NUREG/CR-1697
ORNL/TM-439

Safety Review of the Design, Operation, and Radiation Sections of the General Electric Morris Operation Consolidated Safety Analysis Report

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Prepared for
U.S. Nuclear Regulatory
Commission

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Washington, D. C. 20555

Printed copy price: $2.75

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Springfield, Virginia 22161

NUREG/CR-1697
ORNL/TM-439
Dist. Category AN

CHEMICAL TECHNOLOGY DIVISION

SAFETY REVIEW OF THE DESIGN, OPERATION, AND RADIATION SECTIONS OF THE GENERAL ELECTRIC MORRIS OPERATION CONSOLIDATED SAFETY ANALYSIS REPORT

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Manuscript Completed: January 1981
Date Published: July 1981

Prepared for
U.S. Nuclear Regulatory Commission
Office of Nuclear Material Safety and Safeguards
Division of Fuel Cycle and Material Safety
Washington, D.C. 20555

NRC FIN No. B0102

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SAFETY REVIEW OF THE DESIGN, OPERATION, AND RADIATION SECTIONS OF THE GENERAL ELECTRIC MORRIS OPERATION CONSOLIDATED SAFETY ANALYSIS REPORT

J. P. McBride

ABSTRACT

A safety review was made of Sections 4 through 9 of the Consolidated Safety Analysis Report (CSAR) for the GE Morris Operation spent-fuel storage facility. The sections reviewed include Design Criteria and Compliance, Facility Design and Description, Radiation Protection, Accident Analysis, and Conduct of Operations. The safety review was performed in accordance with the *Code of Federal Regulations*, Title 10, Part 72, "Licensing Requirements for the Storage of Spent Fuel in an Independent Spent Fuel Storage Installation" and contains independent estimations of source terms and dose-commitments from postulated accidents in the storage facility and a structural analysis of the Morris Operation cranes as an appendix. The review confirms that the features of the facility as described in Sections 4 through 9 of the CSAR fulfill the safety requirements of 10 CFR 72, and it is concluded that spent-fuel handling and storage at the Morris Operation do not present significant risks to public health and safety.

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1. INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) in May 1979 requested the Oak Ridge National Laboratory to participate in a safety review of the General Electric Morris Operation as described in the revised Consolidated Safety Analysis Report (CSAR) NEDO-21326C (January 1979).[1] The GE Morris Operation is an irradiated-fuel storage facility located near Morris, Illinois. The scope of ORNL's participation in the safety review covered the following sections of the CSAR:

1. Section 4: Design Criteria and Compliance

2. Section 5: Facility Design and Description

3. Section 7: Radiation Protection

4. Section 8: Accident Safety Analysis

5. Section 9: Conduct of Operations

The Morris Operation was built as an integral part of the GE Midwest Fuel Recovery Plant (MFRP), Docket No. 50-268, and was licensed for the receipt of spent fuel in December 1971. The MFRP Construction Permit (No. CPCSF-3) was terminated on August 23, 1974, but the Materials License No. SNM-1265 for the receipt and storage of spent fuel containing up to 100 metric tons (MT) of heavy metal was continued. Spent-fuel storage capacity was subsequently increased from 100 MT of heavy metal to 750 MT.[2] The revised CSAR is a consolidation of all safety analysis information submitted by the General Electric Company relating to the receipt, storage, and transfer of irradiated nuclear fuel at the Morris Operation and disregards those features of the MFRP not applicable to fuel storage.

The present review was performed in accordance with the *Code of Federal Regulations*, Title 10, Part 72 (10 CFR 72)[3] and includes independent estimations of source terms and dose commitments from postulated accidents in the storage facility. A structural analysis of the Morris Operation cranes is included as an Appendix. A site visit to the Morris Operation was made as a part of the safety review.

2. GENERAL DESCRIPTION

The General Electric facilities are located near Morris, Illinois, adjacent to the Dresden Nuclear Power Station (DNPS), and consist of the Morris Operation buildings and a training center for nuclear power station operators - the Boiling-Water Reactor Training Center (BWRTC).* The Morris Opertion is licensed for the receipt, storage, and transfer of nuclear fuel from boiling-water reactors (BWRs) and pressurized water reactors (PWRs). Operations related to the maintenance of General Electric fuel shipping casks are also conducted at this site.

The Morris Operation fuel storage facility includes three interconnected water-filled basins with cranes, water treatment systems, and other facilities required to receive irradiated fuel and store it underwater for an indefinite period. The fuel storage equipment in the basins is designed to protect the integrity of the fuel rods during seismic or meteorological events. Special procedures and isolation can be provided for storage of damaged or leaking fuel. Security measures are in effect to protect the facility against unauthorized access. Based on the service life of nonreplaceable components (concrete basin and basin liner), the normal service life of the facility would be more than 100 years, although it is intended for interim storage only.

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* The Morris Operation does not encompass BWRTC activities, although both are General Electric operations.

In December 1975, General Electric was authorized by the NRC to increase the fuel storage capacity of the facility from about 100 MT of heavy metal to 750 MT through the installation of a newly designed fuel storage system and appropriate changes in fuel handling and support systems. This project converted the former high-level waste storage basin to a fuel storage basin. The capacity-expansion project was completed in 1976.

3. DESIGN CRITERIA AND COMPLIANCE

The Morris Operation is designed to store in a water basin irradiated, light-water-reactor fuel from nuclear power stations. The design basis fuel in the CSAR was UO_2, with an initial enrichment of 5% ^{235}U or less, clad in stainless steel, zirconium, or Zircaloy. The fuel was assumed to have been irradiated at power loads up to 40 kW/kg U to an exposure of 44,000 MWd/MTU and cooled 90 days. Fuel, typically received, had exposures of 33,000 MWd/MTU or less and much longer cooling times. As of February 1, 1977, the average exposure of stored fuel was less than 1500 MWd/MTU, and the average cooling time was about 4.7 years. Part 72 stipulates that spent fuel must undergo at least one-year's decay prior to storage in an independent spent fuel storage installation. Hence, independent analyses at ORNL, conducted as part of the safety review, are based on PWR fuel irradiated at 40 kW/kg U to an exposure of 44,000 MWd/MTU and cooled 1 year. Estimates of the radionuclide activity in the model spent fuel, made using the ORIGEN code,[4] are given in Table 1.

3.1 General Design Criteria

Structures, systems, and components important to safety are designed to withstand the effects of natural phenomena (i.e., tornadoes, including tornado winds, hurricanes, earthquakes, etc.), without impairing their capability for safe shutdown, radioactive inventory control, and the prevention of significant radiation exposure to operating personnel or the general public.

The design criteria presented in the CSAR were evaluated against the requirements set forth in Subpart F, "General Design Criteria," of 10 CFR 72. We found that the design criteria fulfilled those requirements.

Table 1. Radionuclide activity in spent fuel after one-year decay[a]

Conditions: Burnup - 44,000 MWd/MT
Specific power - 40 kW/kg

Isotope	Half-life	Activity (Ci/MT)
	Fission products	
Kr-85	10.7 y	10,880
Sr-89	50.5 d	5,267
Sr-90	29.1 y	87,430
Y-90	64.0 h	87,450
Y-91	58.5 d	14,220
Zr-95	64.0 d	31,920
Nb-95m	86.6 h	237
Nb-95	35.2	69,330
Tc-99	2.13×10^5 y	16.6
Ru-103	39.3 d	3,015
Rh-103m	56.1 m	2,718
Ru-106	1.01 y	407,500
Rh-106	29.9 s	407,500
Ag-110m	249.8 d	3,019
Ag-110	24.6 s	40.2
Cd-113m	14.6 y	87.3
Cd-115m	44.6 d	8.1
Sn-119m	245.0 d	118
Sn-123	129.1 d	642
Sb-124	60.2 d	35.7
Sb-125	2.77 y	15,640
Te-123m	119.6 d	4.5
Te-125m	58 d	3,808
Te-127m	109 d	1,829
Te-127	9.35 h	1,792
Te-129m	33.6 d	30.1
Te-129	69.6 m	19.6
I-129	1.57×10^7 y	0.0426
I-131	8.04 d	-
Xe-131m	11.9 d	-
Cs-134	2.06 y	183,600
Cs-137	30.0 y	135,000
Ba-137m	2.55 m	127,700
Ba-140	12.8 d	0.005
La-140	40.2 h	0.005

Table 1. (continued)

Isotope	Half-life	Activity (Ci/MT)
Fission products		
Ce-141	32.5 d	721
Ce-144	284.2 d	556,100
Pr-143	13.6 d	0.013
Pr-144m	7.2 m	6,675
Pr-144	17.3 m	556,100
Nd-147	11.06 d	-
Pm-147	2.62 y	101,000
Pm-148m	41.3 d	61.3
Pm-148	5.37 d	3.5
Sm-151	90.0 y	416
Eu-154	8.6 y	16,140
Eu-155	4.96 y	9,345
Eu-156	15.2 d	0.02
Tb-160	72.3 d	58.5
Total activity		2.85×10^6
Transuranics		
Np-239	2.35 d	43.4
Pu-241	14.4 y	155,960
Am-241	432 y	383
Am-243	7378 y	43.4
Cm-242	163 d	15,050
Cm-243	28.5 y	40.4
Cm-244	18.1 y	7,076
Total activity		1.79×10^5

[a] A. G. Croff et al., *Revised Uranium Plutonium Cycle PWR and BWR Models for the ORIGEN Computer Code*, ORNL/TM-6051 (September 1978).

