or redundant slings or lifting devices such that a single component failure or malfunction in the sling will not result in uncontrolled lowering of the load;

OR

(ii) In selecting the proper sling, the load used should be twice what is called for in meeting Section 5.1.1(5) of this report.

(2) New cranes should be designed to meet NUREG-0554, "Single-Failure-Proof Cranes For Nuclear Power Plants." For operating plants or plants under construction, the crane should be upgraded in accordance with the implementation guidelines of Appendix C of this report.

(3) Interfacing lift points such as lifting lugs or cask trunions should also meet one of the following for heavy loads handled in the area where the crane is to be upgraded unless the effects of a drop of the particular load have been evaluated and shown to satisfy the evaluation criteria of Section 5.1:

(a) Provide redundancy or duality such that a single lift point failure will not result in uncontrolled lowering of the load; lift points should have a design safety factor with respect to ultimate strength of five (5) times the maximum combined concurrent static and dynamic load after taking the single lift point failure.

OR

(b) A non-redundant or non-dual lift point system should have a design safety factor of ten (10) times the maximum combined concurrent static and dynamic load.

5.2 Bases for Guidelines

The review of crane historical data in Section 4 of this report indicates the principal causes of load drop or equipment damage accidents involving cranes. The guidelines in the preceding section are intended to give appropriate attention to these causes so that the potential for accidental load drops that impact irradiated fuel or safe shutdown equipment is reduced. These guidelines are further aimed at assuring that the objectives of Section 5.1 are met.

As noted in Section 5.1, these guidelines were developed to provide a defense-in-depth approach to controlling the handling of heavy loads near spent fuel and safe shutdown equipment. Section 5.1.1 provides general guidelines for safe load handling that will reduce the potential for load drops, even though a single-failure-proof crane is provided or evaluations show that the consequences of postulated load drops are within established limits. This is consistent with the defense-in-depth philosophy used for other safety concerns.

General Guidelines

The review of crane historical data indicated the need to give special attention to operator training, guidance on rigging and lifting devices, crane inspection and well defined procedures, which were principal causes of load drop or handling accidents. Additionally, ANSI B30.2 "Overhead and Gantry Cranes," ANSI B30.9, "Slings," and ANSI B30.10, "Hooks" note the following: "The use of cranes, derricks, hoists, jacks and slings is subject to certain hazards that cannot be met by mechanical means, but only by the exercise of intelligence, care and common sense. It is therefore essential to have competent and careful operators, physically and mentally fit, thoroughly trained to the safe operation of the equipment and the handling of the loads. Serious hazards are overloading, dropping or slipping of the load caused by improper hitching or slinging, obstruction to the free passage of the load, or using equipment for a purpose for which it was not intended or designed." Section 5.1.1 guidelines address each of these areas. Safe load paths should be defined that keep heavy loads, to the extent practical, away from irradiated fuel and safe shutdown equipment. Procedures should be developed to assure that required actions and precautions related to load handling are well understood by the operator; this will tend to reduce the occurrence of operator errors. Crane operator training is required to assure operator familiarity with equipment and procedures to further reduce the occurrence of crane operator errors. Guidelines on lifting devices and slings assure adequate safety margins on these components, and their proper installation and use. Inspection, testing, and maintenance of the crane is called for to assure that load bearing components are in proper working order, that worn or damaged components are identified and replaced, and that design safety margins are maintained. The reduced inspection frequency from the ANSI B30.2-1976 guidelines is acceptable for cranes not used frequently, because the B30.2 guidelines are based on expected wear when cranes are in more frequent use. Conformance to the design guidelines of ANSI B30.2 and CMAA-70 is recommended so that cranes whose failure could cause a drop of a heavy load on safe shutdown equipment, fuel in the core, or fuel in the spent fuel pool meet the minimum industrial specifications.

Area Specific Guidelines

Sections 5.1.2, 5.1.3, and 5.1.4 provide various alternatives for specific areas that should be met in addition to conformance with the general guidelines of Section 5.1.1. These alternatives assure that either the potential for a load drop is further reduced (e.g., single-failure-proof crane and lifting devices) or that the potential consequences of postulated load drops are within acceptable limits. Certain criteria contained in these alternatives were based on staff generic evaluations, such as the potential for criticality (Section 2.2), or safe decay times for spent fuel (Section 2.1). However, for certain postulated load drops, generic evaluations could not be performed since these would tend to be plant specific, such as vessel head drop or cask drop analyses. Thus, an alternative may require analyses of these postulated load drops on a plant specific basis if that alternative is selected.

As noted above, certain alternatives in Sections 5.1.2, 5.1.3 and 5.1.4 require specific minimum decay times for spent fuel. The task group's evaluation of offsite release potential due to load drop accidents shows that adequate decay times for spent fuel (i.e., 42 days for PWRs and 44 days for BWRs that exhaust through charcoal filters, and 74 days for PWRs that do not exhaust through charcoal filters) will assure that offsite releases, due to dropping of postulated heavy loads on fuel that has been subcritical for the required decay time, will not cause doses that approach 10 CFR Part 100 limits. Limits used by the task group were 1/4 of Part 100 limits, or 75 rem thyroid and 6.25 rem whole body, for postulated load drop accidents. This assures that dose limits are kept reasonably low for such postulated events that may occur more frequently than the most severe design basis events.

Additionally, certain alternatives call for a neutronics analysis to determine the potential for a postulated load drop to cause criticality. In Section 2.2 it was shown that in a number of cases a significant potential for criticality under load drop conditions exists, and for those cases a neutronics analysis is necessary. A summary of the likelihood for criticality under various load drop conditions is given in Section 2.2.6.

Certain alternatives call for electrical interlocks to keep loads away from the spent fuel pool or away from "hot" spent fuel. Such interlocks are in addition to the definition of safe load paths. These interlocks need not be single-failure-proof, as a failure of these would have to be accompanied by operator error in failing to follow the prescribed load path and a concurrent failure of the handling system when over the spent fuel and when the pool contains "hot" spent fuel. The adequacy of this alternative is evaluated by the fault-tree evaluation in this section.

The 15-foot (4.5 m) separation limit on the mechanical stops or electrical interlocks called for in guideline 5.1.2(2)(a) is based on the maximum dimensions of a cask to assure that in a cask tip, the cask center of gravity will not go beyond the edge of the spent fuel pool. The 25 foot (7.5 m) separation limit on the mechanical stops or electrical interlocks called for in guideline 5.1.2(3)(b) is based on the area containing spent fuel that could be impacted if a cask carried over the pool were to tip when dropped.

Fault Trees

To further evaluate the adequacy of the guidelines of Section 5.1 and to assure a general equivalency between alternatives, fault trees were developed and probabilities for various faults derived or estimated. These trees represent the situation after the guidelines are met.

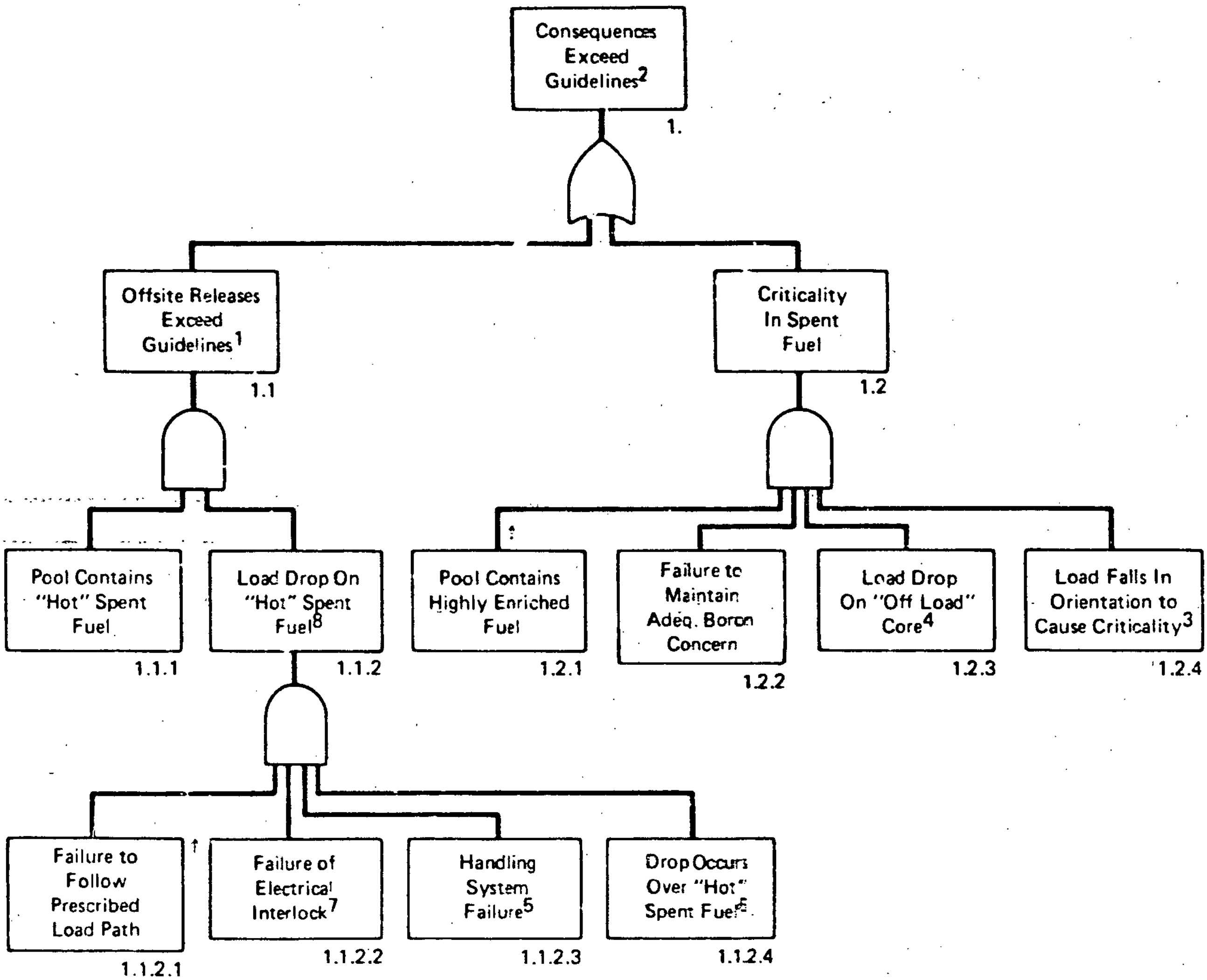
Some alternatives rely on analyses to demonstrate that postulated events would not cause unacceptable consequences, and thus do not lend themselves readily to analysis by fault tree. However, other alternatives in the guidelines rely to a significant degree on probabilities and thus lend themselves to evaluation using fault tree techniques. These may be generalized as three situations:

(1) Loads handled near spent fuel or "hot" fuel, primarily in the spent fuel pool, where reliance is placed on safe load path procedures, electrical interlocks, maintaining adequate boron concentration, and handling system reliability. This is depicted by Figure 5.2-1.
(2) Loads handled over the spent fuel pool where reliance is placed on electrical interlocks, procedures to segregate "hot" spent fuel, handling system reliability, safe load path procedures, and maintaining adequate boron concentration. This is depicted by Figure 5.2-2.
(3) Loads handled by a single-failure-proof crane and lifting devices where reliance is placed on increased handling system reliability through increased safety factors and dual or redundant components, and on safe load paths for loads that are not required to be brought over spent fuel. The single-failure-proof crane may be required to handle loads over fuel (reactor vessel head, vessel internals, etc), but would more frequently be used carrying loads near fuel in the reactor or the spent fuel pool. This is depicted by the fault tree in Figure 5.2-3, sheets 1 and 2.

Probabilities were derived or estimated for the various faults in Figures 5.2-1, 5.2-2 and 5.2-3 as described in Appendix B to this report. Table 5.2-1 summarizes the results of the evaluation of these fault trees using the probabilities of Appendix B. This evaluation shows that:

(1) The likelihood for unacceptable consequences in terms of excessive releases of gap activity or potential for criticality due to accidental dropping of postulated heavy loads after implementation of the guidelines of Section 5.1 is very low; and
(2) The potential for unacceptable consequences is comparable for any of the alternatives evaluated by fault trees, indicating the relative equivalency between alternatives.

These fault trees and the probability estimates received a brief review by the Probabilistic Analysis Staff of RES (NRC). Their comments were incorporated into this report.



