NSP

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

Red Wing, Minnesota

UNITS 1 AND 2

SPENT FUEL STORAGE
LICENSE AMENDMENT REQUEST
DATED JANUARY 31, 1980

1877 041

NORTHERN STATES POWER COMPANY
MINNEAPOLIS, MINNESOTA

8002050517

License Amendment Request
dated January 31, 1980

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EXHIBIT A

Prairie Island Nuclear Generating Plant

License Amendment Request dated January 31, 1980

PROPOSED CHANGES TO THE TECHNICAL SPECIFICATIONS
APPENDIX A OF OPERATING LICENSES
DPR-42 & DPR-60

Pursuant to 10CFR50.59, the holders of Operating License DPR-42 and DPR-60 hereby propose the following changes to Appendix A, Technical Specifications:

1. Specification 3.8.B.1 Refueling and Fuel Handling

PROPOSED CHANGE

Change the footnote at the bottom of the page to read -

"For the purpose of completing the fuel storage pool modifications, the movement and placement of loads shall be in accordance with the installation procedure approved by the plant on-site review committee."

REASON FOR CHANGE

The current footnote referring to hearing testimony and licensing submittals will not be applicable to the forthcoming modification.

SAFETY EVALUATION

The plant Operations Committee (on-site review committee) is composed of senior plant management personnel. These personnel were responsible for review/approval of the procedures used for the modifications performed in 1977. Currently the Operations Committee has 8 of the 10 members originally responsible for review/approval of procedures used in the last rack modification. Thus, extensive experience and familiarity with the appropriate procedures and precautions exist to ensure no safety hazard exists for the procedures used in the modification.

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2. Section 5 Design Features

PROPOSED CHANGE

A. Specification 5.3.A.2, Reactor Core

Change to read:

2. The average enrichment of the reload core is a nominal 2.90 weight percent of U-235. The highest uranium - 235 loading is a nominal 39 grams of U-235 per axial centimeter of fuel assembly (average).

B. Specification 5.6.A Fuel Handling Criticality Consideration

In the second paragraph, change the third sentence to read:

"In addition, fuel in the storage pool shall have a U-235 loading of $\leq$ 39.0 grams of U-235 per axial centimeter of fuel assembly (average)".

Delete the following from the last sentence in Specification 5.6.A - "except for initial new fuel storage".

C. Specification 5.6.B, Fuel Handling Spent Fuel Storage

Change the first paragraph, first sentence to read:

"The spent fuel storage facility is a two compartment pool that may contain up to 1582 storage locations for spent fuel assemblies".

REASON FOR CHANGE

Specifications 5.3.A.2 and 5.6.A are related in that spent fuel criticality considerations establish the upper bound on the U-235 loading. From a conservative standpoint, the reactor core should not have a higher loading than the spent fuel pool requirement since one should always assume that full core discharge of slightly used fuel (which would be the highest enrichment) might be required.

Northern States Power Company has used or intends to use fuel produced by Westinghouse Electric Corporation and Exxon Nuclear Company. The fuel supplied by these two vendors has different clad thicknesses. In order to achieve the same optimum fuel burnup, these suppliers use different enrichments that would correspond to 39 gm U-235/axial cm fuel assembly length, the loading assumed in the safety analyses.

The reference in Specification 5.6.A to initial new fuel storage is no longer appropriate.

Specification 5.6.B must be changed since Northern States Power Company intends to increase the available spent fuel storage locations from the current 687 to 1582. This change should provide adequate storage capacity until either an AFR (away from reactor) storage facility or reprocessing facility is built that can provide a location to ship the Prairie Island spent fuel assemblies.

SAFETY EVALUATION

On April 14, 1978, the NRC Staff distributed the document "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" (Reference 1). This safety evaluation provides the information recommended by that document. Reference will be made to Exhibits C and D, Nuclear Services Corporation documents specifically prepared to address selected areas of this amendment submittal.

This safety evaluation addresses the following areas -

1. Nuclear Evaluation
2. Thermal Hydraulic Evaluation
3. Mechanical, Material, and Structural Evaluation
4. Environmental Effects Evaluation (Cost Benefit, Radiological, Accident)

1.0 INTRODUCTION

The Prairie Island Nuclear Generating Plant (PINGP) fuel storage facility was previously described in sections 9.3 and 9.5 of the Final Safety Analysis Report (FSAR). The fuel storage facility has two spent fuel storage pools as shown in Figures 1 and 9, which have a combined storage capacity for 687 spent fuel assemblies.