3.2 Tornado Design Criteria

Plant structure and components are designed to withstand wind velocities of 177 km/h (110 mph) without impairment of safety-related functions. They are also designed to withstand the effects of potential windborne missiles and short-term wind velocities of 483 km/h (300 mph) with pressure differentials of up to 3 psi without damage to fuel in storage to an extent significantly affecting public health and safety.

3.3 Seismic Design Criteria

The main building, including all portions of the structure now used for irradiated fuel storage, was constructed to seismic specifications for a design-basis earthquake (as defined in the CSAR) of 0.1 x g and a maximum (safe shutdown) earthquake of 0.2 x g. The response criteria used in the design calculations are based on the north-south component of the 1940 El Centro, California earthquake normalized to horizontal ground accelerations of 0.1 x g for the design-basis earthquake and 0.2 x g for the maximum earthquake, thus fulfilling the requirements of Appendix A, 10 CFR 100, for the siting of nuclear power plants and the licensing requirement for seismic design of 10 CFR 72.*

3.4 Compliance

Structural and system resistance to tornado and seismic phenomena were evaluated as part of the safety analyses of the MFRP[5] and were <u>not</u> reevaluated as part of this safety review. However, NRC requested that ORNL perform an independent analysis to determine the effects of a maximum or safe shutdown earthquake on the cask handling crane and support structure and the storage basin crane. The analysis is presented in the Appendix. The results indicate that these systems meet the structural requirements to successfully withstand the stresses induced by a safe shutdown earthquake.

---

*The maximum earthquake is equated in Appendix A, 10 CFR 100, with the "Safe Shutdown Earthquake" (SSE) of a nuclear power plant. Unlike the CSAR use of the term and according to Appendix A, the SSE is also commonly referred to as the "Design Basis Earthquake" (DE). An acceptable seismic criterion in 10 CFR 72 for an independent spent-fuel storage installation (ISFSI) evaluated under the criteria of Appendix A, 10 CFR 100, is that the ISFSI-DE be equivalent to the SSE of a nuclear plant (i.e., in the case of the Morris Operation, the maximum earthquake).

Storage of spent nuclear fuel at the Morris Operation was initiated in January 1972. An evaluation of the design and operation of that facility, based on experience obtained in spent-fuel storage during the following 7 years, appears in a recent report (July 1979) prepared for the NRC by Pacific Northwest Laboratory.[6] The following passage is taken from the abstract of that report:

> The purpose of this report is to provide a description of spent fuel handling activities and systems, and an analysis of the storage performance as developed over the seven year operational history of the Morris Operation.
>
> Design considerations and performance are analyzed for both the basin and key supporting systems. The bases for this analysis are the provisions for containing radioactive by-product materials, for shielding from the radiation they emit, and for preventing the formation of a critical array.
>
> These provisions have been met effectively over the history of storage at Morris. The release of radioactive materials is minimized by the protection of the cladding integrity, the containment of the basin water, the removal of radioactive and other contaminants from the water, and by filtering and then dispersing the basin air.

4. FACILITY DESIGN AND DESCRIPTION

All radioactive material handling related to fuel storage at the Morris Operation is in facilities located within a protected area. No radioactive liquid effluents are released to the environs, and no burial of radioactive or contaminated material occurs on the tract. The only radioactive or contaminated waste materials leaving the site are effluents vented through the ventilation stack or solid low-level radioactive wastes that are shipped offsite. Offsite shipments are made in accordance with applicable U.S. Nuclear Regulatory Commission, U.S. Department of Transportation, and other State and Federal regulations.

The principal structure at the Morris Operation site is the main building. This safety analysis is concerned only with the use of this structure for fuel receipt, storage and shipment. The fuel storage operations utilize the following portions of the main building:

1. cask receiving and decontamination areas,

2. cask unloading basin

3. fuel storage basins,

4. basin support systems (water cooling, filtration, etc.), and

5. control room.

Irradiated nuclear fuel is received at the Morris Operation in shielded shipping casks which are designed, loaded, and transported in accordance with regulatory requirements of the U.S. Department of Transportation. Prior to shipping, fuel is inspected for defects; known defective fuel is not normally accepted by Morris Operation. Prior to unloading the fuel, the casks are decontaminated and flushed to detect any damaged fuel and then lowered into the cask unloading basin. Fuel is unloaded under a minimum of 2.74 m (9 ft) of water and placed in stainless steel basket assemblies designed to protect the fuel from physical damage (four PWR elements or nine BWR elements per basket assembly), maintain the fuel in a subcritical configuration, and permit the transport of the fuel to the storage basins. A doorway guard at the entrance to the storage basin prevents the basket from tipping and discharging the fuel. Basket mounting provisions in the fuel basins provide seismological restraints.

The basins were constructed below ground with stainless-steel-lined, reinforced-concrete walls about .61 m (2 ft) thick poured in contact with the sides of a bedrock excavation. The south wall of the basin is about 1.22 m (4 ft) thick, because it was intended to stand independent of the surrounding rock to facilitate possible future expansion. A leak detection system and pump-out facilities are provided for interconnected channels spaced at regular intervals between the concrete walls and floor and the stainless steel liner.

A ventilation system is provided that is designed so that air passes sequentially from areas of low contamination potential to areas of higher contamination potential and thence through a sand filter and the 91.44-m (300-ft) stack. Special vent hoods are available for fuel bundles containing defective fuel rods to collect escaping gases, which are filtered and then vented via the stack.

Basin water is circulated through a system that reduces radioactive contamination by ion exchange and filtration. The system includes a vertical pressure shell containing a precoated leaf filter. The filter screens are precoated with a cellulose fiber filter aid and then overcoated with either Powdex anion-cation resin or Zeolon, a synthetic alumino-silicate molecular sieve, for removal of both particulates and soluble ions. A suction system is provided to clean the basin floors and remove floating debris. Radioactive materials are collected and stored in the low-activity-waste (LAW) vault.

The underground LAW vault [14.0 m (46 ft) in diameter and 23.5 m (77.2 ft) deep] is constructed of steel-lined, reinforced concrete about 0.61 m (2 ft) thick, poured directly against excavated rock. A vault cover of reinforced concrete is provided. An inner steel tank [11.7 m (38.5 ft) in diameter and 21.0 m (69 ft) deep], having a capacity of 2.27×10^6 L (600,000 gal), is the waste receptacle within the vault. A second, underground waste-storage vault — a stainless-steel-lined, reinforced-concrete cylinder, identified in the CSAR as a cladding vault — which may also be used for low-level-waste storage, is also located onsite. Liquid from this vault can be transferred to the LAW vault. The material in the LAW vault can be pumped to an evaporator located in the canyon area of the reprocessing plant. The overhead vapor from the evaporator is routed through the sand filter and discharged up the stack, while the concentrate is returned to the LAW vault where solids precipitate as the solution temperature drops.

The safety evaluation report of ref. 2 approved the present configuration of the Morris Operation provided certain structural features not then in existence be incorporated in the design. These features included new storage baskets and racks, structural modifications associated with cask handling and unloading, and changes in the basin water cooling and cleanup systems. For the most part these features were proposed by and incorporated in the present design by the applicant and approved by the NRC. Our review confirms that these and the other features of the facility as described in the CSAR fulfill the safety requirements of 10 CFR 72; hence, we are able to conclude that spent-fuel handling and storage at the Morris Operation do not present significant risks to public health and safety.