1 Evaluation Criterion I of Section 5.1
2 Evaluation Criteria of Section 5.1
3 Given That Events 1.2.1, 1.2.2, and 1.2.3 Occur
4 Given That Event 1.2.1 Occurs
5 Given That Events 1.1.2.1 and 1.1.2.2 Occur
6 Given That Events 1.1.1.1, 1.1.2.1, 1.1.2.2, and 1.1.2.3 Occur
7 Given That Event 1.1.2.1 Occurs
8 Given That Event 1.1.1 Occurs

FIGURE 5.2-1 FAULT TREE FOR LOADS HANDLED NEAR SPENT FUEL POOL

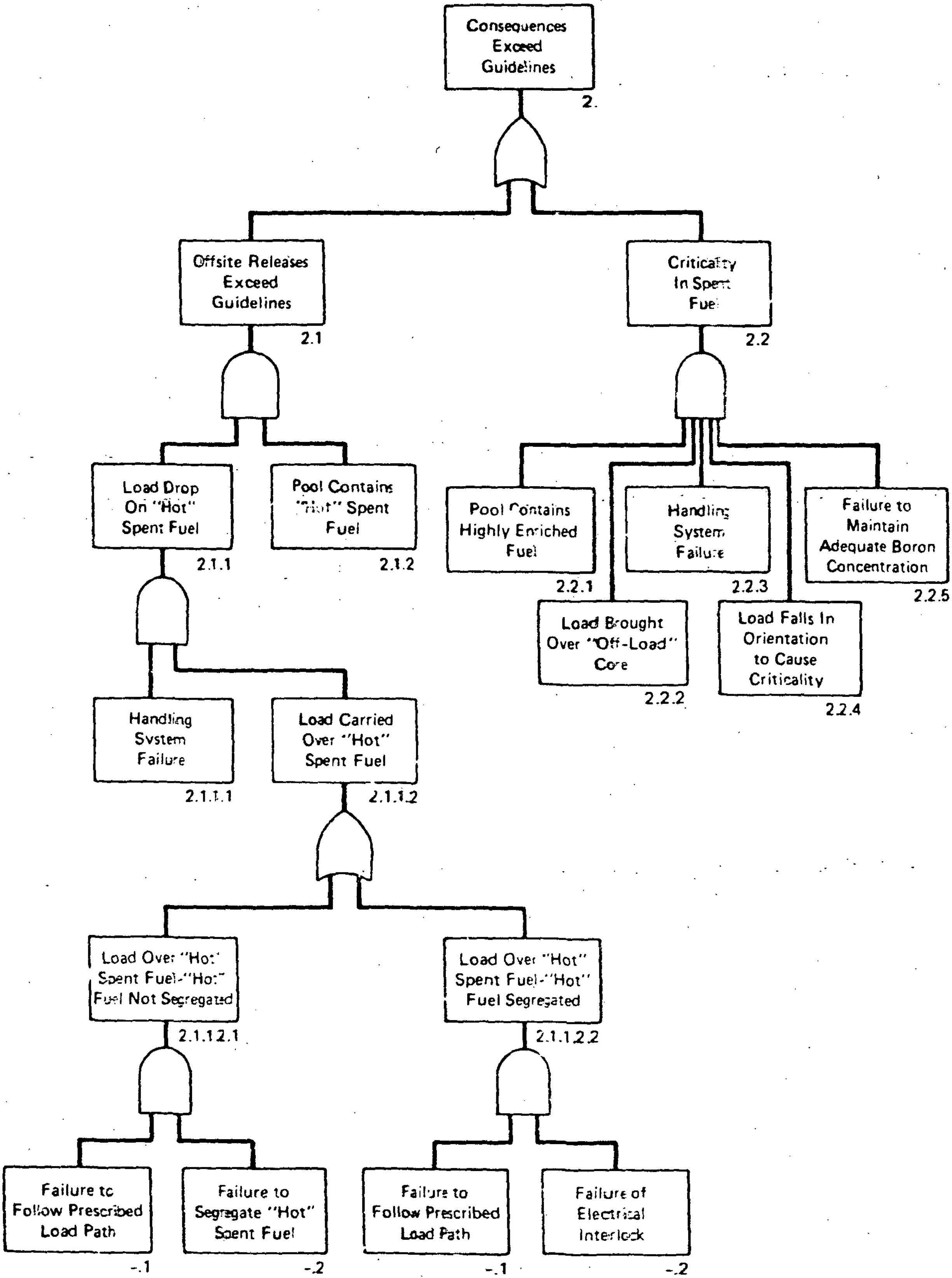
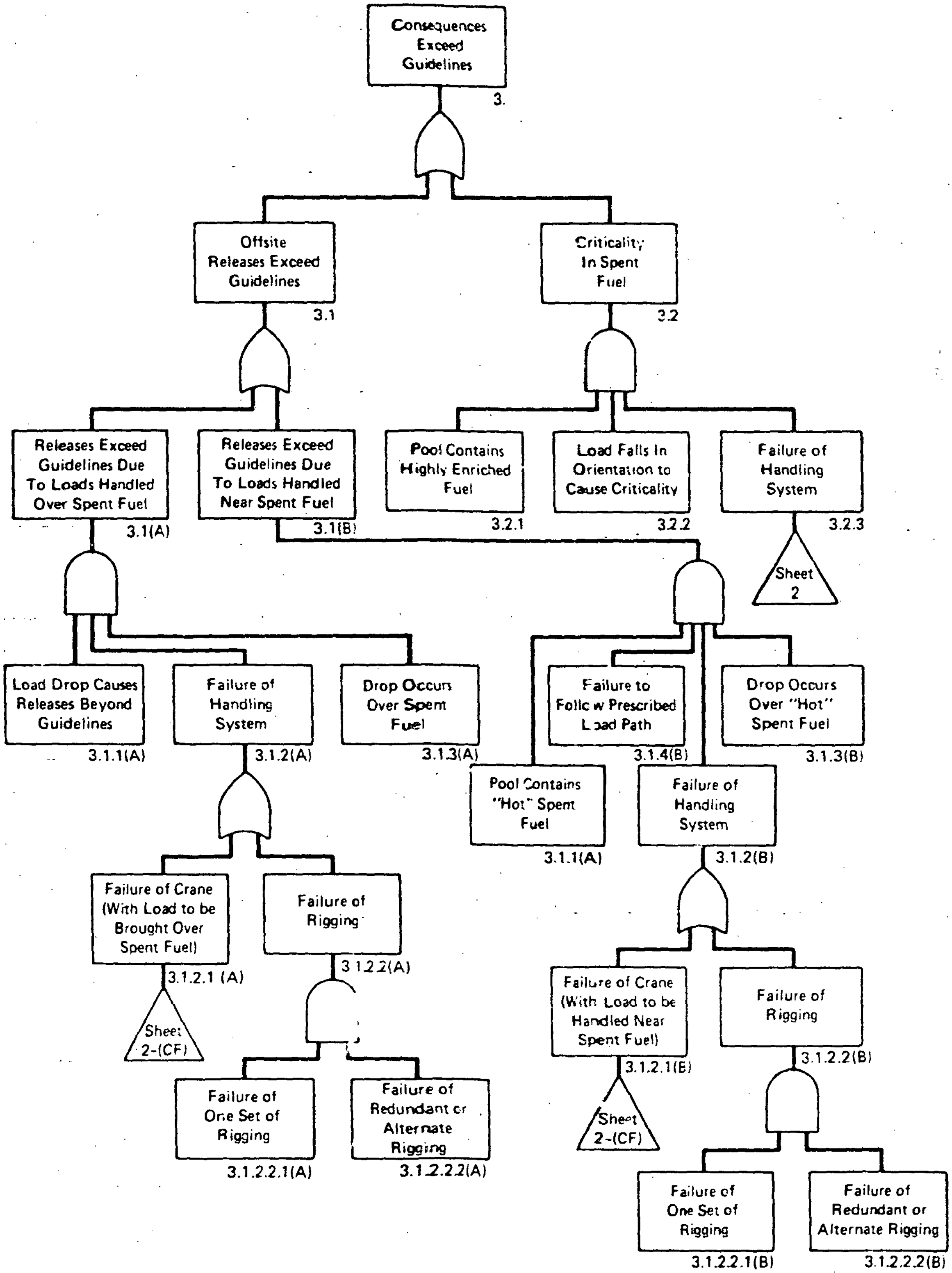
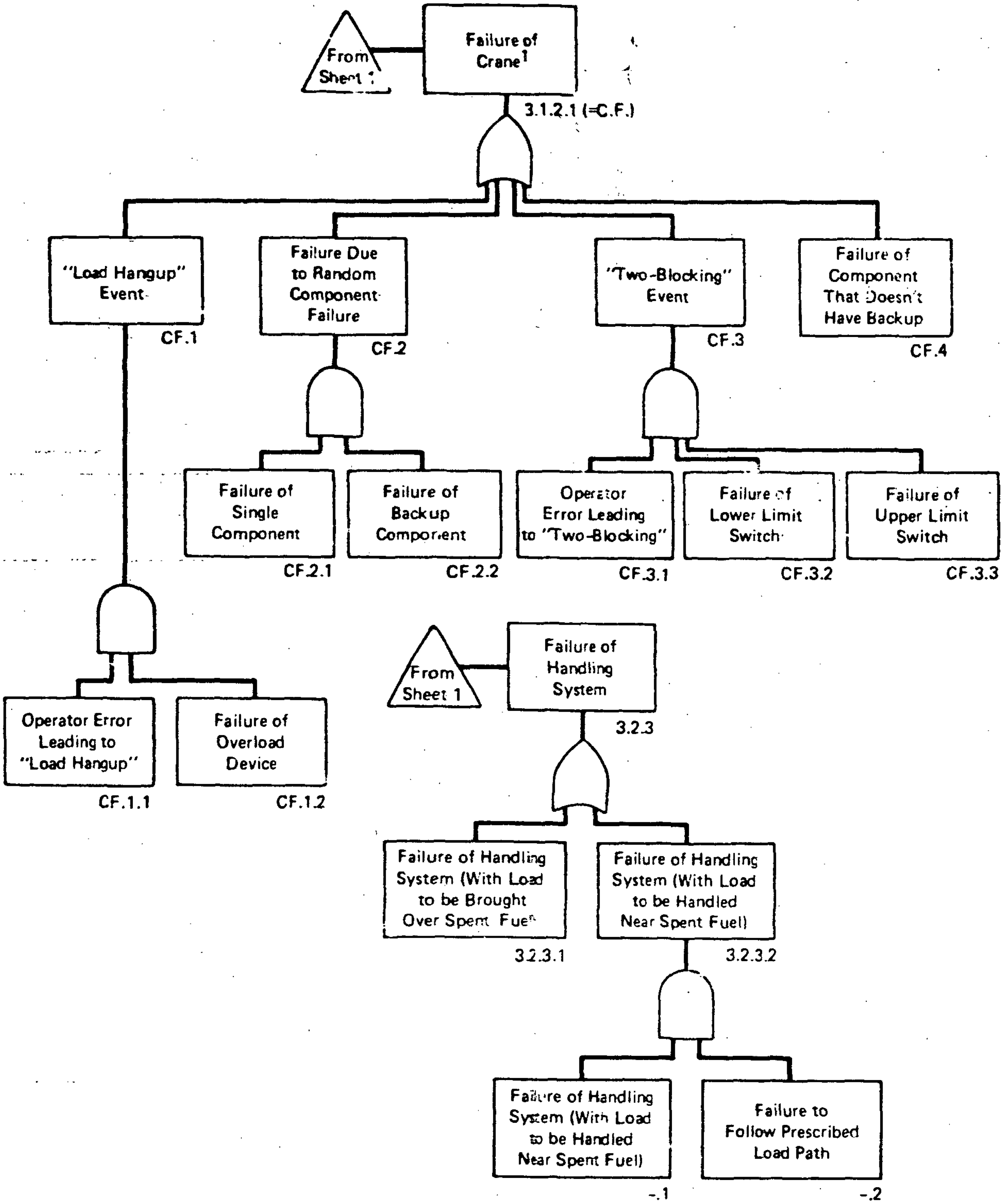


FIGURE 5.2-2 FAULT TREE FOR LOADS HANDLED OVER SPENT FUEL POOL



1/ For Some Loads Safe Load Paths are Defined That Keep Loads Away From Spent Fuel Even if a Single Failure Proof Crane is Provided. This is Depicted by Branch 3.1(B). Certain Other Loads Must be Carried Over Spent Fuel; This is Depicted by Branch 3.1(A) of This Fault Tree.

FIGURE 5.2-3 FAULT TREE IF A SINGLE FAILURE PROOF HANDLING SYSTEMS IS USED[1]



[1] This Fault Tree May be Used for Either Branch (A) or Branch (B), Where Branch (A) Covers Those Loads that Must be Handled Over Spent Fuel, Such as the Reactor Vessel Head, and Branch (B) Covers Loads that Would Normally Only be Handled Near Spent Fuel.

FIGURE 5.2-3 SHEET 2

TABLE 5.2-1

SUMMARY OF EVENT PROBABILITIES FOR HANDLING OF HEAVY LOADS[1/]

	LOWER BOUND	MEDIAN	UPPER BOUND
1. Loads Handled Near Spent Fuel or Reactor (Figure 5.2-1):			
P(Consequences Exceed Guidelines)[2/]	2×10^{-10}	2×10^{-8}	4×10^{-6}
P(Offsite Releases Exceed Guidelines)	2×10^{-10}	2×10^{-8}	4×10^{-6}
P(Criticality In Spent Fuel)		Negligible	
2. Loads Handled Over Spent Fuel Pool (Figure 5.2-2):			
P(Consequences Exceed Guidelines)[2/]	2×10^{-8}	7×10^{-7}	3×10^{-5}
P(Offsite Releases Exceed Guidelines)	2×10^{-8}	7×10^{-7}	3×10^{-5}
P(Criticality in Spent Fuel)	Negligible		3×10^{-6}
3. Loads Handled With A Single Failure Proof Crane (Figure 5.2-3):			
P(Consequences Exceed Guidelines)[2/]	3×10^{-9}	2×10^{-7}	10^{-5}
P(Offsite Releases Exceed Guidelines)	3×10^{-9}	2×10^{-7}	10^{-5}
P(Criticality in Spent Fuel)	Negligible		10^{-6}

1/ These are given in terms of probability of event per reactor year.

2/ Guidelines referred to here are the evaluation criteria of Section 5.1.

5.3 Safety Evaluation

As noted previously, our evaluation of the information provided by licensees indicated that existing measures at operating plants to control the handling of heavy loads did not adequately cover all areas or the major causes of load handling accidents, and that these major causes include operator errors, rigging failures, lack of adequate inspection and inadequate procedures. The measures in effect vary from plant to plant, with some having detailed procedures while others do not, some have performed analyses of certain postulated load drops, some plants have single-failure-proof cranes, some PWRs have rapid containment isolation on high radiation, and many plants have technical specifications or other licensing restrictions that prohibit handling of heavy loads or a spent fuel cask over the spent fuel pool. To provide measures that assure an adequate level of defense-in-depth for handling of heavy loads near spent fuel and safe shutdown systems, the measures in effect should be upgraded to satisfy the guidelines of Section 5.1.