Since Spring 1976, forty spent fuel assemblies have been transferred annually from each reactor to the spent fuel pool for interim storage. There are currently 320 assemblies stored in Pool 2. At this rate, Pool 2 will be filled during the 1983 Unit 2 Spring refueling outage (as shown in Table 1). Thus spent fuel would have to be placed in Pool 1.

The full core-off-load capability will be lost after the 1983 Unit 1 refueling outage. The current storage capacity limit will be met after the Fall 1984 refueling outage.

An Away From Reactor storage facility or reprocessing facility is not expected to be operational by the time when the PINGP spent fuel storage pools would be full. For this reason, Northern States Power (NSP) plans to increase the storage capacity of the current fuel storage facility from 687 to 1582 storage locations.

1.1 Storage Needs/Contractural Arrangements

Northern States Power Company had a fuel reprocessing contract with Nuclear Fuel Services of West Valley, NY since December 31, 1965. This contract initially covered the Pathfinder plant, but was amended on May 7, 1970 to include the Prairie Island and Monticello plants. On September 20, 1976, NFS notified NSP that the company was withdrawing from the nuclear fuel reprocessing business. Since no other facilities were available and there was inadequate storage space, a modification was applied for on November 24, 1976 and authorized on August 16, 1977. This original modification expanded the storage capacity to the current 687 storage spaces.

The first Prairie Island reactor refueling occurred in March 1976. After the initial cycle, each Prairie Island unit has operated at very high availability, resulting in one refueling per unit per year. Thus, 80 fuel assemblies per year have been added to the spent fuel pool resulting in the current number of 320 spent fuel assemblies. In addition to spent fuel, there are 1 Rod Control Cluster Assembly (RCCA), 2 Source assemblies, 100 - Burnable Poison Rod Assemblies (BPRA's), 8 Part Length Rod Control Cluster Assemblies (PLRCCA's), and 11 Thimble Plugs stored in the storage pool. These components are inserted into spent assemblies to reduce the storage space consumption.

1.2 Storage Modification Plans

There are three 7x7, six 7x8, three 8x8, and one 3x4 storage racks presently in the pools. Northern States Power Company plans on reracking the small pool (Pool 1) in February 1981 with the nine racks (as shown in Figure 2), which would have a storage capacity for 462 assemblies.

All spent fuel assemblies (400 expected) would then be transferred to the small pool. So also would the spent assemblies from the 1981 Unit 1 refueling outage. The modification is being scheduled for early 1981 because enough time must be available to procure materials, fabricate new racks, remove the existing racks and install all of the new racks before the Unit 2 1982 refueling. With that outage, 482 spent fuel assemblies would have been discharged which is greater than the new Pool 1 capacity. After the modification, all spent fuel assemblies would be transferred to Pool 2. The four 7x7 racks on the southeast corner of Pool 1 would be removed if a cask is brought into the SFP facility for fuel shipment. This would result in a storage configuration as shown in Figure 3. These four racks would only be used if a full core off-load is required. Thus the resulting storage capacity for normal refuelings would be 1386 spaces.

1.3 Construction Costs

The estimated installed cost of the proposed modifications to the Prairie Island facility is $5,230,000 (1980 dollars). Key costs are broken down as follows:

Material	$2,930,000
Labor	830,000
Permits and Licenses	340,000
Escalation, AFC, A&G, Engineering Construction Supervision, AE	1,130,000

for both installation of the new racks and removal of the old racks. Consideration has also been given in the estimate to escalation and depreciation costs.

1.4 Alternatives to Increasing Storage Capacity

NSP has considered the alternatives -

(1) Ship fuel to a fuel reprocessing facility
(2) Ship fuel to an Away-From-Reactor (AFR) storage facility
(3) Ship fuel to another reactor site
(4) Shut down the reactors

Spent fuel is currently not being reprocessed in the United States on a commercial basis. None of the three commercial reprocessing facilities in the United States is currently operating. The General Electric Morris, Illinois aand Nuclear Fuel Service West Valley, N.Y. facilities are both in a decommissioned status. The Allied General Nuclear Service (AGNS) facility at Barnwell, South Carolina has not been licensed to operate.

The Morris facility owner is primarily using the storage space for GE-owned fuel (which had been leased to utilites) or for fuel which GE had previously contracted to reprocess. GE as a matter of policy will not store spent fuel unless they had previously committed to reprocess that fuel. There is no such commitment for Prairie Island.

The West Valley facility has capacity for about 260 MTHM, with approximately 170 MTHM presently stored in the pool. Although the storage pool at West Valley is not full, officials have indicated they are not accepting additional spent fuel for storage, even from those reactor facilities with which it had reprocessing contracts, e.g. Northern States Power Company (as discussed previously).