4.1 Utility and Support Systems

Utility and support systems provided for the MFRP are utilized for the operation of the spent-fuel storage facility. A deep well is the normal supply for the potable, utility, demineralized, and fire-protection water systems. A second well, furnishing 113.6 L/min (30 gal/min) to the adjoining Boiling-Water Reactor Training Center, is interconnected with the Morris Operation system and is used as a backup supply. Natural gas is supplied for the steam system by the local utility. Electrical power is furnished by Commonwealth Edison Company through two separate 34,000-V transmission lines connected to two onsite transformer systems. While loss of electrical power would not result in unsafe conditions, a diesel-driven, standby generator is available to supply power for essential services in the event that both of the incoming power sources from the utility are lost. A sanitary sewer system is routed to lagoons for treatment prior to chlorination and release to the river. An industrial sewer system, meeting state requirements for release, also discharges to the Kankakee River. Chemical wastes having concentrations above discharge limits are sent to an onsite, earth-diked evaporation pond having no discharge.

4.2 Ventilation System

The ventilation system for the Morris Operation has been maintained as installed for the fuel recovery operation. Fresh air supplied to the cask decontamination area and the basin area flows through the "canyon" and the process cells into an exhaust duct, through a sand filter, and is discharged through a 91.4-m (300-ft) stack to the atmosphere. Canopy hoods for placement over possible leaking fuel elements in storage also vent to the canyon. The ventilation system is considered acceptable for the spent fuel storage operation.

4.3 Waste Handling

Liquid and slurry wastes generated by the Morris Operation consist largely of cask coolant, decontamination solutions, and ion exchange resins and filter media from the pool water cleanup system. These are transferred to the LAW vault, where the supernatant liquid is routinely concentrated for volume reduction by an evaporator and the vapor discharged through the sand filter and up the monitored 91.4-m (300-ft) stack. Periodic flushing of the evaporator boiler with water and/or acid and the return of the solutions and suspended solids to the LAW vault during normal operations prevent the accumulation of significant levels of radioactivity in the evaporator.[7] A decision on the ultimate disposal of the accumulated solid material in the LAW vault has not been made.[7]

The low-level solid wastes generated in cask decontamination, laboratory operations, and other work at the site are packaged in drums and shipped to a commercial waste burial site. Management of these wastes is consistent with plans previously reviewed and approved for the spent fuel recovery operation.[8]

5. RADIATION PROTECTION

During the site visit to Morris Operation, special attention was given to the measures being taken for the confinement of radioactivity and minimization of personnel exposure. Plant management provided charts demonstrating the continued ability of the basin water decontamination system to quickly remove any unusual release of radioactivity to the storage pool and routinely maintain contamination levels below the occupational maximum permissible concentrations (MPCs) of 10 CFR 20 (ref. 9). We observed that radiation alarm and monitoring systems were adequate and well-placed and access to areas of potential contamination were controlled. As a result of the site visit and evaluation of the measures for radiological protection and effluent control given in the CSAR, it is our opinion that routine operation of the facility does not present a significant radiological health or safety risk and fulfills all the requirements for radiological protection of 10 CFR 72.

6. ACCIDENT SAFETY ANALYSIS

The applicant has calculated the possible consequences from the following postulated incidents:

1. Cask Drop in Unloading Basin
2. Basin Leakage
3. Loss of Basin Cooling
4. Cooling System Leak
5. Low-Activity-Waste Vault Leakage
6. Missile Impact on Basin Structure
7. Fuel Bundle Drop
8. Fuel Basket Drop
9. Tornado-Generated Missile
10. Criticality

The cask drop and basin leakage were previously evaluated by the NRC staff.[2] We reviewed the CSAR evaluation of the cask drop, basin leakage, loss of basin cooling, cooling systems, and LAW leakage and agree that there would be no danger to the health and safety of plant personnel or general public should any of these incidents occur. We agree with the conclusion in the CSAR that the penetration of the basin liner by a tornado-generated missile is unlikely; in any event, the ability of the Morris Operation to detect and expeditiously repair such leaks has been demonstrated (cf. CSAR, Sect. 8.3, p. 8-5).

The radiological consequences of postulated accidents that could result in offsite radioactive releases were independently evaluated as given below. The stored fuel is presumed to have a burnup of 44,000 MWd per metric ton of heavy metal at 40 kW/kg U and cooled 1 year. The ORIGEN code[4] was used to estimate the radionuclide activity in the spent fuel (Table 1). Our calculations assumed ground-level releases, a wind speed of 1 m/s and Pasquill diffusion category F. Exposures (50 year dose commitments) were estimated using the AIRDOS code[10] at the 150-m (500-ft) site boundary (i.e., the country road; cf. Fig. 3-3 and Sect. 3.2.2.4, pp. 3.4 and 3.7, respectively, of the CSAR); and at 800 m (2600 ft), the nearest permanent residence. The estimated X/Q's at 150 and 800 m (500 and 2600 ft) were 5.6×10^{-3} and 2.4×10^{-4}, respectively, for noble gases and 3.4×10^{-3} and 5.8×10^{-5}, respectively, for iodine, where a deposition velocity of 0.01 m/s was assumed.

6.1 Fuel Bundle Drop

It is assumed that all the rods in a dropped PWR fuel bundle were ruptured, releasing all fission gases in the plenum (30% of the ^{85}Kr and 10% of the iodine in the fuel bundle).[11] Curies released, assuming that none of the ^{85}Kr and 99% of the iodine dissolve in the basin water,[11] would be about 1510 curies of ^{85}Kr and 1.97×10^{-5} curies of ^{129}I. Estimated maximum exposures at the site boundary, 150 m (500 ft), are 4.0 mrem whole body and 0.11 mrem thyroid. These exposures are fractions of 8.0×10^{-4} and 2.2×10^{-5}, respectively, of the 5-rem limit for the dose commitments to offsite individuals from design-basis accidents given in 10 CFR 72. Estimated individual whole body and thyroid doses at the nearest permanent residence, 800 m (0.50 mi), would be 0.17 mrem and much less than 0.01 mrem, respectively.

6.2 Fuel Basket Drop

The maximum drop of a fuel storage basket in the unloading pit of the storage pool is 6.86 m (22.5 ft) in water. It is unlikely that in the event a basket containing spent fuel (four PWR or nine BWR fuel bundles) is dropped that the fuel liner will be penetrated or the fuel rods in the fuel bundle ruptured. However, for the purpose of establishing an upper limit to the radiological consequences of such an accident, it is assumed that all of the fuel rods in the contained spent fuel (four PWR fuel bundles) rupture and the plenum gases are released to the basin water. The curies released would be four times those assumed for the single-fuel bundle drop discussed above (i.e., ^{85}Kr, 6040 curies and ^{129}I, 7.88×10^{-5} curies). The estimated maximum, individual whole-body and thyroid exposures would be 16.0 and 0.44 mrem, respectively, at the 150-m site (500-ft) boundary and 0.68 and about 0.01 mrem, respectively, at 800 m (2600 ft).

6.3 Tornado-Generated Missile

It is postulated that a tornado-generated missile has the potential of impacting as many as six BWR or four PWR fuel assemblies and that the rods in these assemblies would fail. Assuming that a basket containing four PWR fuel assemblies was struck, the releases and exposures would be the same as those calculated for the fuel-basket drop discussed above.

6.4 Criticality

Criticality accidents which have occurred in the nuclear industry have been associated with chemical reprocessing or assemblies involving plutonium or highly enriched uranium. No criticality accidents have occurred in systems containing low-enriched uranium. We agree with the statements in the CSAR that a criticality event in a spent-fuel storage pool is highly improbable and precluded by many factors including fuel basket design, geometric restraints imposed on fuel storage, operating procedures and low fissile content of the fuel assemblies. However, as a conservative measure, the radiological consequences associated with criticality events at various fission yields were evaluated. It was assumed that all fission products, including fission gases, were contained within the UO_2 fuel matrix and not released as a result of the postulated criticalities.

Criticality accidents which have occurred in heterogeneous, water-moderated systems have resulted in total fission yields ranging from 3×10^{16} to 1.2×10^{20}. (See ref. 12.) Radiation doses at the surface of the storage pool from prompt gammas produced by nuclear excursions involving 10^{18}, 10^{19}, and 10^{20} fissions/s were calculated.[13] The nuclear excursions were assumed to occur 4.88 m (16 ft) under the surface of the pool, and calculations were made assuming (1) a point source and (2) a 30-cm (11.8-in.) diam sphere. Source energies and photons (Table 2) were taken from a distribution curve in ref. 14, and 7.5 prompt gammas per fission were assumed. Surface radiation doses from the postulated criticalities are given in Table 3. In the event of a criticality excursion, radiation levels at the surface of the pool would be low enough to allow

Table 2. Prompt gamma fission spectra[a]

Source strength (Mev)	% of total photon
.3	19.48
1.0	28.14
1.5	20.30
2.0	10.82
2.5	8.12
3.0	6.09
4.0	4.33
5.0	1.62
6.0	0.65
7.0	0.48

[a]E. K. Hyde, "Fission Phenomena," Vol. 3, *The Nuclear Properties of the Heavy Metals*, Prentice-Hall, New Jersey, 1964.