Our review of regulatory criteria and guidelines that are used in the licensing of new plants indicates that many of the elements of the guidelines of Section 5.1 of this report are already included in standard review plans and regulatory guides. However, certain measures called for in the guidelines of Section 5.1 are presently not included in these standard review plans and regulatory guides but are appropriate for new plants, such as establishment of safe load paths, training of crane operators, crane inspection and testing, and potential for a load drop to cause criticality. These standard review plans and regulatory guides could be upgraded to include those guidelines of Section 5.1 that are appropriate for new plants.

As noted in Section 5.2, the guidelines of Section 5.1 provide a defense-in-depth approach to assure the safe handling of heavy loads. In addition the fault trees and probability estimates further demonstrate the adequacy of these guidelines. In summary, we find that upon completion of modifications, required analyses, and changes to procedures to satisfy the guidelines of Section 5.1, adequate measures will be established to:

(1) Reduce the potential for accidental dropping of heavy loads;
(2) Reduce the potential for a heavy load to impact on spent fuel or safe shutdown equipment, should a drop occur; and
(3) Provide further protection by either employing a single-failure-proof handling system, or implementing measures and performing analyses such that the calculated potential effects of postulated load drops satisfy the following:

    (a) Releases of radioactive material that may result from damage to spent fuel involving the dropping of a postulated heavy load produce doses that are 1/4 of 10 CFR Part 100 limits, i.e., less than 75 rem thyroid and 6.25 rem whole body;
    (b) damage to fuel in the core or spent fuel pool storage racks involving the dropping of a postulated heavy load does not result in a configuration of the fuel such that k_{eff} approaches or is larger than 0.95;
    (c) damage to the reactor vessel or the spent fuel pool involving the dropping of a postulated heavy load is limited so as not to result in leakage that could uncover the fuel; and

(d) damage to equipment from redundant safe shutdown paths involving the dropping of a postulated heavy load will be limited so as not to result in loss of required safe shutdown functions.

For those guidelines that rely on probabilities of events being small, the fault trees discussed in Section 5.2 demonstrate that the probability of unacceptable consequences is very low.

Interim Protection

At present there is little handling of spent fuel shipping casks. Once offsite waste repositories are established, the frequency of cask handling will increase significantly. To provide reasonable assurance that no casks or other heavy loads are handled over the spent fuel pool until final implementation of the guidelines of Section 5.1, technical specifications should be upgraded to prohibit handling of heavy loads over the spent fuel pool. As noted previously, many plants already have such a specification.

Definition of safe load paths, development of load handling procedures, training of crane operators, and inspection of cranes are procedural or administrative measures that can be accomplished in a relatively short time period and need not be delayed for completion of evaluations and modifications to satisfy the guidelines of Section 5.1. Implementation of these measures will further reduce the potential for accidental load drops to impact on fuel in the core or spent fuel pool. Additionally a special review of procedures, equipment, and personnel for handling loads over the core provides greater assurance of the safe handling of such loads.

We therefore find that to assure safe handling of heavy loads in the interim period until measures at operating plants are upgraded to satisfy the guidelines of Section 5.1, implementation of the following measures should be initiated:

(1) Licenses for all operating reactors not having a single-failure-proof overhead crane in the fuel storage pool area should be revised to include a specification comparable to Standard Technical Specification 3.9.7, "Crane Travel - Spent Fuel Storage Pool Building' for PWR's and Standard Technical Specification 3.9.6.2, "Crane Travel," for BWR's, to prohibit handling of heavy loads over fuel in the storage pool until implementation of measures which satisfy the guidelines of Section 5.1 (see Table 3.2-1).
(2) Safe load paths should be defined per the guidelines of Section 5.1.1(1);
(3) Procedures should be developed and implemented per the guidelines of Section 5.1.1(2);
(4) Crane operators should be trained, qualified and conduct themselves per the guidelines of Section 5.1.1(3); and
(5) Cranes should be inspected, tested, and maintained in accordance with the guidelines of Section 5.1.1(6).
(6) In addition to the above, special attention should be given to procedures, equipment, and personnel for the handling of heavy loads over the core, such as vessel internals or vessel inspection tools. This special review should include the following for these loads: (1) review of procedures for installation of rigging or lifting devices and movement of the load to assure that sufficient detail is provided and that instructions are clear and concise; (2) visual inspections of load bearing components of

cranes, slings, and special lifting devices to identify flaws or deficiencies that could lead to failure of the component; (3) appropriate repair and replacement of defective components; and (4) verify that the crane operators have been properly trained and are familiar with specific procedures used in handling these loads, e.g., hand signals, conduct of operations, and content of procedures.

Implementation of the above measures will provide reasonable assurance that handling of heavy loads will be performed in a safe manner, until final implementation of the guidelines of Section 5.1. Additionally, operating experience has shown that no heavy load drop accidents damaging irradiated fuel have occurred in over 400 reactor years of operating experience. The above recommended interim actions will further reduce the potential for accidental load drops to damage irradiated fuel. On the basis of previous operating experience and the additional interim measures, we find that continued power operation and refueling operations until final implementation of the guidelines of Section 5.1 does not present undue risk to the health and safety of the public.

6. RESOLUTION OF THE ISSUE

The following is a summary of those recommended actions that should be taken to resolve the concern over the handling of heavy loads near irradiated fuel, or safety related equipment.

6.1 Implementation of Guidelines - Operating Plants

It is recommended that a program be initiated to assure that the guidelines of Section 5.1 of this report are implemented at operating facilities. This program should include the following:

(1) Transmittal of a generic letter to licensees requesting details describing how the guidelines of Section 5.1 will be met, including required modifications and results of analyses;

(2) Availability of NRC staff personnel or outside technical assistance to evaluate information submitted by licensees in response to the above generic letter. Such outside technical assistance would require expertise in various areas, including crane design and operation, structural and mechanical analyses, accident analysis (radiological doses), criticality calculations, and plant refueling operations and administrative controls. The assistance required on specific plants may vary, depending on the alternatives selected; however, expertise in each of the above areas will be required for the program.

    As noted in Section 3 many operating plants already meet certain of the guidelines, such as single-failure-proof cranes at 15 plants, and thus the impact of satisfying the guidelines will be reduced.

(3) A safety evaluation should be prepared on each facility providing the basis for the conclusion that load handling will be carried out in a safe manner at that facility.

6.2 Interim Actions

To provide adequate assurance that handling of heavy loads will be performed safely in the interim period until final implementation of changes required to satisfy the guidelines of Section 5.1, it is recommended that the interim measures described in Section 5.3 be implemented.

6.3 Changes to SRPs and RGs

At new facilities certain of the problems that are present in older operating facilities do not exist. For example, many operating plants require placement of the shipping cask in the spent fuel pool for loading with spent fuel. Such an operation makes fuel in the storage pool and storage pool integrity more susceptible to damage due to an accidental load drop. However, new facilities provide a separate cask loading pit that is well separated from the spent fuel pool, and spent fuel assemblies are individually transported through a canal from the spent fuel pool for loading in the cask. Because many of the potential load handling problems that exist at present operating facilities are not present in new facilities, certain of the guidelines in Section 5.1 are not appropriate for new reactors.

To incorporate the guidelines from Section 5.1 that are appropriate for new reactors, the following changes to Standard Review Plans and Regulatory Guides should be made:

(1) SRP 9.1.2 - "Spent Fuel Storage" - Rev. 1

Recommended Change:
Add a statement in the acceptance criteria of this or some other SRP that includes the following: The spent fuel pool ventilation system should be designed to maintain at least a -1/8 inch (3 mm) water gauge negative pressure during fuel handling operations and should automatically switch to ventilation thru Engineered Safety Feature (ESF) grade filters in the event of a high radiation signal. (Note a revision to SRP 9.1.3 is already in process that will include the above criterion. With the revision to SRP 9.1.3, the above change to SRP 9.1.2 is not required.)

(2) SRP 9.1.4 - "Fuel Handling System" - Rev. 1

(a) Recommended Change:
Add the following: "The ICSB will also verify that the instrument response time capability of the airborne activity monitoring system satisfies the required response time identified by AAB to prevent the release of activity through isolation valves or to assure that ventilation flow is switched to an ESF grade filter system prior to release to the environment. The ICSB should advise AAB of any reactor system which does not meet either of these functions both in the containment building and in the spent fuel storage facility".

(b) Recommended Change:
Add to the lead-in paragraph of Part A to ASB-BTP 9-1 a reference to guidelines on selection and use of rigging and lifting devices and minimum crane requirements (Similar to that in Section 5.1.1 of this report). These guidelines would apply to any area where heavy loads could be handled near spent fuel, fuel in the reactor, or safe shutdown equipment.

Basis:
These measures together with other actions taken to meet options 1, 2, or 3 in Part A of ASB-BTP 9-1 and those listed in items 3 and 4 below will provide defense-in-depth for load handling operations. The measures identified above will also assure that proper attention is given to the major contributors to load handling accidents to reduce the occurrence of such events.

(c) Recommended Change:
Option 1 of Part A to ASB-BTP 9-1 should include a statement that both electrical interlocks and mechanical stops are provided to keep the cask from being transported over the spent fuel pool.

Basis:
This option presently relies on electrical interlocks and mechanical stops to keep the cask away from the spent fuel pool, however the SRP does not include this detail. The above change only documents the criteria that are presently being used.

(d) Recommended Change:
Add to the statement on the evaluation criteria of option 3 of Part A to ASB BTP9-1 that the consequences of a postulated load drop should also not result in <u>criticality</u> or <u>excessive leakage</u> that could uncover the fuel.

Basis:
These are potential consequences that should be considered in the analyses.

(e) Recommended Change:
Guidelines should be added on rigging, special lifting devices, and interfacing lift points to be used with a single-failure-proof crane. These guidelines should be similar to Section 5.1.6 of this report.

Basis:
Guidelines in ANSI standards on slings and special lifting devices are available and should be used to assure the reliability of these components. Additionally, guidance should be provided on interfacing lift points since failure of these could potentially result in a load drop.

(3) <u>SRP 13.1.3</u> "Qualifications of Nuclear Plant Personnel"

Recommended Change:
Add to the acceptance criteria a statement that crane operators that may handle heavy loads over or near fuel in the reactor, fuel in the storage pool, or safe shutdown equipment are qualified and conduct themseves in accordance with the guidelines of ANSI B30.2-1976 (Chapter 2-3) "Overhead and Gantry Cranes."

(4) <u>R.G. 1.33</u> - "Quality Assurance Prgram Requirements (Operation)," Rev. 2

(a) Recommended Change: Add to section 2 of Appendix A to this guide that general plant operating procedures should also be developed for the following: (1) handling of heavy loads near fuel in the reactor, fuel in the storage pool, or safe shutdown equipment; and (2) identification of safe load paths;

(b) Recommended Change: Add to Section 1 of Appendix A to this guide that administrative procedures should also be developed for qualification, training, and conduct of crane operators.

(5) Regulatory Guide 1.13, Revision 1 - "Spent Fuel Storage Facility Design Basis"

Recommended Change:
Regulatory Guide 1.13, Regulatory Position C.5 should be changed to delete the three options listed and to list option 1 of ASB 9-1, with recommended changes (see changes to SRP 9.1.4).

Basis:
ASB 9-1 offers only the option of keeping the cask away from the spent fuel pool using electrical interlocks and mechanical stops. This makes the regulatory guidance consistent with the review criteria.

(6) Regulatory Guide 1.XX -

Recommended Change:
A regulatory guide should be developed endorsing ANSI N14.6, 1978 "Standard For Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More For Nuclear Materials". ANSI N14.6 provides guidelines that are not peculiar to lifting devices for shipping containers; these guidelines would be applicable to any special lifting device. Therefore, the regulatory guide that is developed to endorse ANSI N14.6 should endorse this standard for use in designing and using special lifting devices that handle heavy loads over or near spent fuel, fuel in the core, or safe shutdown equipment.

Basis:
Such a regulatory guide will facilitate use of industry guidelines in evaluating the adequacy to special lifting devices in the licensing review process, and will provide guidance to applicants, licensees, and vendors in designing special lifting devices.

6.4 Technical Specification Changes

Following implementation of modifications and changes to satisfy the guidelines of Section 5.1, changes to facility technical specifications should be made. Items which should be covered by technical specifications will vary depending on the alternatives selected by the particular plant. The following summarizes the types of specifications required for each of the guidelines of Section 5.1.

Guideline	Related Technical Specification (see code below)
5.1.1(1)-(6)	(No T.S. Change required)
5.1.1(7)	A
5.1.2(1)	(Same as 5.1.6)
5.1.2(2)	A, B, C, E
5.1.2(3)	A, B, C, E
5.1.2(4)	A - E (as appropriate, based on analysis)
5.1.3(1)	(Same as 5.1.6)
5.1.3(2)	A, E, F
5.1.3(3)	A, E
5.1.4(1)	(Same as 5.1.6)
5.1.4(2)	A, C, (and B or D if appropriate)
5.1.5(1)(a)	(Same as 5.1.6)

Guideline	Related Technical Specification (see code below)
5.1.5(1)(b)	G
5.1.5(3)(1)(c)	G (if interlocks or mechanical stops are relied on)
5.1.5(2)	A
5.1.6	H, I

Where the following defines the types of technical specifications corresponding to each code letter above:

Code	Technical Specification
A	The maximum load that may be carried by the crane should be specified in the Technical Specifications.
B	Technical Specifications should specify that electric interlocks are operable at all times (if load drop could cause criticality) or when spent fuel is less than _*_ days subcritical (if load drop would not cause criticality).
C	Technical Specifications should specify that the load/cask is not carried greater than _________ inches off the floor of the refueling area.
D	Technical Specifications should specify that movement of the overhead crane load block is prohibited over spent fuel which is less than _*_ days subcritical.
E	Technical Specifications should define the minimum boron concentration as relied on in criticality analyses.
F	Technical Specifications should require operability and surveillance of devices and circuitry that provide containment isolation, and/or transfer to ESF grade filters, on high radiation, and require equipment and personnel access hatches to be closed when handling loads where this alternative is relied on.
G	Technical Specifications should require functional capability of specified electrical interlocks or mechanical stops when equipment within the area protected by the interlocks or mechanical stops is required to be operable.
H	Technical Specifications should require the operability and periodic surveillance of slings or special lifting devices used to handle heavy loads carried over or in proximity to spent fuel in the pool, fuel in the core, or redundant safe shutdown systems.