The Allied General Nuclear Service facility has not been licensed to operate and cannot be depended upon for receipt of spent fuel due to the indefinite deferral of licensing proceedings for the plant.

Department of Energy (DOE) plans for AFR storage have not been finalized nor has funding been received. Thus it is unlikely that DOE's 1983 target will be met.

The indefinite deferral of reprocessing has left potentially all commercial nuclear reactor operators with limited storage capacity and no place to ship spent fuel. Other utility nuclear plant operators are unwilling to risk premature shutdown of their own plant because of storage of off-site fuel generated by another facility.

Shipping the Prairie Island spent fuel for storage in the Monticello spent fuel pool has been considered. The fuel assemblies for the two plants differ significantly. Because of this, the fuel handling and storage equipment for fuel assemblies from the two plants are not compatible. Storage of Prairie Island fuel at Monticello has been rejected for the following reasons:

1. A substantial modification of the Monticello fuel handling and storage equipment would be required.

2. The storage space at Monticello is needed to ensure continued operation of the Monticello plant.

3. Storage of Prairie Island fuel at Monticello would result in increased handling and shipping of spent fuel.

Thus the alternatives that involve fuel shipment from the Prairie Island Plant are not viable at this time nor in the immediate future when the PINGP spent fuel pools would be full.

NSP does not consider shutting down PINGP a viable option for the following reasons.

Operation of the Prairie Island Plant could be in jeopardy following loss of full core reserve in 1983. Even if NSP were successful in avoiding full core discharges in 1983 and 1984, both units at Prairie Island will have to be shut down by 1985 if additional spent fuel storage is not provided. Assuming knowledge of an impending long term shutdown occurs in early 1980, a new generating facility to replace Prairie Island could not be placed in operation until 1989 or 1990.

The effects of a shutdown were examined for the year 1985. Without Prairie Island, NSP's remaining generating capacity, including oil generating facilities, would barely equal the anticipated peak demand. In addition to decreased reliability of its power supply system, NSP's electric production expenses for the twelve month period examined are projected to increase by approximately $160 million if Prairie Island is not available. Most of this expense would be caused by the increased utilization of NSP's coal-fired and oil-fired generating facilities which would have to supply much of the energy that would have been produced by Prairie Island. The projected increase in oil consumption would be approximately 40 million gallons, while coal consumption would be projected to increase by over 3 million tons in the initial twelve month period. Increased purchases from other utilities to offset the deficit caused by the shutdown of Prairie Island also contribute to the increased electric production expenses.

Beyond 1985, the electric production expenses will continue to increase due to escalation of fuel prices and other operating costs, until replacement generating capacity is in service. Expected load growth in 1986 will cause additional increase in electric production expenses since there will not be a corresponding generating capacity addition in that year. Existing generating capacity additions scheduled for 1987 and beyond will keep pace with load growth, but will not replace the Prairie Island capacity.

1.5 Material Resource Effects

For the new spent fuel storage racks the material resource commitment would be:

Material	Weight
Stainless steel	273 tons
Silicone polymer	32,300 lbs
Boron carbide	31,900 lbs

not including usage attributable to manufacturing tolerances. This usage does not constitute a significant resource commitment since the materials are not in short supply.

The silicone polymer and boron carbide are used in the form of Boraflex, previously authorized by the NRC for use at the Point Beach and Nine Mile Point facilities.

2.0 NUCLEAR SAFETY EVALUATION

Exhibit C, Section 3.3, describes the analyses conducted to verify that Keff $\leq$ 0.95 under a variety of conditions. Included are (1) normal storage and (2) postulated accident conditions. Sections and tables described below refer to Exhibit C, unless otherwise noted.

2.1 Design Criteria

Section 3.3.1 describes the design criteria (fuel design, Keff limit, standard review plan compliance, and temperature and fuel position considerations).

2.2 Analytical Methods

Section 3.3.2 describes the analytical method used to evaluate uncertainty considerations including: (1) transport correction, (2) method bias, (3) fuel location effects, (4) storage tube pitch tolerance effect, (5) methods bias uncertainty, (6) B^{10} tolerance effect, (7) storage tube dimensional effect. This section discusses the calculational model and computer code used to determine Keff, as well as the comparison of code calculations with actual experimental data.

2.3 Normal Storage Case Results

Section 3.3.3 contains the analytical results for the normal storage case (including effects of uncertainties). Tables 3.3-4 and 3.3-5 show the effects of storage tube pitch and pool water temperature on Keff, respectively.

2.4 Postulated Accidents

Section 3.3.4 describes the evaluation of the postulated accidents specified by Reference 1 Section III 1.2. As noted in section 9 of the FSAR, the spent fuel pool structure is designed as a Class I structure that meets seismic and tornado criteria in Appendix B of the FSAR.