Table 3. Surface radiation doses from subsurface [4.88 m (16 ft)] nuclear excursions in spent-fuel storage pool

Fissions/s	Source geometry	Surface dose (mrem)
1×10^{18}	sphere[a]	0.10
1×10^{18}	point	1.05
1×10^{19}	sphere[a]	0.97
1×10^{19}	point	10.5
1×10^{20}	sphere[a]	9.66
1×10^{20}	point	10.5

[a]30-cm (11.8-in.) diam.

for prompt remedial action. The effects of a criticality event in a spent-fuel storage pool would be similar to those resulting from the short-term operation of a low-power, swimming-pool-type reactor commonly used in research.

7. CONDUCT OF OPERATIONS

7.1 Organization and Staff Qualifications

General Electric Company, the sole owner and operator of the Morris Operation, has demonstrated competence in the nuclear industry. Personnel in key positions at the Morris Facility have obtained experience at handling radioactive materials while employed by General Electric or other companies at other nuclear facilities. General Electric has established minimum qualifications for management, supervisory, and technical positions necessary for the safe and efficient operation of the Morris Operation satisfying the requirements of 10 CFR Part 72.17, "Contents of Application: Technical Qualifications." A plant safety committee composed of the managers of the various plant organizations has jurisdiction over matters having radiological or nuclear safety implications. The Licensing and Radiological Safety Senior Engineer, the secretary of this committee, reports directly to the manager of Morris Operation and is responsible for coordinating site regulatory matters. Besides participating in general orientation and safety courses, operator technicians are required to participate in onsite training programs prepared by management and engineering personnel qualified in the assigned topical or functional area. Records of activities relating to plant safety are accumulated to assist in the application of safety principles and objectives to plant operation. Periodic internal audits are conducted by Morris Operation management in safeguards, criticality and radiation safety. These audits are subject to review by teams from the corporate organization external to the Morris Operation.

7.2 Emergency Plans

The applicant has described in the CSAR plans for coping with the following classes of emergencies: (1) criticality incidents, (2) contamination accidents, (3) fire, (4) major equipment failures or operational accidents, and (5) other conditions such as effects from natural phenomena. These plans have been compared with and fulfill the requirements of Appendix E to 10 CFR Part 50, "Emergency Plans for Production and Utilization Facilities." Details of emergency agreements and assistance arrangements with law enforcement, medical, and other local agencies, and services are given in a supporting document, Appendix 1, NEDO-21894, "Radiological Emergency Plan for Morris Operation," which was not available for review by ORNL.

7.3 Decommissioning

The decommissioning plan for the Morris Operation is contained in Appendix A.7 of the CSAR. We have compared the plan with the technical requirements of 10 CFR Part 72.18, "Decommissioning Plan, Including Its Financing," and find it in compliance, but the financial arrangements for the decommissioning of the Morris Operation are not included in Appendix A.7.

8. REFERENCES

1. General Electric Company, *Consolidated Safety Analysis Report for Morris Operation, Morris, Illinois,* NEDO-21326C (January 1979).

2. *Safety Evaluation Report by the Division of Fuel Cycle and Material Related to License Amendment for Material License No. SNM-1265 Morris Operation Facility, Grundy County, Illinois for General Electric Company Docket No. 70-1308,* U.S. Nuclear Regulatory Commission, Washington, D.C. (Dec. 3, 1975).

3. *Code of Federal Regulations,* Title 10, Part 72, "Licensing Requirements for the Storage of Spent Fuel in an Independent Spent Fuel Storage Installation," Office of Standards Development, U.S. Nuclear Regulatory Commission, Washington, D.C. (October 1980).

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APPENDIX:

STRUCTURAL ANALYSIS OF MORRIS OPERATION CRANES

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UCC-ND Engineering

March 20, 1980

A.1 Introduction

This study is an independent evaluation of the effects of a safe shutdown earthquake (SSE) on the cask handling crane, its support structure, and the storage basin crane at the General Electric Morris Operation.[1] The analysis was performed at the request of the U.S. Nuclear Regulatory Commission.

A finite element model was developed for the cask handling crane and its support structure. The dynamic stresses in the support structure in a safe shutdown earthquake were calculated by the response spectrum method. The [illegible] of the seismically induced lateral forces acting on the crane rail and its accessories were also evaluated.

The lateral forces acting on the storage basin crane rail were treated as statically applied loads with magnitudes directly proportional to the seismic ground accelerations. The effects of these forces on the crane rail and tie-down clips were evaluated.

The results show the following:

1. The maximum combined stress that would occur in any component of the support structure of the cask handling crane during a SSE is 14,000 psi. The combined stress is the sum of the stresses due to the weight of the crane and structure and those induced by the seismic motions. The maximum combined stress found by the analysis is less than the minimum yield strength of structural steel and would not be expected to cause a failure in the support structure.

2. The maximum seismically induced lateral force acting on the cask handling crane from a SSE would be 14.2 std tons (28,400 lb). When this lateral force is transferred to one tie-down clip, it is likely that the clip would yield and become a plastic hinge. However, the adjacent clips would absorb the lateral loadings on the crane rail.

3. The loading on the tie-down clips of the storage basin crane from a SSE would be minimal. It is unlikely that these tie-down clips would fail as the result of a SSE.

A.2 Method of Analysis

The cask handling crane (CHC) is located on an elevated crane runway in the cask receiving area. The elevation is about 10.1 m (33 ft) above the floor of the building. Loadings imposed on the support structure during a safe shutdown earthquake were determined using the finite element method. The results were used to evaluate the stresses in the support structure and the interacting forces between the crane wheels and the supporting rail.

The storage basin crane (SBC) is at ground level. The dynamic amplification effects on the crane during a SSE were judged to be minimal. Forces acting on the basic crane rail during a SSE were evaluated by a static analysis.

A.2.1 Safe shutdown earthquake

The safe shutdown earthquake used in the studies is the maximum earthquake in the design criteria of the Morris Operation facility (ref. 1, p. 4-20). The maximum horizontal ground acceleration is 0.2 x g, with the vertical ground acceleration two-thirds that of the horizontal. The horizontal and vertical design response spectra used in the studies are from the NRC Regulatory Guide on the design response spectra for seismic design of nuclear power plants.[2] A damping value of 5% of critical damping was used.

A.2.2 SAP computing code

The SAP computing code is a general purpose finite element program for structural analysis.[3] The program was originally developed at the University of California, Berkeley. Currently, it is maintained by the Civil Engineering Department at the University of Southern California. The program is available at the ORNL Computing Center (SAP V - USC Version Two) and was used in this study.

A.2.3 Cask handling crane

The cask receiving area enclosure is a steel frame building attached to a concrete foundation. The crane rail is elevated and attached to girders supported by inside columns. The "A" frame supports are provided to one side of the crane runway. Purlins are attached to the outside columns and connected by sag rods. The building is covered with industrial insulation panels. Building dimensions and identities of structural members were obtained from General Electric's drawings numbered MFR-E-2228A,[4] MFR-E-2228B,[5] MFR-E-2228D,[6] MFR-E-2228E,[7] and Fluor Corporation's drawings numbered 5-2104A[6] and 5-2103A.[9]

Information on the crane rail and accessories were taken from Fluor Corporation's drawing No. 5-2105A[10] and Bethlehem Steel's Booklet 3351.[11] Cross-sectional properties of structural members were evaluated in accordance with AISC *Manual of Steel Construction*.[12] The cask handling crane is supplied by the Whiting Corporation. It has a rated capacity of 125 std tons (250,000 lb). Information on the CHC was provided by Whiting's drawing No. U-59991.[13]

A finite element model was developed for a middle section of the cask receiving area enclosure. The CHC was assumed to be located in the center of the section. A sketch of the model is shown in Fig. A.1. The crane rail is parallel to the Y axis. The Z axis is vertical, and the direction transverse to the crane runway is the X axis. The two "A" frames defined the section modeled.