*Decay time depends on the facility. See Section 2.1 of this report.

Code	Technical Specification
I	Technical Specifications should require operability of both load paths in the single-failure-proof crane, where dual load paths are provided.

6.5 Issues Requiring Further Staff Review

In the course of completing Task A-36, certain areas of potentially adverse safety consequences were identified that were beyond the original scope of Task A-36, such as the potential for heavy loads to damage fuel in the core and the potential for heavy loads, if dropped, to damage safe shutdown systems. To resolve these areas of concern, the scope of Task A-36 was expanded to include these concerns because the loads handled, the equipment used to handle the loads, and the guidelines for safe handling would be the same as that which was already under review in Task A-36.

Task A-36 did not consider loads that weighed less than a "heavy" load, where a "heavy" load is defined as any load greater than the weight of a spent fuel assembly and its handling tool. The handling, and accidental dropping, of a spent fuel assembly is already reviewed as a fuel handling accident, and therefore was not within the scope of Task A-36. In the hearing before the Atomic Safety and Licensing Board in the matter of increased spent fuel storage capacity for Trojan Nuclear Plant, the board raised concerns over the potential for loads which weighed less than a fuel assembly to be carried at greater heights and thus be able to cause more damage than a dropped fuel assembly. The board accepted a technical specification which required that loads should not be handled over the spent fuel pool at heights such that the kinetic energy of the load, if dropped, would be greater than the kinetic energy of a fuel assembly if dropped from its maximum carrying height.

It was determined that an evaluation of the handling of lighter loads, the potential for dropping, measures to preclude dropping, potential consequences, and required staff guidelines were beyond the scope of Task A-36. Additionally the generic letter sent to licensees in June of 1978 did not request any information on lighter loads, such as type and size of loads, frequency of movement, or measures in effect to preclude dropping.

It is therefore recommended that a separate task be established to review the handling of loads weighing less than a spent fuel assembly and to establish necessary guidelines for their safe handling. This task should identify types of small loads handled and frequency of movement over spent fuel, potential for a load drop to occur, potential consequences of a small load drop, and required guidelines that are consistent with the philosophy used for the control of heavy loads. To the extent practical, guidelines for the control of small loads should be similar to those used for heavy loads.

In the interim period until completion of this new task, it is recommended that a technical specification change be made to the licenses of all operating facilities to include a limit on kinetic energy of loads carried over the spent fuel pool similar to technical specification 3.9.7 for Trojan (See Appendix D). We do not have information available or precedents to rely on for establishing interim measures for the control of small loads handled over the reactor core. The above recommended task would have to establish such required measures.

APPENDIX A

ANALYSES OF POSTULATED LOAD DROPS

Certain of the alternatives in Sections 5.1.2 through 5.1.5 of this report call for an analysis of postulated load drops and evaluation of potential consequences to assure that the evaluation criteria of Section 5.1 are met for such an event. Section A-1 of this appendix identifies certain considerations that should be included in such evaluations. Sections A-2 and A-3 identify certain additional considerations and assumptions that should be used in analyzing the potential consequences of a drop of the reactor vessel head assembly or the spent fuel shipping cask; other load drops that are analyzed should use similar considerations and assumptions that are appropriate for these other loads. Section A-4 provides guidance in performing criticality calculations.

1. GENERAL CONSIDERATIONS

Analyses of postulated load drops should as a minimum include the considerations listed below. Other considerations may be appropriate for the particular load drop being analyzed; for example, for a reactor vessel head assembly or a spent fuel cask drop analysis, the additional considerations listed in Sections A-2 or A-3 should be used. In evaluating the potential for a load drop to result in criticality, the considerations of A-4 should also be followed. The following should be considered for any load drop analysis, as appropriate:

(1) That the load is dropped in an orientation that causes the most severe consequences;

(2) That fuel impacted is 100 hours subcritical (or whatever the minimum that is allowed in facility technical specifications prior to fuel handling);

(3) That the load may be dropped at any location in the crane travel area where movement is not restricted by mechanical stops or electrical interlocks;

(4) That credit may not be taken for spent fuel pool area charcoal filters if hatches, wall, or roof sections are removed during the handling of the heavy load being analyzed, or whenever the building negative pressure rises above (-)1/8 inch (-3 m) water gauge;

(5) Analyses that rely on results of Table 2.1-1 or Figures 2.1-1 or 2.1-2 for potential offsite doses or safe decay times should verify that the assumptions of Table 2.1-2 are conservative for the facility under review. X/Q values should be derived from analysis of on-site meteorological measurements based on 5% worst meteorological conditions.

(6) Analyses should be based on an elastic-plastic curve that represents a true stress-strain relationship.

(7) The analysis should postulate the "maximum damage" that could result, i.e., the analysis should consider that all energy is absorbed by the structure and/or equipment that is impacted

(8) Loads need not be analyzed if their load paths and consequences are scoped by the analysis of some other load.

(9) To overcome water leakage due to damage from a load drop, credit may be taken for borated water makeup of adequate concentration that is required to be available by the technical specifications.

(10) Credit may not be taken for equipment to operate that may mitigate the effects of the load drop if the equipment is not required to be operable by the technical specifications when the load could be dropped.

2. REACTOR VESSEL HEAD DROP ANALYSIS*

Where a reactor vessel head drop analysis is to be performed to satisfy the PWR Containment or BWR Reactor Building guidelines (Sections 5.1.3 or 5.1.4) of this report, the analysis should consider the following to assure that the evaluation criteria of Section 5.1 are satisfied.

(1) Impact loads should include the weight of the reactor vessel (RV) head assembly (including all appurtances), the crane load block, and other lifting apparatus (i.e., the strongback for a BWR).

(2) All potential accident cases during the refueling operations. Areas of consideration as a minimum should be:

   (a) Fall of the RV head from it's maximum height while still on the guide studs followed by impact with the RV flange;

   (b) Fall of the RV head from its maximum height considering possible objects of impact such as the guide studs, the RV flange, the steam dryer (BWR) or structures beneath the path of travel; and

   (c) Impact with the fueling cavity wall due to load swing with the subsequent drop of the RV head due to lifting device or wire rope failure.

(3) All cases which are to be considered should be analyzed in the actual medium present during the postulated accident, e.g., for a PWR prior to reassembly of the reactor, the fueling cavity is drained after the head engages the guide studs to allow for visual inspection of the reactor core control drive rods insertion into the head. During this phase it should be considered that the head will only fall through air, without any drag forces produced by a water environment.

*These guidelines only consider the dropping of the RV head assembly during refueling and do not apply directly to dropping of the reacter internals such as the steam dryer (BWR), moisture separator (BWR) or the upper core internals (PWR); however, similar assumptions and considerations would apply to analyses of dropping of reactor internals.

(4) In those Nuclear Steam Supply Systems where portions of the reactor internals extend above the RV flange, the internals should be analyzed for buckling and resultant adverse effects due to the impact loading of the RV head. It should be demonstrated that the energy absorption characteristics (causing buckling failure) of these internals should be such that resultant damage to the core assembly does not cause a condition beyond the acceptance criteria for this analysis.

(5) Reactor vessel supports should be evaluated for the effects of the transmitted impact loads of the RV head. In the case of PWRs where the RV is supported at its nozzles, the effects of bending, shear and circumferential stresses on the nozzles should be examined. For BWRs the effects of these impact loads on the RV support skirt should be examined.

(6) The RV head assembly should be considered rigid and not experience deformation during impact with other components or structures.

3. SPENT FUEL CASK DROP ANALYSIS

Where a cask drop analysis is to be performed to satisfy the guidelines in Sections 5.1.2, 5.1.4, or 5.1.5 of this report, it should consider the following in addition to the general considerations of Section A-1 to assure that the evaluation criteria of Section 5.1 are satisfied:

(1) Applying a single-failure to the lifting assembly, consider that the cask is dropped in an orientation that will result in the most severe consequences.

(2) Impact loads should include a fully loaded cask (with water, where applicable) and all equipment required for lifting and set down such as baseplates, lifting yokes, wire ropes and crane blocks.

(3) Restricted path travel of the spent fuel cask (defined by electrical interlocks, mechanical stops, and crane travel capability) should be evaluated to determine the locations and probable accident cases along the path where damage could occur to:

  (a) the floor and walls of the Spent Fuel Pool (SFP);
  (b) racks within the SFP which support the spent fuel;
  (c) the spent fuel itself;
  (d) the refueling channel gate; or
  (e) safety related systems, components and structures beneath or adjacent to the travel path of the cask.

(4) In the analysis consideration may be given to drag forces caused by the environment of the postulated accident case, e.g., when the spent fuel cask is postulated to drop into the SFP, credit may be taken for drag forces caused by the water in the SFP. Water level assumed for such analyses should be the minimum level allowed by technical specifications.

(5) Credit may be taken for energy absorbing devices integral to the cask if attached during the handling operations in determining the amount of energy imparted to the spent fuel or safety related systems, components or structures.

(6) For the purpose of the analysis the cask should be considered rigid (except for devices and appurtences specifically designed for energy absorption and in place) and not to experience deformation during impact.

(7) In the calculating the center of gravity, consideration should be given to modifications made to the cask after purchase, e.g., addition of a perforated metal basket within the cask.

4. CRITICALITY CONSIDERATIONS

4.1 Spent Fuel Pool Neutronics Analysis

In Sections 5.1.2, "Spent Fuel Pool Area - PWR," and 5.1.4, "Reactor Building - BWR," a number of alternatives are presented for the control of heavy loads in spent fuel pool areas. Some of these alternatives include neutronics calculations to demonstrate that crushing the fuel and fuel rack will not result in criticality. This section is included here to give the licensees guidance in performing their neutronics calculation.

A discussion of the potential for criticality under load drop conditions is discussed in Section 2.2, and summarized in Section 2.2.6. The results of this section should be used as a guide to determine which neutronics or other analyses are required to evaluate the potential for criticality for a specific plant area. A licensee may choose to use the results of section 2.2, rather than performing an independent neutronics analysis for his plant. If a licensee uses the results of Section 2.2 rather than performing an independent neutronics analysis, he should verify that the assumptions and model fuel assembly of Section 2.2 are valid for his plant.

For PWR spent fuel pools, credit may be taken under the accident conditions of a load drop for the boron in the spent fuel pool water to maintain subcriticality. In this case the required boron concentration should be specified in the facility Technical Specification, and regular monitoring of the boron concentration in the spent fuel pool should also be specified. Likewise, if the neutronics analysis postulates a bounding distribution of non-spent fuel within the spent fuel pool, then the Technical Specifications must be modified to require that the actual distribution of fuel is no more deleterious than that assumed in the analysis. In postulating a limiting distribution of non-spent fuel, the licensee may either assume an infinite array or a finite array. The largest finite array of non-spent fuel a licensee should have to consider would be that of an off-load core.

In this neutronics analysis the licensee must demonstrate that the fuel remains subcritical in the optimum crushed configuration. It is adequate to assume that the optimum configuration is with the rack crushed to uniformly reduce the separation between assemblies and the spacing between fuel pins uniformly reduced to maximize k_{eff}. All boral and structural material may be assumed to remain in its original configuration relative to the fuel, and not forced out of the fuel array.

The neutronics analysis for the spent fuel pool should consider the case where it has become necessary to off-load an entire core into the spent fuel pool and a heavy load is dropped on fuel in the pool.

As noted in Section 5.1.4 it is not necessary to analyze the effects of crushing on k_{eff} for BWR spent fuel pools that use boron plate cans and do not rely on spacing to maintain subcriticality.

4.2 Reactor Core Neutronics Analyses

4.2.1 Neutronics Analyses for a BWR Core

For a BWR core, the potential for a load drop to drive control rods out of the core should be analyzed using the appropriate considerations of Sections A-1 and A-2. If this analysis shows that postulated load drops could drive control rods out of the core, the number of rods that could be affected should be determined, and a neutronics analysis performed to determine the potential for criticality to result. If in the analysis it is assumed that all rods are in the core just prior to the load drop, then the facility technical specifications should require that all rods are in when handling a heavy load over the core.

4.2.2 Neutronics Analyses for a PWR Core

In Table 2.2-2, we see that crushing the model PWR core in 2000 ppm boron refueling water increases k_{eff} by about 0.02. Since only one model fuel geometry was considered here, other fuel geometries could have a slightly higher reactivity insertion due to crushing. A value of 0.05 may be used as a bounding worst case reactivity insertion value due to crushing of a PWR core. In performing a neutronics evaluation of a postulated load drop on a PWR core, a licensee may use this estimated reactivity insertion limit in lieu of performing a plant specific calculation. If a licensee can demonstrate that for his fuel a value less than 0.05 is bounding, then he may use this lower value instead.

The current Technical Specifications require that during refueling k_{eff} should be maintained at 0.95 or less. This is based on an uncrushed core. To perform a neutronics analysis to demonstrate that crushing the core will not drive it critical at least two alternatives for demonstrating this are acceptable.

(1) The licensee can perform a neutronics analysis on his core uniformly crushed in the x-y direction to maximize k_{eff}. If the licensee chooses this option he must demonstrate that the maximum k_{eff} is no greater than 0.95, with all uncertainties taken into account.