All structural members, with the exception of the crane, were modeled by beam elements. The crane was treated as a rigid body. The crane wheels were simulated by short beams. The load (shipping cask) carried by the crane was modeled by a concentrated mass attached to a soft beam element suspended from the center of the crane. The soft beam element has the same stiffness as a suspended pendulum of the same length. The soft beam element is 109 cm (43 in.) long, and it is also the minimum length of the load carrying crane cable. The stiffness of the purlins, sag rods, and insulation panels was not included in the model. However, their masses were distributed to the appropriate nodes as concentrated masses. The finite element model is one section of the cask receiving area enclosure and the effects of the remaining parts of the building were accounted for by imposing appropriate constraints on the model.

The horizontal and vertical ground accelerations were input along the X and Z axes, respectively. Motion in the Y direction (along the crane runway) would be small, and, consequently, the translation along the Y axis was eliminated at all nodes. Fixed-end supports were assumed at the ground level. In order to simulate the support provided by the rest of the enclosure, the rotational degrees of freedom about the X and Z axes were also eliminated at nodes on the sections defined by the two "A" frames.

A.2.4 Storage basin crane

The storage basin crane is located at ground level. It has a rated capacity of 7.5 std tons (15,000 lb). Because of its low elevation, the dynamic amplification effect was assumed to be negligible and the total transverse forces acting on the crane rail were taken to be 20% of the total weight. These transverse forces were divided equally among the four wheels. The effects of the SSE on the crane rail and its accessories were evaluated by a static analysis.

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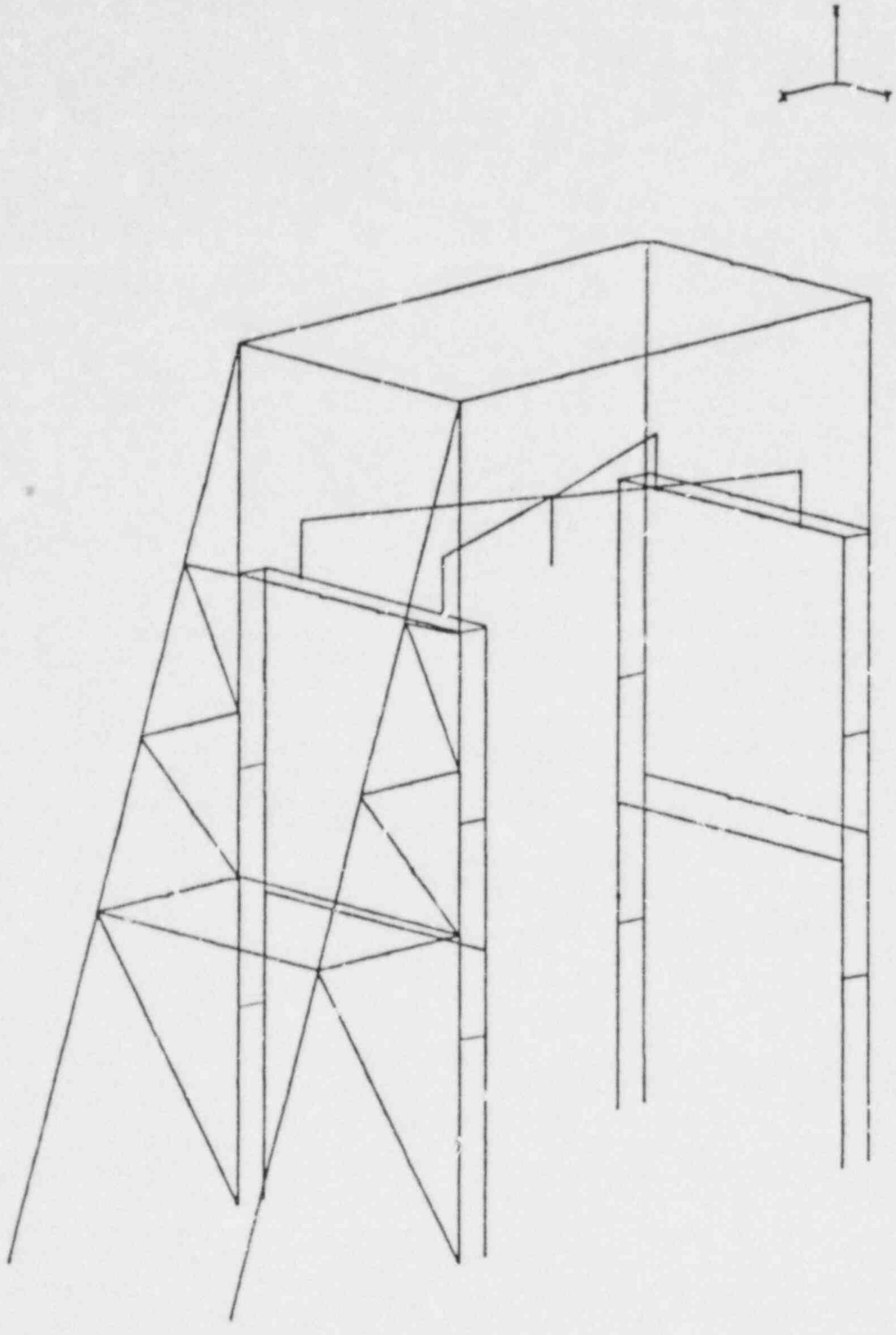

Fig. A.1. Finite element model - cask handling crane and support structure.

A.3 Analytical Results

A.3.1 Cask handling crane

The dynamic loadings on the support structure of the CHC were evaluated using the response spectrum analysis option in SAP with the NRC design response spectra.[2] The lowest 12 natural frequencies were used in the calculations and are shown in Table A.1.

Support structure. Important structural members of the CHC support structure are identified in Fig. A.2. The structural members and the stress distribution are symmetrical to a midplane between the two "A" frame sections. The total stress in each structural member consists of two components; a static and a dynamic stress. Each component was computed separately. The static stresses in the support structure were calculated using the static analysis option in SAP. The dynamic stresses were evaluated using the response spectrum analysis option in SAP and in accordance to the NRC response combination method.[14]

The total stresses in the important structural members are summarized in Table A.2. All the calculated stresses are less than the nominal yield stress for structural steel, thus the safe shutdown earthquake is not expected to produce permanent deformation or collapse of the CHC support structure.

Crane rail and accessories. The crane rail is held in place by double clips and reversible fillers spaced at a 0.61-m (2-ft) interval. A section of the crane rail, its accessories, and key dimensions are shown in Fig. A.3. The section is that of a standard 135-lb rail.[12]

The transverse forces acting on the crane rail are the computed lateral inertia (shear) forces acting on the runway. The calculated maximum shear was 14.2 std tons (28,400 lb). It was assumed that this transverse shear acted on the top of the rail. Its effects on the rail accessories are discussed below.

- Sliding of Rail Section. The free-body diagram for the forces under consideration is shown in Fig. A.4. Under the action of the transverse shear V, the rail could slide and impact in the filler plate if the friction force F is not sufficient to overcome the transverse shear V. The friction force F is proportional to the net vertical load P. The net vertical load P is the static wheel loading (one-fourth of the total weight) less the maximum vertical shear produced by the seismic motion. For the CHC, the static wheel loading is 29.6 std tons (59,200 lb). The calculated maximum shear is 8.2 std tons (16,400 lb) and the corresponding net vertical load P is 21.4 std tons (42,800 lb). The static friction coefficient between dry steel surfaces is approximately 0.7 (ref. 15) and the maximum friction force that can be developed between the rail and the

Table A.1. Crane support structure natural frequencies[a]

Mode No.	Frequency (Hz)
1	0.62
2	3.61
3	8.92
4	11.07
5	11.50
6	11.84
7	12.42
8	12.42
9	13.32
10	15.31
11	15.32
12	17.44

[a]NRC Regulatory Guide 1.60, *The Design Response Spectra for Seismic Design of Nuclear Power Plants* (December 1973).

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Detail of Front "A" Frame Section not shown.