<u>OR</u>

(2) Using his core refueling neutronics analysis (uncrushed), the licensee can demonstrate that k_{eff} for the uncrushed core is no greater than 0.90. Then, using the estimated 0.05 maximum reactivity insertion due to crushing, the maximum achievable k_{eff} is still less than 0.95.

5. ACCEPTANCE CRITERIA

In performing the above analyses, the acceptance criteria for resultant damage should be that it does not cause a condition that may exceed evaluation criteria I-IV stated in Section 5.1 of this report.

APPENDIX B
ESTIMATES OF EVENT PROBABILITIES

Fault trees (Figures 5.2-1, 5.2-2, and 5.2-3) for various load handling scenarios are contained in Section 5.2 of this report. This appendix develops probability estimates for the various events contained in these fault trees.

The numbers in the left hand margin correspond with the event numbers shown in Figures 5.2-1, 5.2-2, and 5.2-3. Figures B-1 through B-3 correspond to the fault trees in Section 5.2, but show the probabilities developed in this appendix.

Probabilities used are best estimates of upper and lower bounds with conservative margins to allow for uncertainties. Where little data is available, estimates are based on engineering judgment as to a conservative value. For the purposes of determining a median within the range between the upper and lower bounds, it was assumed that the variability of failure rates was distributed log normally between the bounds.

A. Figure 5.2-1 - Loads Handled Near Spent Fuel Pool: Offsite Releases

1.1.1 The probability that the spent fuel pool contains "hot" spent fuel depends on the decay time of the spent fuel. From Figures 2.1-1 and 2.1-2, between 42 and 74 days is a safe decay time if a full core were damaged. If we assume that heavy loads are handled uniformly through the year and that any heavy load could cause excessive releases, then the P(1.1.1) is between 0.1 and 0.2 per reactor year (i.e., 38/365 to 72/365).

1.1.2.1 For the purposes of this review, it is estimated that the probability of failure to follow a given procedure is between 5×10^{-2} and 10^{-2} per event, or 2 to 10 failures per year assuming 200 lifts per year. This presumes that the guidelines of Section 5.1.1 are met, whereby crane operators are trained in proper conduct of operation and procedures to be followed. The 200 lifts per year is based on the number of cask and other load handling events that may occur per year as shown in Table 3.1-1. The above estimate of failure to follow a given procedure is also consistent with human reliability estimated failure rate of 10^{-2} obtained in the Reactor Safety Study WASH-1400 based on data from the United Kingdom Atomic Energy Agency and the U.S. military.

1.1.2.2 Electrical interlock reliability was estimated to be between 10^{-2} and 10^{-3} per demand. These interlocks are not challenged unless there is a failure to follow prescribed load paths (i.e., event 1.1.2.1). This estimate is more conservative than the electrical interlock failure rate assessment used in WASH-1400 of 10^{-3} to 10^{-4} per demand. The higher failure rate was used to account for the potential that there may be some interaction with the event of failure to follow the prescribed load path, whereby an inexperienced operator violates the load path procedure and also fails to verify that interlocks are operable or intentionally bypasses the interlock.

1.1.2.3 Based on the data collected from the Navy, it is expected that the probability of handling system failure for nuclear plant cranes will be on the order of between 10^{-5} and 1.5×10^{-4} per lift. This presumes an improvement by a factor of 0.5 over the Navy cranes based on improved procedures at nuclear plants, and conformance to guidelines in Section 5.1.1 of this report concerning operator training and crane inspection (i.e., the failure rate will be cut in half due to these measures).

However, the probability of handling system failure, given that the prescribed load path has not been followed and that electrical interlocks have failed, would be greater than the above estimate. This would be due to common mode effects such as a poorly trained or unqualified operator that fails to follow the prescribed load path, fails to check the operability of the interlocks, and then proceeds to improperly operate the handling system leading to a load drop. This then is a connective link between events 1.1.2.1, 1.1.2.2, and 1.1.2.3. If we presume that this reduces the handling system reliability by a factor of ten, this gives a result of probability of handling system failure given that interlocks have failed and the operator has failed to follow the prescribed load path of between 10^{-4} and 1.5×10^{-3} per lift.

1.1.2.4 Even though interlocks fail and procedures are violated, the load may not be brought over spent fuel or the load may be dropped at some point prior to or after being brought over spent fuel. Even if the drop occurs over spent fuel it may not impact "hot" spent fuel. Based on the length of the load paths that could be followed for such loads as contaminated waste casks, transfer canal gate, spent fuel cask, shield plugs and other loads normally handled near spent fuel, spent fuel usually occupies less than 10% of any potential path length and many possible paths do not even go over spent fuel. Based on this, it is estimated that the probability that the drop could occur over "hot" spent fuel, given that the prescibed load path has not been followed, is between 10^{-1} and 10^{-2}, given that events 1.1.2.1, 1.1.2.2, and 1.1.2.3 occur.

1.1.2 Combining probabilities, it is estimated that P(1.1.2) is between 2×10^{-5} and 2×10^{-9} with a median of approximately 10^{-7} per reactor year.

1.1 Combining 1.1.1 and 1.1.2, we obtain an estimate of the <u>probability of offsite releases that exceed guidelines</u> due to a load drop for loads handled near spent fuel of between 4×10^{-6} and 2×10^{-10} with a median of 2×10^{-8} <u>per reactor year</u>.

1.2.1 <u>Potential For Criticality</u>

This event can occur if a core off load were to occur, whereby the fuel in the core has been subcritical for a short period of time such that it still contains some enriched fuel. It is estimated that a core off-load event may occur once every 50 to 200 reactor years, giving a probability of core off-load of between 2×10^{-2} and 5×10^{-3} per reactor year.

1.2.2 This requires failure of an operator to follow prescribed refueling procedures. As in other similar operator actions, the probability of failure is estimated to be between 5×10^{-2} and 10^{-2} per event, if independent from other 1.2 events. However, given that the prescribed load path has not been followed, that the electrical interlocks have failed, and that the handling system has failed (i.e., given that 1.2.3 occurs), the probability that boron concentration is inadequate could be greater than the above estimate. This would be due to common mode effects, such as a poorly trained or unqualified crane operator which increases the probability of each of these events occurring simultaneously. If we pressume that such an effect increases by a factor of 10 the probability of having inadequate boron concentration, this gives a probability of 5×10^{-1} to 10^{-1} per event.

1.2.3 This is similar to event 1.1.2, except that the area of concern is highly enriched fuel that could be brought critical rather than "hot" spent fuel in terms of potential for an excessive release. The probability that the load strikes this fuel, given that highly enriched fuel is in the pool is the same as the probability of event 1.1.2.

1.2.4 This is also largely unknown; however, if we consider all the possible configurations that spent fuel could be in after impact by a heavy load, only very few of these are such that the spent fuel is brought uniformly close together with the potential for criticality. It is estimated that this probability is between 10^{-1} and 10^{-3}.

1.2 Combining probabilities we find that the probability of criticality in spent fuel to result from a load drop for loads handled near spent fuel is negligible.

1. Combining 1.1 and 1.2 gives, for loads handled near spent fuel: an estimate that consequences exceed guidelines of between 2×10^{-10} and 4×10^{-6} with a median of approximately 2×10^{-8} per reactor year.

B. Figure 5.2-2 - Loads Handled Over Spent Fuel Pool: Offsite Releases

2.1.1.1 This is the same as the probability of event 1.1.2.3.

2.1.2 This is the same as the probability of event 1.1.1.

2.1.1.2.1.1 This is the same as the probability of event 1.1.2.1.

2.1.1.2.1.2 The requirement to segregate "hot" spent fuel would be specified in facility refueling procedures, as well as facility technical specifications. Failure to segregate "hot" spent fuel would be a failure to follow prescribed procedure. For other operator actions, the probability of failure to follow a procedure is estimated to be between 5×10^{-2} and 10^{-2} per event. However, in this case, refueling operations call for a check on fuel position in the pool following refueling. This will tend to reduce the probability of failing to segregate the "hot" spent fuel away from the cask area. It is estimated that this probability is between 10^{-2} and 10^{-3} per event.

2.1.1.2.2.1 This is the same as the probability of event 1.1.2.1.

2.1.1.2.2.2 This is the same as the probability of event 1.1.2.2.

2.1 Combining probabilities, we obtain an estimate of the probability of offsite releases that exceed guidelines due to a load drop for loads handled over the spent fuel pool of between 3×10^{-5} and 2×10^{-8} with a median of approximately 7×10^{-7} per reactor year.

2.2.1 Potential For Criticality

This is the same as the probability of event 1.2.1.

2.2.2 This is the same as the probability of event 2.1.1.2, except the fuel protected is enriched rather than a concern for the release of gap activity.

2.2.3 This is the same as the probability of event 1.1.2.3.

2.2.4 This is the same as the probability of event 1.2.4.

2.2.5 This is the same as the probability of event 1.2.2.

2.2 Combining probabilities we obtain an estimate of the probability of criticality in the spent fuel pool due to a load drop for loads handled over the spent fuel pool of less than 3×10^{-6} per reactor year.

2. Combining 2.1 and 2.2 gives, for loads handled over the spent fuel pool, a probability estimate that consequences exceed guidelines of between 2×10^{-3} and 3×10^{-5}, with a median of approximately 7×10^{-7} per reactor year.

C. Figure 5.2-? - Single Failure Proof Handling System:
Offsite Rel...s

3.1.1(A) Loads covered by branch (A) of this fault tree are large enough to cause excessive releases if dropped from a sufficient height. The fuel impacted would in most cases be fuel in the core, and, therefore, would be "hot" fuel unless the drop occurs before initial criticality. However, the drop may occur at low heights where little or no fuel damage occurs. In fact, most drops due to mechanical failures occur at low lift heights where weak components fail shortly after a load is applied. Additionally, the load may be deflected by impact with the vessel flange or internal surfaces, reducing the energy that may be imparted on spent fuel. To account for this potential for loads carried over spent fuel, it is estimated that between 10% and 25% of load drops directly over "hot" spent fuel result in releases that approach 10 CFR Part 100 limits for loads such as the reactor vessel head, vessel internals, inspection platform, etc. Therefore, an estimate of 10^{-1} and 2.5×10^{-1} per load drop was used for the probability that a load dropped on spent fuel results in excessive offsite releases.

3.1.1(B) This is the same as the probability of event 1.1.1.

3.1.3 Event 3.1.3(B) is the same as the probability of event 1.1.2.4. Event 3.1.3(A) covers loads carried over spent fuel. For these loads, between 5% and 25% of the path length is over spent fuel, and, therefore, an estimate of the probability that the load drop occurs over spent fuel is between 5 x 10^{-2} and 2.5 x 10^{-1} per each event where the prescribed load path is not followed.

3.1.2.1(CF) Section 4 of this report estimates the probability of handling system failure of between 10^{-5} and 1.5 x 10^{-4} per reactor year for a crane that does not have single failure proof features. With the "single-failure-proof" crane guidelines (NUREG-0554), certain load-bearing components are provided with dual or counterpart components such that if one were to fail, its counterpart could handle the load and preclude dropping. Events CF.2.1 and CF.2.2 (sheet 2 of Figure 5.2-3) pertain to these components and their counterparts. Certain other components are allowed to have increased design safety factors per the guidelines of NUREG-0554, in lieu of having backup or counterpart components. Event CF.4 pertains to these components which do not have redundant counterparts.

Additionally, the guidelines of NUREG-0554 call for protection against possible "two-blocking" and "load-hangup" events. This may be done by limit switches and overload protection devices respectively, as shown by the fault trees for events CF.1 and CF.3. NUREG-0554 allows use of the limit switches to terminate hoisting as an alternate to designing the crane to withstand a "two-blocking" event. If designed to withstand "two-blocking", a test would be performed to demonstrate this ability. It was deemed that the use of limit switches was the less reliable of the two options, and therefore the fault trees modeled use of limit switches.

The fault tree on sheet 2 of Figure 5.2-3 may be used both for branches (A) and (B), (i.e., for events 3.1.2.1(A) and 3.1.2.1(B) with appropriate probabilities used for the loads covered by each branch).

For branch (A) which covers loads carried over spent fuel, from Table 3.1-1, we see that there are typically between 4 and 10 lifts per year over spent fuel, mostly over the reactor vessel (vessel head, vessel internals, vessel inspection equipment, etc). Thus, probabilities for branch (A) are estimated on the basis of loads being handled at a frequency of 4 to 10 times per year. However, loads covered by branch (B) are handled on the order of up to 200 times per year near spent fuel (spent fuel shipping cask, waste/debris/ spent resin casks, refueling plugs and gates, shield plugs). Similarly, branch (B) probabilities are estimated using these larger frequencies.

In estimating the probability of handling system failure for event 1.1.2.3, it was estimated that there could be some common cause connection with procedural events, and therefore the estimate of the failure probability of the handling system was increased. A similar connection may exist for events 3.1.2(B) and 3.1.4(B). However, for event 3.1.2(A) failure of handling system, (if single-failure-proof), no such common causes could be identified. The probability estimates used in arriving at the failure probability of the handling system are sufficiently conservative to encompass minor interactions between events 3.1.2(A) and 3.1.1(A) or 3 1.3(A

CF.1.1 Of the 43 events reported in the Navy data report (Section 4 of this report), 2 "load-hangup" events occurred due to operator error. This gives us an estimate of probability of "load-hangup" of between $(\frac{2}{43}) \times (10^{-5})$ and $(\frac{2}{43}) \times (1.5 \times 10^{-4})$ or between 4.7×10^{-7} and 7×10^{-6} per lift. For CF.1.1(A), this gives a result of between 2×10^{-6} and 7×10^{-5} per reactor year (4 to 10 lifts per year).

For CF.1.1(B), this gives a result of between 10^{-4} and 1.4×10^{-3} per reactor year (200 lifts per year); however, due to potential common cause effects with event 3.1.3(B) (failure to follow prescribed load path due to a poorly trained operator, for example) a more reasonable estimate would be between 10^{-3} and 1.4×10^{-2} per reactor year.

CF.1.2 Limit switches are similar to the interlock switches discussed for event 1.1.2.2, and thus the same probability estimates for event 1.1.2.2 may be used for the limit switches. The complexity of the overload protection devices is similar to these limit switches, and thus the probability estimate for limit switches and interlock switches was also applied to the overload protection devices. The same probability estimate may be used for CF.1.2(A) and for CF.1.2(B).

CF.3.1 "Two-blocking" due to operator error occurred in 15 of the 43 events reported in the Navy data. This gives an estimate for CF.3.1 of between $(\frac{15}{43}) \times (10^{-5})$ and $(\frac{15}{43}) \times (1.5 \times 10^{-4})$, or between 3.5×10^{-6} and 5.2×10^{-5} per lift. For CF.3.1(A), this gives a probability estimate of between 1.4×10^{-5} and 5.2×10^{-4} per reactor year (4 - 10 lifts per year). For CF.3.1(B), this gives an estimate of between 7×10^{-4} and 10^{-2} per reactor year (200 lifts per year); however, due to potential common cause effects with event 3.1.4(B) (failure to follow prescribed load path due to a poorly trained operator, for example) a more reasonable estimate would be between 7×10^{-3} and 10^{-1}

CF.3.2 Failure of the limit switch is the same as the probability of event CF.1.2 (see discussion of CF.1.2 above).

CF.3.3 Due to common mode failures, the probability of failure of the upper limit switch given that the lower has failed is greater than the probability of failure of the lower limit switch due to common mode effects. However, the NUREG-0554 guidelines call for these two limit switches to be independent, of different designs, and activated by separate mechanical means. This will tend to make common mode failure for these limit switches much less likely.

If we assume that one out of every 10 to 100 failures of the first limit switch causes a failure of the second limit switch or that the mechanism that caused failure of the first limit switch also causes failure of the second component, then the second limit switch has a failure probability of between 10^{-1} and 10^{-2} due to common mode effects, and thus has a probability of failure, given that the first switch has failed, of between $(10^{-2} + 10^{-1})$ and $(10^{-3} + 10^{-2})$ or between 10^{-1} and 10^{-2} per demand.

CF.2.1 Of the 43 load drop events reported in the Navy data (Section 4.2 of this report), 23 events were due to crane component failures. Some of these are random material failures, while others may be due to personnel errors such as design deficiencies, improper maintenance or inadequate inspection. From event 1.1.2.3 an estimate of the probability of failure of a single crane component would be between $(\frac{23}{43})x(10^{-5})$ and $(\frac{23}{43})x(1.5 \times 10^{-4})$ per lift, or between 5.3×10^{-6} and 8×10^{-5} per lift. For CF.2.1(A), this gives an estimate of component failure of between 2×10^{-5} and 8×10^{-4} per reactor year (4 - 10 lifts per year). For CF.2.1(B) this gives an estimate of component failure of between 10^{-3} and 2×10^{-2} per reactor year. We were not able to identify any common cause link between event 3.1.4(B) (failure to follow prescribed load path) and event CF.2.1(B). The above estimate is sufficiently conservative to account for minor interactions or common cause links.

CF.2.2 Again if we assume that one out of every 10 to 100 failures of a crane component causes a failure of the backup component, or that the mechanism that caused failure of the first component also causes failure of the second component, then the second or backup component has a failure probability of between 10^{-1} and 10^{-2} due to common mode effects. This gives an overall probability of failure for the backup component for CF.2.2(A) given that the first component has failed, of between $(8 \times 10^{-4} + 10^{-1})$ and $(2.1 \times 10^{-5} + 10^{-2})$ or between 10^{-1} and 10^{-2}.

For CF.2.2(B), this gives an overall probability of failure for the backup component due to random causes and common mode effects, given that the first component has failed, of between $(10^{-3} + 10^{-2})$ and $(2 \times 10^{-2} + 10^{-1})$ or between 10^{-2} and 1.2×10^{-1}

CF.4 None of the load drop events in the Navy data (43 events - see Section 4.2) occurred due to failures in components where NUREG-0554 does not require a dual or redundant component. If we assume that the 44th load drop event could have been due to a failure in one of these components, then an estimate of failure for such components is: $(\frac{1}{44})x(10^{-5})$ to $(\frac{1}{44})x(1.5 \times 10^{-4})$ per lift, or 2.3×10^{-7} to 3.4×10^{-6} per lift.

Conformance to NUREG-0554 requires increased design safety factors for these components, usually increased by a factor of about 2 (e.g., for certain components, from a safety factor of 5:1 to a factor of 10:1). This will tend to reduce the probability of failure of these components.

We will conservatively assume that this reduces the failure probability only by a factor of 10^{-1}, although we would expect that such a large change in the safety factor would have a much greater effect on reducing the failure probability. Using an improvement of 10^{-1} in failure probability gives an estimate of the failure of components which do not have a dual or redundant counterpart after compliance with NUREG-0554 of between 2.3×10^{-8} and 3.4×10^{-7} per lift.

For CF.4(A) this gives a failure probability of between 9×10^{-8} and 3×10^{-6} per reactor year (4 - 10 lifts per year).

For CF.4(B) this gives a failure probability of between 5×10^{-6} and 7×10^{-5} per reactor year (200 lifts per year). We were not able to identify any common cause effects between CF.4(B) and 3.1.4(B). The above estimate for CF.4(B) is sufficiently conservative to account for minor interactions or common cause links.

3.1.2.1(A) (or CF(A)) Combining probabilities, we obtain the following:

$2 \times 10^{-9} \leq P(CF.1(A)) \leq 7 \times 10^{-7}$ per reactor year.
$2 \times 10^{-7} \leq P(CF.2(A)) \leq 8 \times 10^{-5}$ per reactor year.
$10^{-10} \leq P(CF.3(A)) \leq 5 \times 10^{-7}$ per reactor year.
$9 \times 10^{-8} \leq P(CF.4(A)) \leq 3 \times 10^{-6}$ per reactor year.

We can combine the above probabilities through an "or" gate to obtain the following estimate of probability of failure of:
$3 \times 10^{-7} \leq P(CF(A)) \leq 8 \times 10^{-5}$
with a median of 5×10^{-6} per reactor year.

3.1.2.1(B) (or CF(B)) Combining probabilities, we obtain the following:

$10^{-6} \leq P(CF.1(B)) \leq 10^{-4}$
$10^{-5} \leq P(CF.2(B)) \leq 2 \times 10^{-3}$
$7 \times 10^{-9} \leq P(CF.3(B)) \leq 10^{-6}$
$4.5 \times 10^{-6} \leq P(CF.4(B)) \leq 6.8 \times 10^{-5}$

Similarly, combining these through an "or" gate gives:
$2 \times 10^{-5} \leq P(CF(B)) \leq 2 \times 10^{-3}$, with a median of 2×10^{-4} per reactor year.

3.1.2.2 - (Failure of Rigging):

3.1.2.2.1 From the data on Navy cranes contained in Section 4.2 of this report, we can obtain an estimate of rigging failure of between 7×10^{-7} and 10^{-5} per lift, since rigging accounted for 7% of all failures. This presumes an improvement on the order of .5 per lift based on improved procedures at nuclear power plants, and conformance to guidelines in Section 5.1.1 of this report concerning rigging. These limits become 2.8×10^{-6} to 10^{-4} per reactor year for branch (A) loads (4-10 lifts per year). Similarly, these limits become 10^{-4} and 2×10^{-3} per reactor year for branch (B) loads (200 lifts per year). Because a poorly trained crane operator could select improper rigging (event 3.1.2.2(B)) and could fail to follow the proper load path (event 3.1.4(E)), there is a common cause link between these two events. If we presume that this increases the failure probability of the rigging by a factor of 10, this gives a probability of event 3.1.2.2.1(B), given that 3.1.4(B) has occurred, of between 10^{-3} and 2×10^{-2} per reactor year.

3.1.2.2.2 Use of dual or redundant rigging may compensate for random material failures in the rigging or personnel errors that occur on only one set of rigging. However, an individual may select or install both sets of rigging in the same, although incorrect, manner thus leading to failure of both sets of rigging due to a single common cause. Therefore, the probability of failure of the second set of rigging, given that the first has failed will be somewhat greater than the probability for event 3.1.2.2.1.

If we estimate that between 5% and 25% of rigging failures are such that they are likely to occur in the counterpart rigging due to common mode effects, then an estimate for probability of failure of the redundant or counterpart rigging given that the first set of rigging has failed is between 5×10^{-2} and 2.5×10^{-1}. This holds for branch (A) and branch (B) loads.

3.1.2.2 The above estimates result in a probability of failure of the rigging of between 10^{-7} and 3×10^{-5} per reactor year for branch (A), and between 5×10^{-5} and 5×10^{-3} per reactor year for branch (B).

3.1.2 (Failure of Handling System):
Combining probabilities, we obtain an estimate of failure of the handling system that satisfies single-failure-proof guidelines of between 4.6×10^{-7} and 1.2×10^{-4} per reactor year for loads carried over spent fuel (3.1.2(A)), and between 6.5×10^{-5} and 3.2×10^{-3} for loads carried near spent fuel (3.1.2(B)).

3.1(A) Combining 3.1.1(A), 3.1.2(A), and 3.1.3(A), we obtain an estimate of probability of excessive offsite releases, if a single-failure-proof crane is relied on, of between 2.3×10^{-9} and 7.5×10^{-6} per reactor year, with a median of 1.3×10^{-7}, for loads carried over spent fuel.

3.1(B) Similarly, an estimate of probability of excessive offsite releases, if a single-failure-proof crane is relied on, is between 2×10^{-9} and 2.7×10^{-6} per reactor year with a median of 7.3×10^{-8}, for loads handled near spent fuel.

3.1 Combining 3.1(A) and 3.1(B) through an "or" gate gives a probability of a load drop resulting in offsite releases that exceed guidelines if a single-failure-proof crane is provided of between 3×10^{-9} and 10^{-5}, with a median of 2×10^{-7} per reactor year.

3.2.1 Criticality
This is the same as the probability of event 1.2.1.

3.2.2 This is the same as the probability of event 1.2.4.

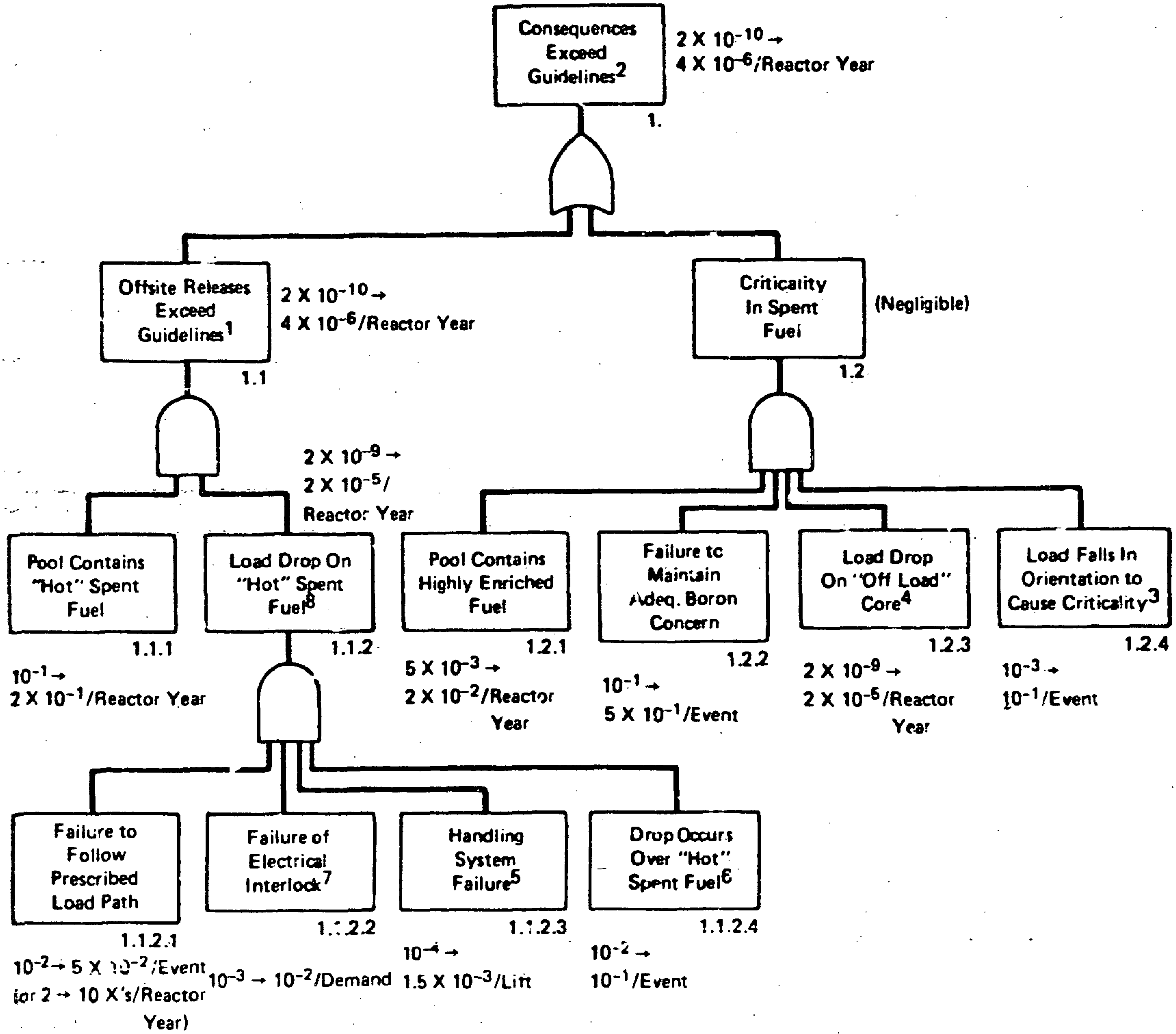
3.2.3 As shown on Figure 5.2-3, sheet 2, this event can occur either as a result of a failure of the handling system carrying a load normally handled over spent fuel, or as a result of a breakdown in following a prescribed load path and a failure of the handling system for loads normally handled near spent fuel. Event 3.2.3.1 is the same as the probability of event 3.1.2(A); event 3.2.3.2.1 is the same as the probability of event 3.1.2(B); and event 3.2.3.2.2 is the same as the probability of event 1.1.2.1. The potential for common cause effects for events

3.2.3.2.1 and 3.2.3.2.2 is already taken into account in the estimate for event 3.1.2(B).