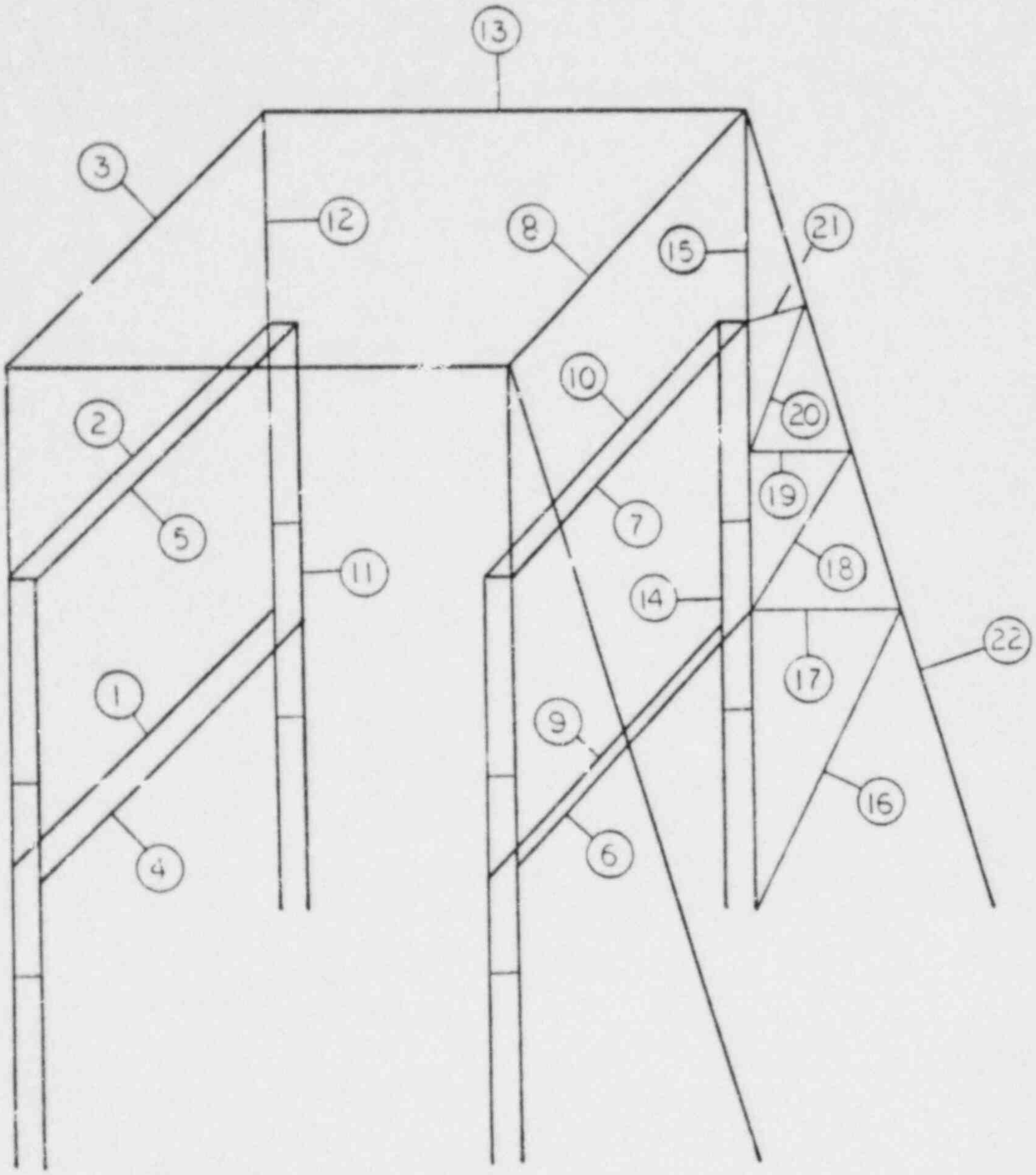

Fig. A.2. Numbering system for important structural members - crane support structure.

Table A.2. Stress distribution of crane support structure

Component[a] No.	Designation	Reference drawing No.	$(s_c)_1$[b]	$(s_c)_2$[c]
1	6W15.5	GE MFR-E-2228D (Rev. 1)	603	-603
2	6W15.5	GE MFR-E-2228D (Rev. 1)	882	-882
3	10W33	GE MFR-E-2228A (Rev. 2)	563	-563
4	8W24	Fluor 5-2104A (Rev. 4)	446	-446
5	rail and girder	Fluor 5-2105A (Rev. 2)	9,775	-9,774
6	6W15.5	Fluor 5-2104A (Rev. 4)	447	-447
7	6W15.5	GE MFR-E-2228B (Rev. 2)	664	-664
8	6W15.5	GE MFR-E-2228A (Rev. 2)	823	-823
9	8W24	Fluor 5-2104A (Rev. 4)	342	-342
10	rail and girder	Fluor 5-2105A (Rev. 2)	14,139	-14,139
11	14W61	Fluor 5-2103A (Rev. 1)	7,960	-7,799
12	21W62	GE MFR-E-2228B (Rev. 2)	12,827	-12,500
13	10W33	GE MFR-E-2228A (Rev. 2)	5,461	-5,008
14	14W61	Fluor 5-2103A (Rev. 1)	9,000	-9,919
15	10W49	GE MFR-E-2228E (Rev. 1)	9,587	-9,040
16	2L 4 x 3 x 5/16	GE MFR-E-2228E (Rev. 1)	6,059	-5,931
17	2L 3 x 2 x 1/2	GE MFR-E-2228E (Rev. 1)	7,745	-7,833
18	2L 3 x 2 1/2 x 5/16	GE MFR-E-2228E (Rev. 1)	7,301	-7,365
19	2L 3 x 2 x 1/2	GE MFR-E-2228E (Rev. 1)	10,026	-9,880
20	2L 5 x 3 x 7/16	GE MFR-E-2228E (Rev. 1)	8,360	-8,231
21	2L 5 x 3 x 7/16	GE MFR-E-2228E (Rev. 1)	8,559	-9,072
22	10W49	GE MFR-E-2228E (Rev. 1)	7,415	-7,155

[a]See Fig. A.2.

[b]$(s_c)_1$ - maximum total tensile stress in psi.

[c]$(s_c)_2$ - maximum total compressive stress in psi.

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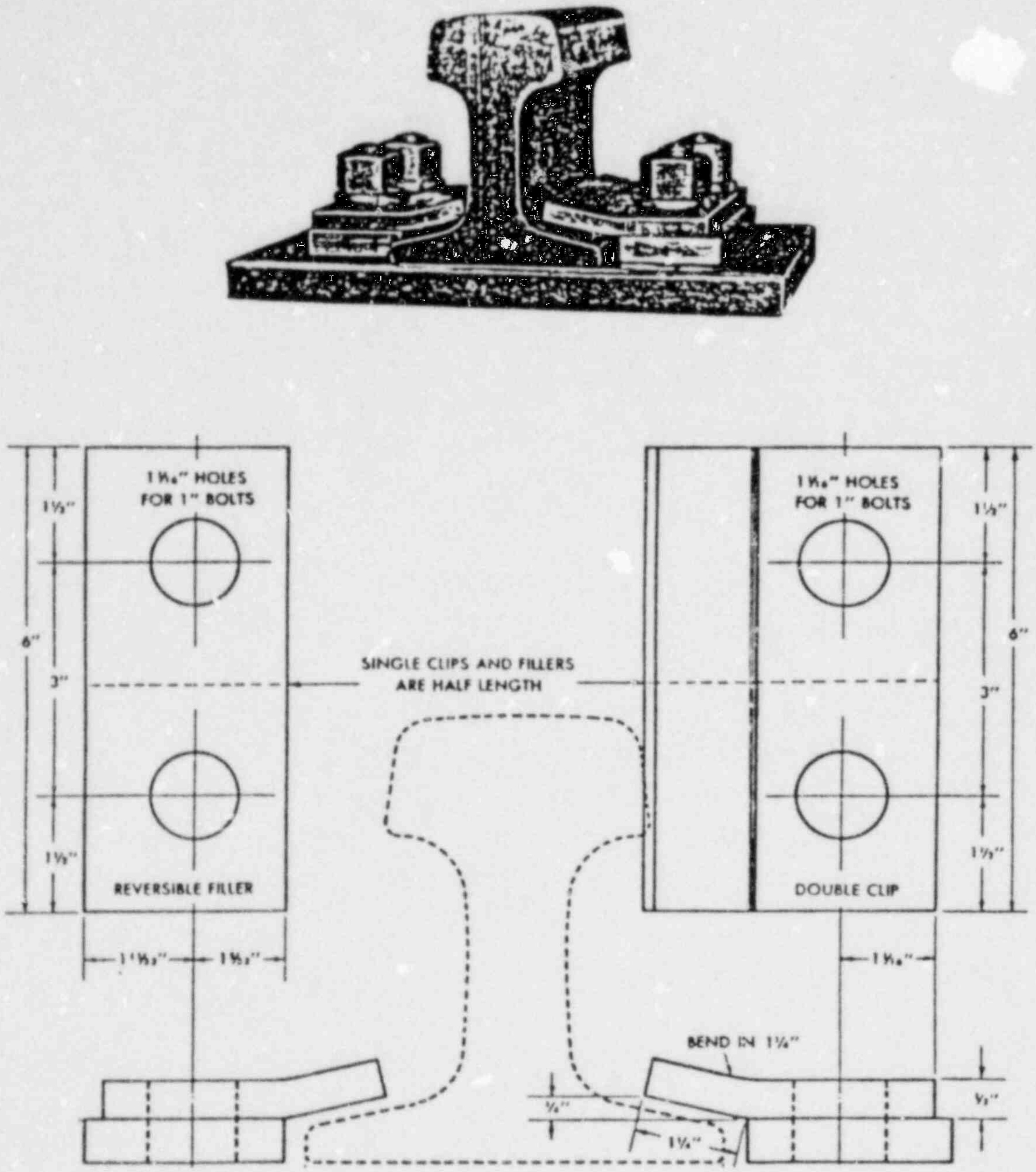

Fig. A.3. Crane rail and accessories.