3.2 Combining probabilities, we obtain an estimate of the probability of a load drop causing criticality, where a single-failure-proof crane is provided, of less than 10^{-6} per reactor year.

3. Combining 3.1 and 3.2 gives the following where a single-failure-proof crane is provided:

$3 \times 10^{-9} \leq$ P(Consequences that Exceed Guidelines) $\leq 10^{-5}$ with a median of 2×10^{-7} per reactor year.



1 Evaluation Criterion I of Section 5.1
2 Evaluation Criteria of Section 5.1
3 Given That Events 1.2.1, 1.2.2, and 1.2.3 Occur
4 Given That Event 1.2.1 Occurs
5 Given That Events 1.1.2.1 and 1.1.2.2 Occur
6 Given That Events 1.1.1.1, 1.1.2.1 1.1.2.2, and 1.1.2.3 Occur
7 Given That Event 1.1.2.1 Occurs
8 Given That Event 1.1.1 Occurs

FIGURE B-1 FAULT TREE FOR LOADS HANDLED NEAR SPENT FUEL POOL

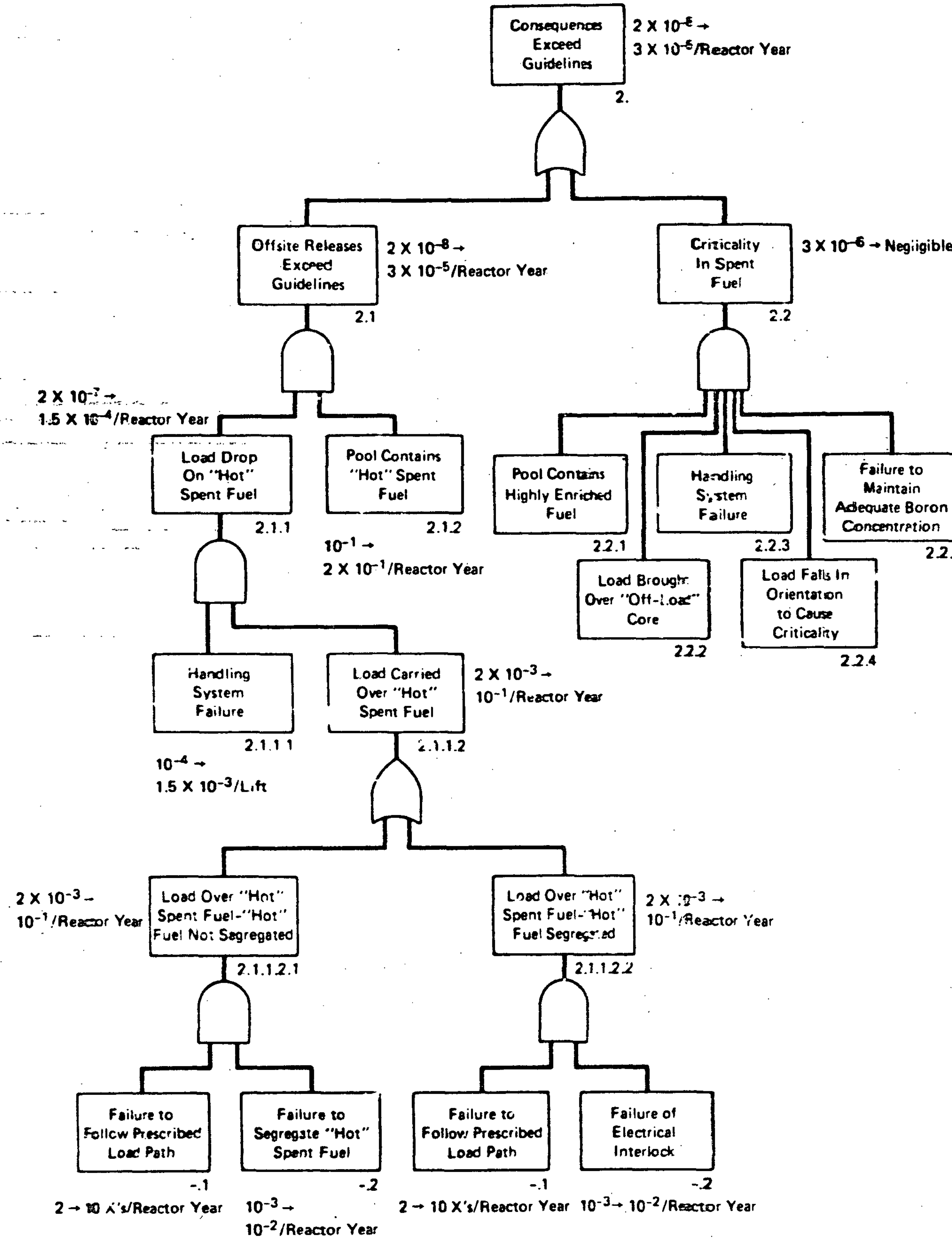
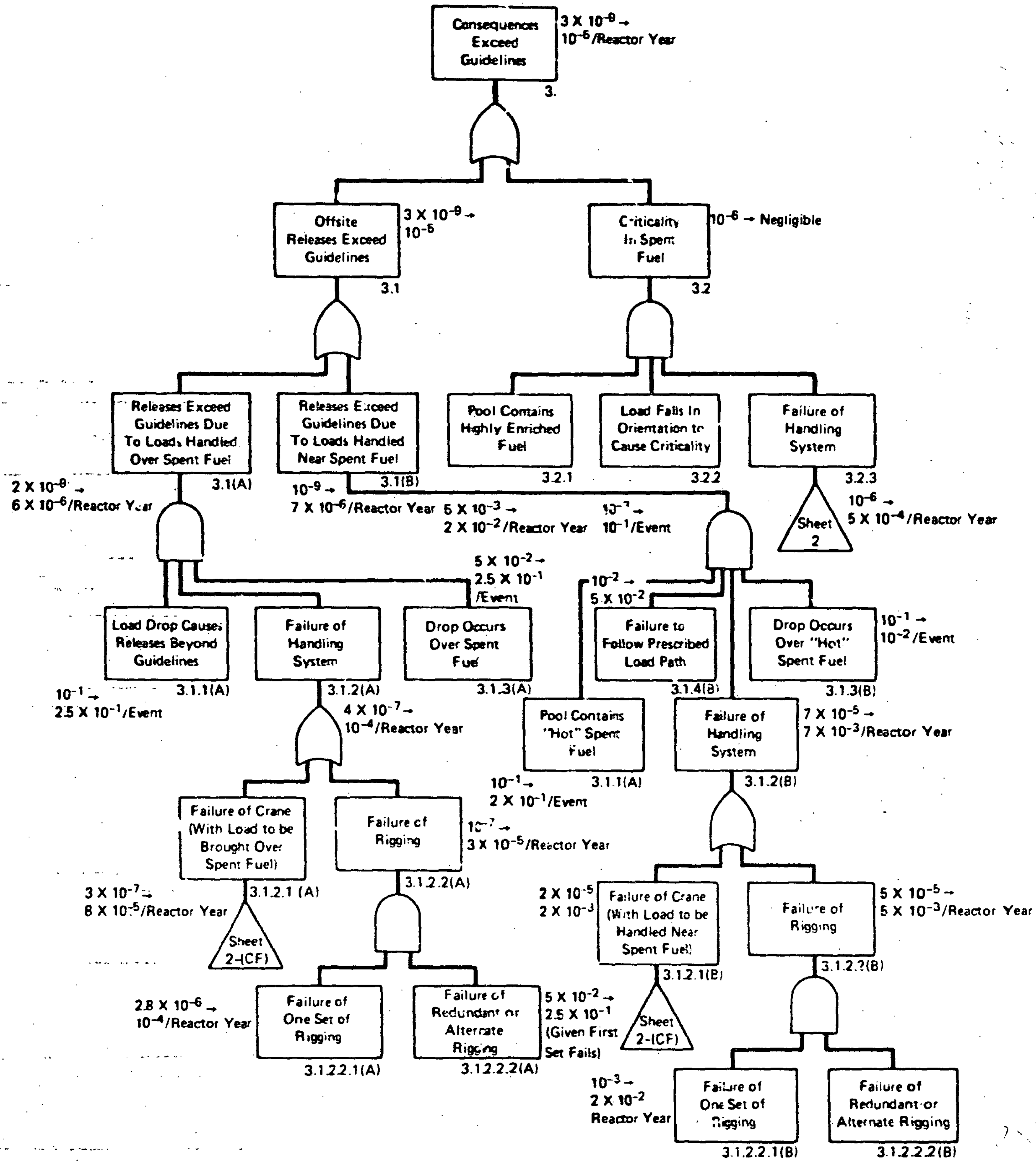


FIGURE B-2 FAULT TREE FOR LOADS HANDLED OVER SPENT FUEL POOL



1/ For Some Loads Safe Load Paths are Defined That Keep Loads Away From Spent Fuel Even if a Single Failure Proof Crane is Provided. This is Depicted by Branch 3.1(B). Certain Other Loads Must be Carried Over Spent Fuel; This is Depicted by Branch 3.1(A) of This Fault Tree.

FIGURE B-3 FAULT TREE IF A SINGLE-FAILURE-PROOF HANDLING SYSTEM IS USED[1]

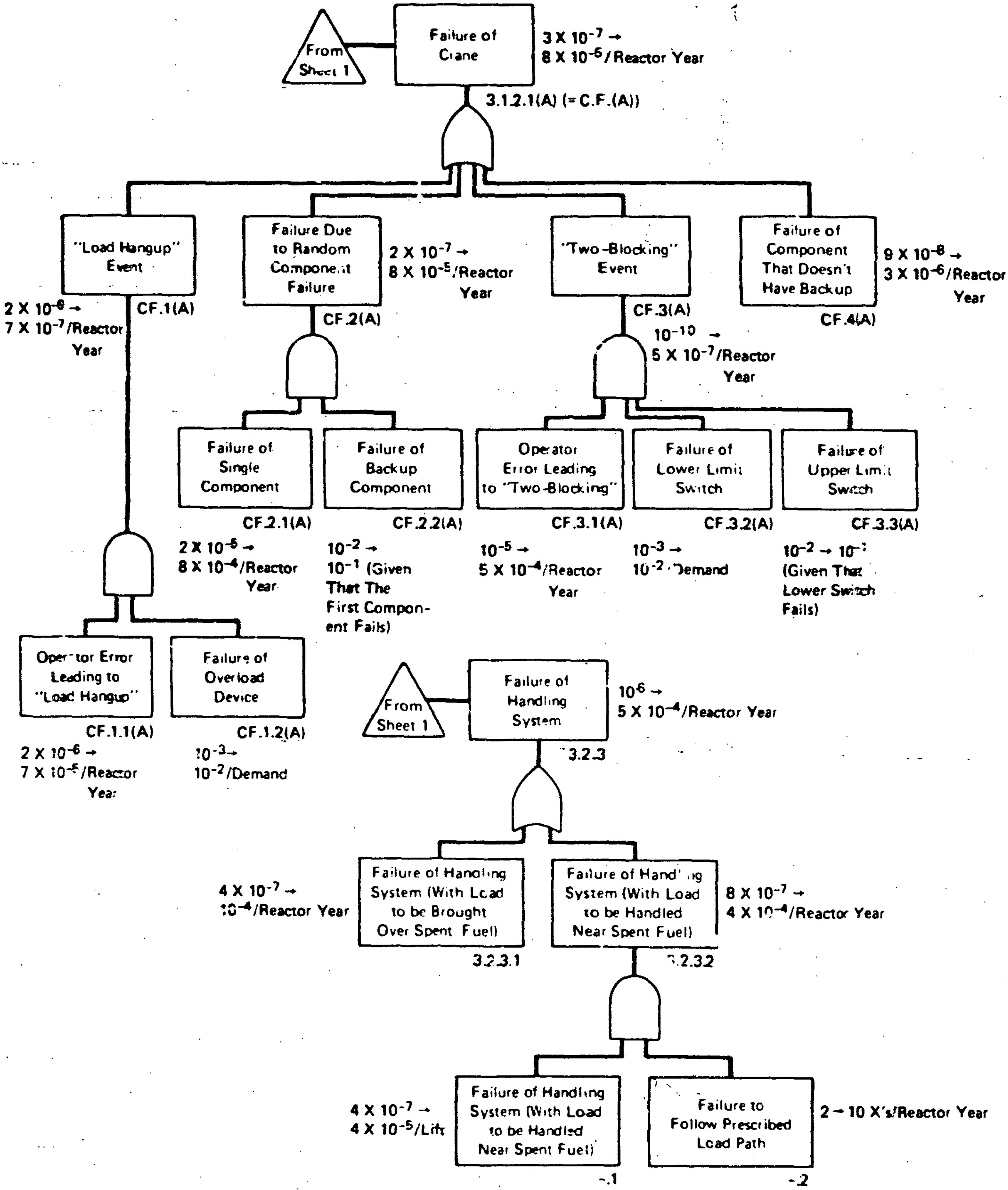


FIGURE B-3 SHEET 2 (A)

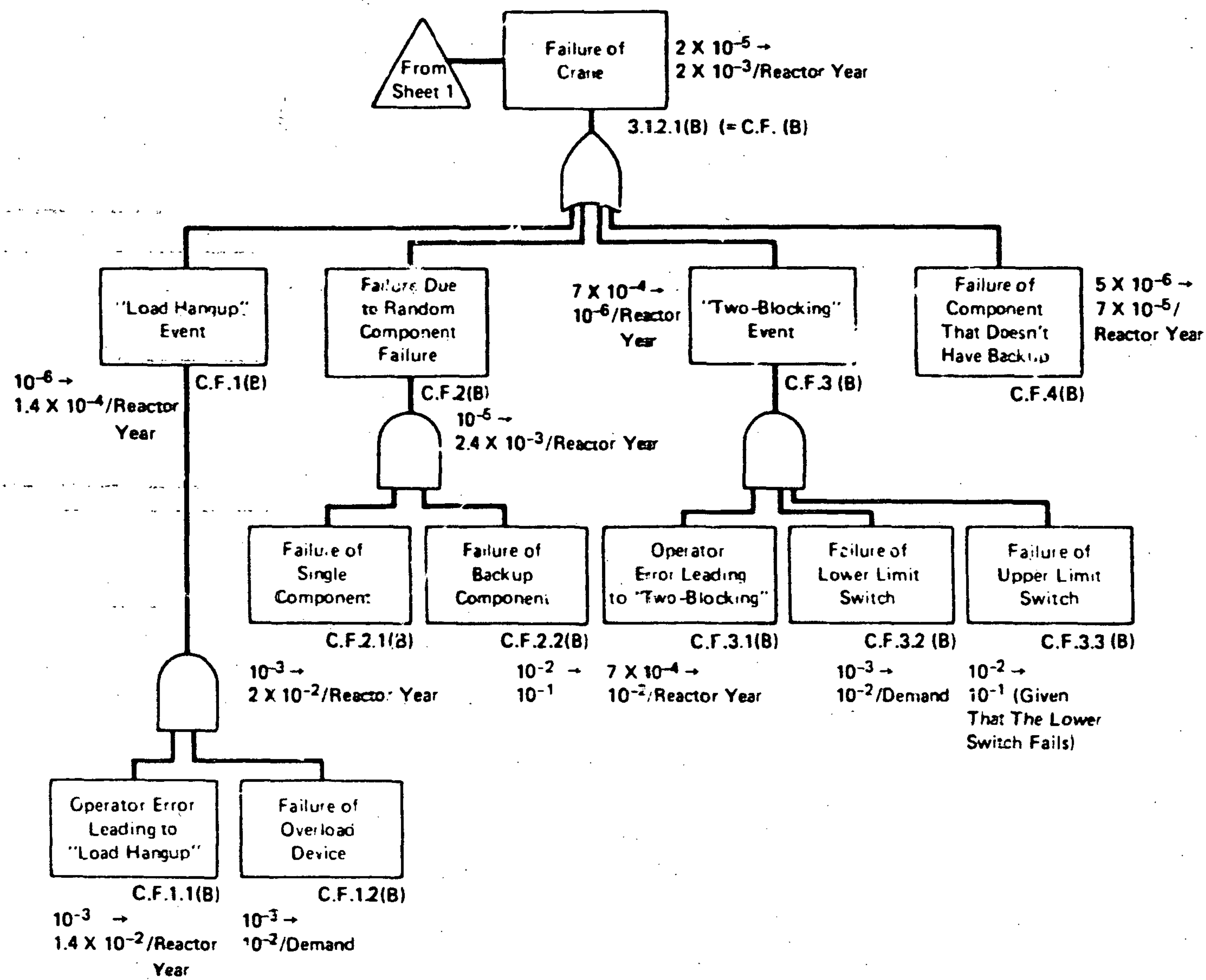


FIGURE B-3 SHEET 2 (B)

APPENDIX C

MODIFICATION OF EXISTING CRANES

The safe operation of cranes is necessary when hoisting or transferring loads that could cause the direct or indirect release of radioactivity if the load was dropped due to malfunction or failure of the crane. The guidelines of this report provide various alternative means of assuring safe crane operation. One of these alternatives includes design of a crane using conservative design safety factors in the structural members that are affected by the lifted load, by careful attention to material properties and by using dual or diverse components and circuits for the reeving system and for controls for travel limits and other systems intended to protect against adverse crane operation that would affect the crane's ability to stop and hold the load safely.

NUREG-0554 Guidelines

A report entitled "Single-Failure-Proof Cranes," NUREG-0554, has been published providing guidelines that incorporate the above philosophy. Although titled "Single-Failure-Proof Cranes," certain components are allowed to not have a redundant counterpart if a sufficient design safety factor is used. These components are typically ones that are not susceptable to wear or degradation.

Comments from industry on the contents of the report were enlisted by the NRC prior to final issuance. This resulted in many recommendations for changes which were considered, and many of which were incorporated in the report. The guidelines of NUREG-0554 are, briefly stated:

(1) The allowable stress limits should be identified and be conservative enough to prevent permanent deformation of the individual structural members when exposed to maximum load lifts.

(2) The minimum operating temperature of the crane should be determined from the toughness properties of the structural materials that are stressed by the lifting of the load.

(3) The crane should be capable of stopping and holding the load during a seismic event equal to a Safe Shutdown Earthquake (SSE) applicable to that facility.

(4) Automatic controls and limiting devices should be designed so that component or system malfunction will not prevent the crane from stopping and holding the load safely.

(5) Design of the wire rope reeving system should include dual wire ropes.

(6) Sensing devices should be included in the hoisting system to detect such items as overspeed, overload, and overtravel and cause the hoisting action to stop when limits are exceeded.

(7) The reeving system should be designed against the destructive effects of "two-blocking."

(8) The hoisting drum(s) should be protected against dropping should its shafts or bearings fail.

(9) Safety devices such as limit switches provided to reduce the likelihood of a malfunction should be in addition to those normally provided for control of maloperation or operator error.

(10) The crane system should be given a cold proof test if material toughness properties are not known.

As a result of public comments, the recommendations in the NUREG-0554 report include some alternate solutions where a direct compliance with some of the recommendations would be difficult or impractical to follow. Furthermore, for some power plant layouts it may be acceptable to relax some safety feature if an equivalent degree of safety can be obtained by adding other features such as a higher design safety factor for a load-bearing component.

Implementation of NUREG-0554 For Operating Plants

In the case of a new crane, all the recommendations contained in NUREG-0554 should be followed; however, in the case of an existing crane that is to be upgraded to the guidelines of Section 5.1.6, space economies for the crane may not allow ready application of all the safety features to the crane. Additionally, application of certain other features may not be practical since they would require replacement of certain components whose adequacy can be verified by alternative measures. Thus, certain adjustments may be necessary to compensate for those features that will not be included. The following identify alternatives that may be used for certain applications when upgrading an existing crane in lieu of complying with certain recommendations of the NUREG-0554.

(1) Paragraph 2.2 of NUREG-0554 recommends that the crane be designed to the MCL (Maximum Critical Load, defined in NUREG-0554) but that those component parts that are subject to wear or degradation be designed to a greater load to prevent the load-handling safety factor to drop below the MCL rating due to wear between maintenance periods. However, a specific application was accepted in which the wear susceptible components were designed to the MCL rating and not to a greater load rating to allow for wear.

Although the recommendation to design certain components to a greater load rating was not met, an equivalent margin of safety was achieved because the drive gear contained a torque limiting device that with the proper setting effectively limits the load which the wire rope and other wear susceptible components will experience. (NOTE: an overload sensing device that has been energized in order to stop the electric drive motor would not have adequately accomplished this due to the time delay inherent in such a device).

(2) Paragraph 2.4 of NUREG-0554 recommends a coldproof test as an alternate method of assuring absence of brittle-fracture tendency in lieu of material testing for cranes that are already built and operating. For a modified crane in an operating plant, the coldproof test was omitted because the

minimum ambient temperature was 70°F (21°C), which exceeded the NDTT + 60°F requirement (of paragraphs NC-2300 and ND-2300 of Section III of the ASME Code) for most structural steels of comparable dimensions (NDTT is the "nil-ductility transition temperature").

(3) Paragraph 2.8 of NUREG-0554 recommends that preheat and postweld heat treatment temperatures be specified in the weld procedure. For a modified crane in an operating plant, the weldments may not have been heat treated in accordance with Subarticle 3.9 of AWS D1.1, "Structural Welding Code." As a substitute for weld heat treatment of crane structures already built or in operation, the welds whose failure could result in the drop of a critical load should be nondestructively examined to ascertain that the weldments are acceptable.

(4) Paragraph 4.1 of NUREG-0554 recommends that fleet angles in the wire rope reeving system be limited to prevent excessive wear on the wire rope. Larger than recommended fleet angles have been accepted for an application where space limitation prevented the use of larger sheaves. However, a more frequent inspection program was included to assure the continued integrity of the wire rope.

(5) Paragraph 4.3 recommends that the load blocks have two attachment points or hooks. Because of an existing building height limitation and difficulty in getting sufficient lift height at one installation being upgraded to single-failure-proof criteria, a single attachment sister hook was accepted. However, the safety factor was increased to 10:1 to compensate for loss of the single-failure-proof feature and to equal the total safety factor for the wire rope.

(6) Paragraph 4.9 of NUREG-0554 recommends that the hoist holding brake system be single-failure-proof. Normally the holding brakes are located near the motor drive in order to reduce the size of the brake unit, and consequently the gears or transmissions interposed between the motor and the hoist drum must be of dual design to be single-failure-proof. Omission of a second gear train has been accepted for cranes where two emergency brakes were applied directly to the hoisting drum, thus eliminating the need of the dual gear trains to provide assurance that the load will be safely held in case of a single failure.

(7) Paragraph 4.5 of NUREG-0554 recommends load hangup protection. However, a system of interlock circuitry preventing movement of the trolley and the bridge while hoisting the load has been accepted in lieu of load hangup protection.

(8) Paragraph 8.3 of NUREG-0554 recommends that if the design includes an energy-controlling device between the load and head blocks a test be made to verify the hoisting machinery's ability to withstand a "two-blocking" event.

As an alternative to designing to withstand a "two-blocking" event, Paragraph 4.5 of NUREG-0554 allows the crane to be furnished with two independent travel limit switches. If this alternative is selected, the "two-blocking" test should be verification of the proper functioning of these switches. In addition, some cranes are furnished with a load-limiting device (e.g., strain gage, etc.) that will automatically protect the reeving system. If such a load limiting device is used, a substitute for the "two-blocking" test may be made that demonstrates the proper functioning of the load limiter.

(9) Paragraph 8.3 of NUREG-0554 also recommends a load hangup test. Where interlock circuitry is provided in lieu of load hangup protection, testing should be performed to verify the operability of the interlocks.

Impact

Non-single-failure-proof cranes already in use at operating plants typically meet several of the above guidelines or alternatives. Modification of existing cranes to satisfy NUREG-0554 will vary from plant to plant but generally will not require complete replacement of the crane. Modifications would involve use of techniques and components that are readily available. For certain cranes to meet the guidelines or alternatives to NUREG-0554, modifications may be limited to addition of a double reeving system to the existing trolley. However, for other cranes, the entire trolley may need to be replaced. Such trolleys are commercially available as a retrofit unit, and at least one manufacturer's unit may be disassembled so as to permit it being transferred into areas with limited entrance sizes, such as a containment building equipment hatch.

The cask handling cranes at several operating power plants have already been upgraded or are planned to be upgraded to meet single-failure-proof criteria as shown in Table 3.2-3 of Section 3.2.

Implementation of the guidelines of Section 5.1 will require that loads handled by PWR polar cranes and, at certain plants, loads handled by turbine building cranes be evaluated for potential consequences if dropped. If evaluation criteria of 5.1 are not met, it may require that these cranes be upgraded at certain plants to single-failure-proof criteria.

APPENDIX D

REFUELING OPERATIONS

CRANE TRAVEL - FUEL BUILDING

LIMITING CONDITION FOR OPERATION

3.9.7 Loads carried over the spent fuel pool and the heights at which they may be carried over racks containing fuel shall be limited in such a way as to preclude impact energies over 240,000 in.-lbs., if the loads are dropped.

APPLICABILITY: With fuel assemblies and water in the storage pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 The potential impact energy due to dropping the crane's load shall be determined to be less than or equal to 240,000 in.-lbs. prior to moving each load over racks containing fuel.

APPENDIX E

REFERENCES

1. ANSI B30.2-1976, "Overhead and Gantry Cranes," The American Society of Mechanical Engineers, New York, NY.

2. ANSI B30.9-1971, "Slings," The American Society of Mechanical Engineers, New York, NY.

3. ANSI N14.6-1978, "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials," American National Standards Institute, New York, NY.

4. "Specifications for Electric Overhead Travelling Cranes," CMAA - 70 - 1975, (supersedes EOCI - Specification 61). Available from Crane Manufacturers Association of America, Pittsburgh, PA, copyrighted.

5. U.S. Nuclear Regulatory Commission, "Single-Failure-Proof Cranes for Nuclear Power Plants," USNRC Report NUREG-0554, May 1979.*

6. U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition," USNRC NUREG-75/087, Section 9.1.2, "Spent Fuel Storage."*

7. U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition," USNRC NUREG-75/087, Section 9.1.4, "Fuel Handling System."*

8. U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition," USNRC NUREG-75/087, Section 13.1.3, "Qualifications of Nuclear Plant Personnel."*

9. U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition," USNRC NUREG-75/087, Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents."*

10. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.13, Revision 1, "Spent Fuel Storage Facility Design Basis."**

11. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."**

12. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)."**

* Available for purchase from GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 and/or National Technical Information Service, Springfield, Virginia 22161.

** Available for purchase from GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.