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135-lb. Rail Section:
H = 5.75 (in.)
L = 5.1875 (in.)

ASCE 40-lb. Rail Section
H = 3.5 (in.)
L = 3.5 (in.)

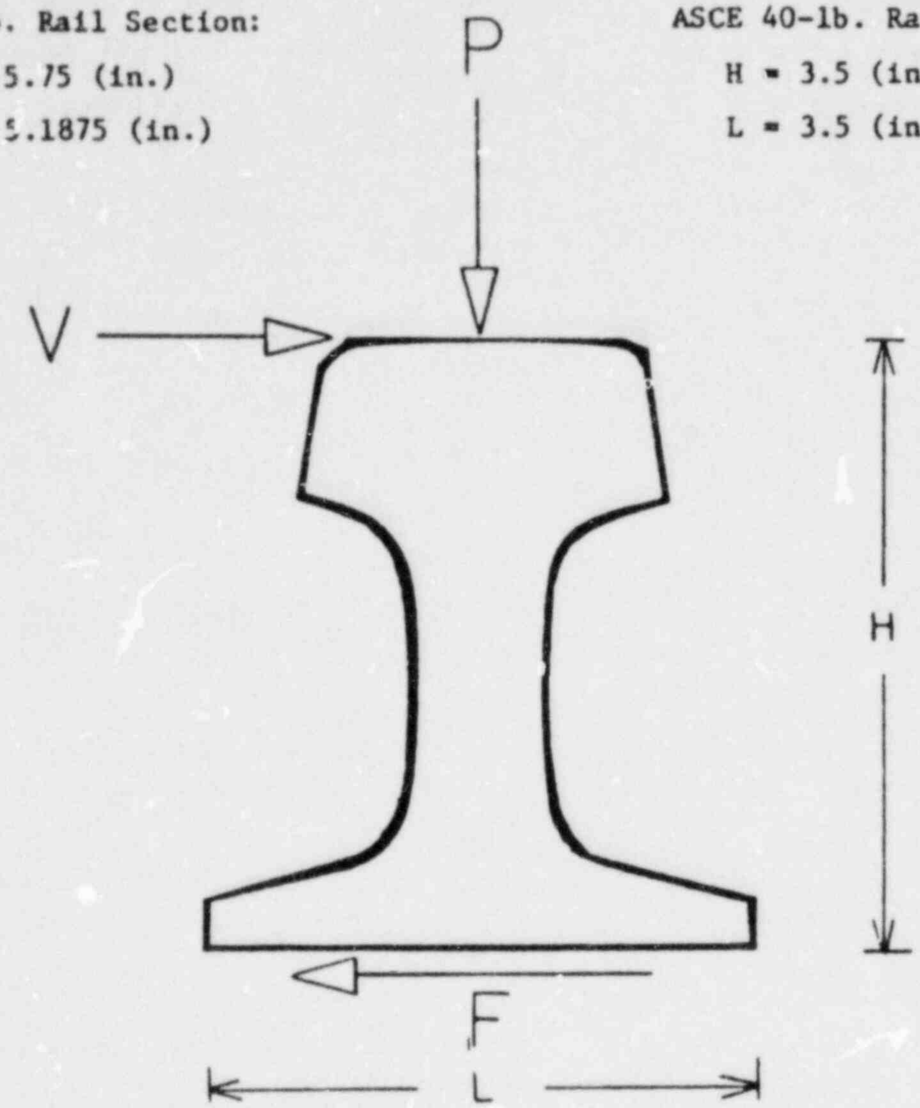

Fig. A.4. Free-body diagram for rail sliding case.

girder is 15.0 std tons (30,000 lb). Therefore, the friction force can balance the maximum transverse shear of 14.2 std tons (28,400 lb) and slipping of the rail is not anticipated.

• Forces on the Clip. The transverse force V might cause an impending rotation about one edge of the rail section. The section is assumed to pivot about the Point A, and the free-body diagram of the rail section under loading is shown in Fig. A.5. The reaction force of the slip is represented by the downward force C. The value of R is the ground reaction force point A. The net vertical load, determined in the previous subsection is 21.5 std tons (42,900 lb). Using the dimensions of a standard rail[12] and summing moments about the point A, the clip reaction force was found to be .49 std tons (9870 lb).

To evaluate the loading on the bolts, it was conservatively assumed that the force C acted on one double clip (Fig. A.3). The axial stress produced in the two 1-in. bolts is approximately 6300 psi. This estimated bolt tensile stress is much smaller than the normal yield stress for a 1-in. carbon steel bolt.

To evaluate the stresses in the clip caused by the reaction force C, the clip was modeled as a straight cantilever beam with an end force E. A sketch of the beam model is shown in Fig. A.6. The length of 1.3 in. is measured from the center of the bolt to the clip bend. The clip is 0.5 in. thick, has a width of 6 in. and is made of carbon steel with an ASTM designation of A663, Grade 60. The yield strength of that grade of carbon steel is 30,000 psi.

The crane has a wheel base of 2.84-m (9 ft 4 in.) and the rail clips are on 0.61-m (2-ft) intervals. On the average, the transverse shear acting on the rail would be transferred to at least two clips. In the two clip case, the end force E would be one-half of the reaction force C as calculated previously. The corresponding maximum clip bending and shear stresses were 23,000 psi and 2470 psi, respectively. These stresses are less than the nominal yield stress for the carbon steel and failure of the clip is not likely.

If a wheel is resting directly over a clip during an earthquake, most, if not all, of the wheel transverse shear would be transferred to one clip and the end force E would equal the clip reaction force C. For this case, the maximum clip bending stress would exceed yield and the clip could become a plastic hinge. As soon as one clip began to yield, the transverse load would be transferred to the two adjacent clips. Results from the previous discussion indicated that two clips can absorb the maximum transverse shear without yielding. Therefore, it is conceivable that the transverse shear produced by seismic motions could cause yielding in one clip but the adjacent clips would provide more than adequate compensation for such a failure.

ORNL-DWG. 81-611

135-lb. Rail Section
H = 5.75 (in.)
L = 5.1875 (in.)

ASCE 40-lb. Rail Section:
H = 3.5 (in.)
L = 3.5 (in.)

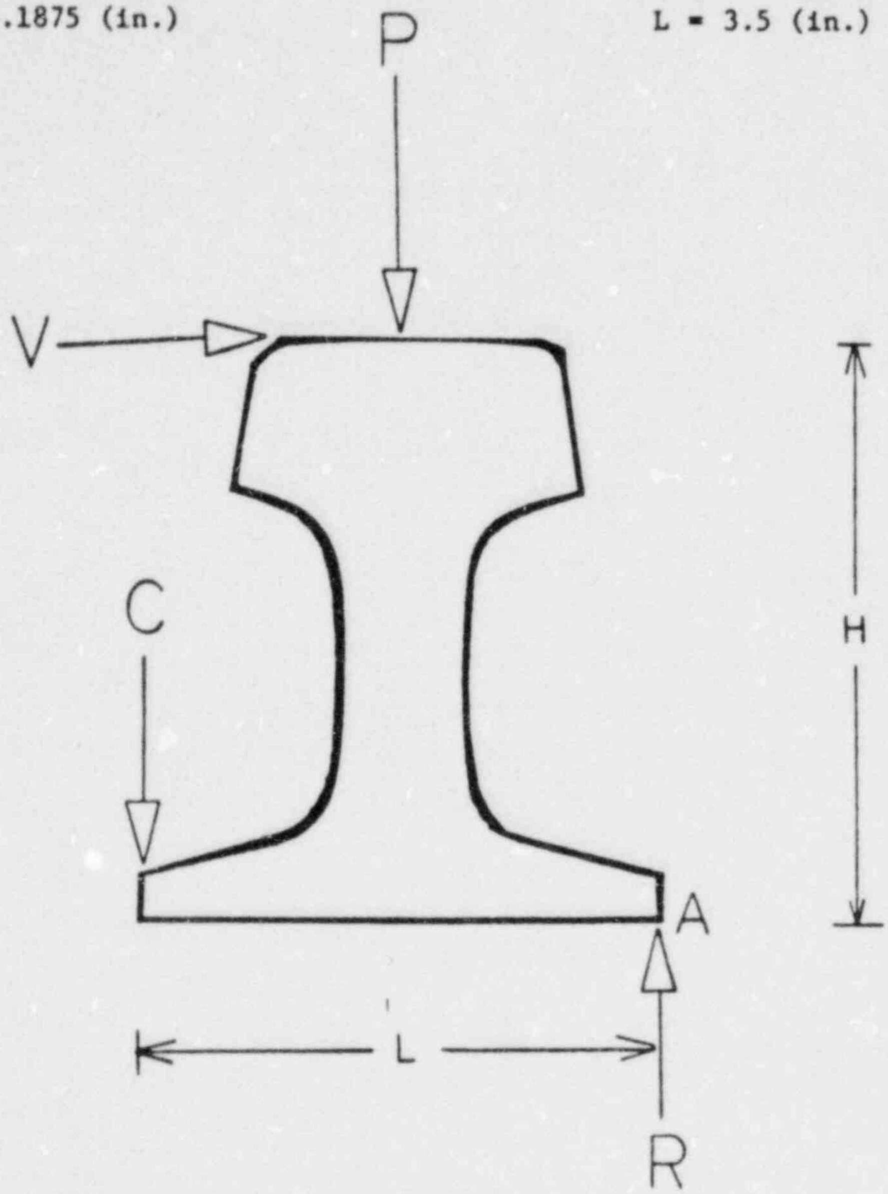

Fig. A.5. Free-body diagram for clip reaction force.

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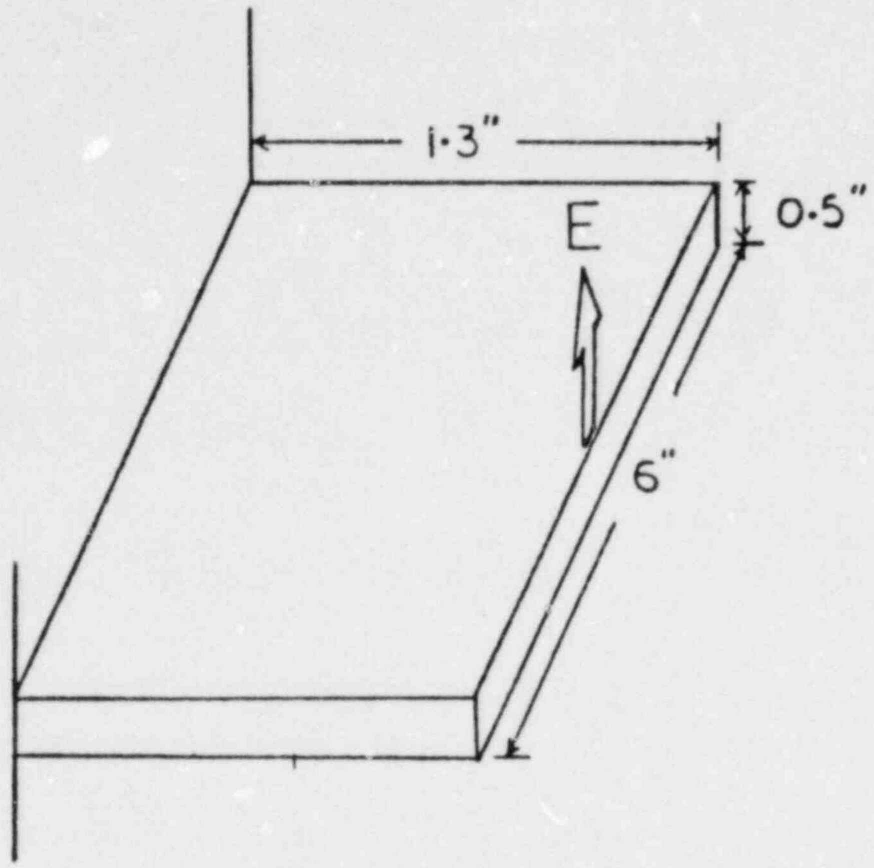

Fig. A.6. Structural model for clip bending analysis.

A.3.2 Storage basin crane

The total weight (dead weight and load) of the storage basin crane is 21.5 std tons (43,000 lb). The crane rail is an ASCE 40-lb rail section.[12] The runway is held in place by pressed, single clips and reversible filler at 0.61-m (2-ft) intervals.

Sliding of rail sections. Due to its ground level elevation, dynamic amplification was assumed to be negligible for the SBC. The free-body diagram under consideration is shown in Fig. A.4. The transverse shear V was taken to be 20% of the total weight (i.e., 0.2 x g lateral acceleration), equally divided, or 2150 lb per wheel. The corresponding vertical shear is 1075 lb. The net vertical force at each wheel, P, namely, one-fourth of total weight minus the vertical shear, is 9680 lb. Using a static coefficient of 0.7 (ref. 15), the maximum friction force that can be developed between the rail and the steel support plate is 6770 lb. The friction force F is significantly larger than the transverse force V and slipping of the crane rail on the support plate is not expected.

Forces on the clip. The free body diagram of the forces under consideration is shown in Fig. A.5. The transverse shear V is 2150 lb. The net vertical force P is 9680 lb, R is the ground reaction force at point A, and C is the clip reaction force acting on the rail. Using the dimensions of an ASCE 40-lb rail[12] and summing moments about the point A, it was found that the moment produced by the force P was more than sufficient to counter balance that produced by the trnsverse force V. The rail section cannot rotate and, consequently, there is no reaction force from the clip. The loadings on the clips would be negligible and the transverse shear is not expected to cause a failure in the clips of the SBC.

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4. TITLE AND SUBTITLE (Add Volume No., if appropriate) Safety Review of the Design, Operation, and Radiation Sections of the General Electric Morris Operation Consolidated Safety Analysis Report	2. (Leave blank)
	3. RECIPIENT'S ACCESSION NO.
7. AUTHOR(S) J.P. McBride, K.H. Luk, C.R. Hammond	5. DATE REPORT COMPLETED MONTH Janua y YEAR 1981
9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) Oak Ridge National Laboratory Oak Ridge, TN 37830	DATE REPORT ISSUED MONTH July YEAR 1981
	6. (Leave blank)
	8. (Leave blank)
12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) Office of Nuclear Material Safety and Safeguards Division of Fuel Cycle and Material Safety U.S. Nuclear Regulatory Commission Washington, DC 20555	10. PROJECT/TASK/WORK UNIT NO.
	11. CONTRACT NO. FIN B0102
13. TYPE OF REPORT	PERIOD COVERED (Inclusive dates)
15. SUPPLEMENTARY NOTES	14. (Leave blank)

16. ABSTRACT

A safety review was made of Sections 4 through 9 of the Consolidated Safety Analysis Report (CSAR) for the GE Morris Operation spent-fuel storage facility. The sections reviewed include Design Criteria and Compliance, Facility Design a d Description, Radiation Protection, Accident Analysis, and Conduct of Operations. The safety review was performed in accordance with the *Code of Federal Regulations*, Title 10, Part 72, "Licensing Requirements for the Storage of Spent Fuel in an Independent Spent Fuel Storage Installation" and contains independent estimations of source terms and dose-commitments from postulated accidents in the storage facility and a structural analysis of the Morris Operation cranes as an appendix. The review confirms that the features of the facility as described in Sections 4 through 9 of the CSAR fulfill the safety requirements of 10 CFR 72, and it is concluded that spent-fuel handling and storage at the Morris Operation do not present significant risks to public health and safety.

17. KEY WORDS AND DOCUMENT ANALYSIS	17a. DESCRIPTORS
17b. IDENTIFIERS/OPEN-ENDED TERMS	

18. AVAILABILITY STATEMENT	19. SECURITY CLASS (This report)	21. NO. OF PAGES
Unlimited	Unclassified	
	20. SECURITY CLASS (This page) Unclassified	22. PRICE S

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