2.5 Conclusions

Section 3.3.5 provides the conclusions that proposed design meets the specified criteria of Keff $\leq$ 0.95 for the postulated conditions.

Thus the Section 3.3 description addresses the section III 1.1-1.4 requirements of Reference 1.

2.6 Acceptance Criteria for Criticality

Since the new spent fuel racks will contain boron as the neutron absorbing medium, on-site verification tests will be conducted to ensure within 95% confidence limits that the neutron absorber is present in the racks.

A separate check of each tube will be conducted. Recalibration will be performed on a periodic basis to assure adequate equipment calibration. A device having a neutron source and thermal neutron detectors would be lowered into the storage tubes. This device would monitor the effectiveness of the borated sheets at absorbing the neutrons.

Sample coupons will be removed every five years and tested to confirm that the neutron absorber condition has not changed.

3.0 THERMAL-HYDRAULIC EVALUATION

3.1 Decay Heat Calculations

The calculations for determining the amount of thermal energy that would be removed by the spent fuel pool cooling system were conducted in accordance with Branch Technical Position APCSB9-2 entitled "Residual Decay Energy for Light Water Reactors for Long Term Cooling". These calculations are summarized in Section 3.5.2 of the Exhibit C report.

3.2 Analytical Results

Section 3.5 of the Exhibit C report addresses the following - (1) Fuel Assembly Cooling Analysis (2) Pool Cooling Analyses. These analyses assume that no fuel assemblies are removed from the core until 100 hours have elapse after reactor shutdown (in accordance with Technical Specification 3.8.A.7). For both the normal refueling and full core discharge cases, the complete transfer of fuel assemblies is assumed to be completed by 150 hours after reactor shutdown. This assumption is conservative. Based on the eight refuelings conducted at Prairie Island to date, at least 150 hours after shutdown is required for normal refuelings.

3.3 System Description

The spent fuel cooling system is described in Section 9.3 of the FSAR. Figure 4 shows a sketch of the existing system. The capacity of the SFP cooling pumps has been increased to 2200 gpm.

Heat is transferred from the spent fuel cooling system fluid to the component cooling system in the SFP heat exchangers. Either unit's component cooling system may be used to remove the heat from the spent fuel cooling system. If an event occurs which requires safety injection, component cooling will continue to be supplied to the SFP heat exchangers. The operator would transfer the spent fuel cooling load to the alternate unit's component cooling system in the event of a LOCA to reduce the heat loads on the component cooling system of the affected unit.

This modification does not affect the largest heat load (due to full core discharge) which is nominally 58% of the total spent fuel cooling heat load at design conditions. The increased heat load, due to those spent fuel assemblies stored for longer periods of time, can easily be accommodated by the component cooling system design. Compared to the maximum heat rejection rate from the plant to the cooling tower water, 8.37×10^9 BTU/hr, (Reference 4), the increased heat rejected would be insignificant.

3.4 System Indications

The water in the spent fuel pools serves a two-fold function - (1) provides adequate radiation shielding and (2) acts as a cooling medium to remove decay heat from the spent fuel assemblies. Three systems have been provided to alert the operator to the existence of unusual conditions (in addition to those described in Section 5). These are the SFP level and temperature detection systems and the SFP leak detection system as described below.

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Spent Fuel Pool Level

Level detectors are provided for both pools. These sensors input to provide high and low level alarms for each pool on each unit's control board alarm panel (8 total). Since the high level alarm is set at 753' 8" (nominal), and the low level alarm is set at 751' 6" (nominal), the operators will be alerted promptly so that corrective action can be taken to eliminate the cause of the alarm.

Spent Fuel Pool Temperature

Temperature detectors are provided for both spent fuel pools. A high temperature alarm, nominally set at 130F, is provided for each pool on each unit's control room alarm panel (4 total).

Spent Fuel Pool Leak Detection

The spaces between the spent fuel pools' stainless steel liners and the concrete support structures are divided into twelve (12) sections. Any leakage into these areas is routed via a separate pipe to a common open sight drain trough and then to the Waste Disposal systems.

3.5 Operational Monitoring

The auxiliary building operator monitors SFP level, radiation, temperature, and leak detection systems periodically as a routine shift responsibility. Anomalous conditions are reported to the control room operators and shift supervisors. In addition, the control room operator reviews SFP radiation monitor levels as a normal shift responsibility. Unusual level and temperature are alarmed on both units' control board panels, as described in section 3.4.

3.6 Conclusions

The analytical results demonstrate that the existing systems will satisfactorily handle the additional heat load. The FSAR design basis (normal refueling conditions) SFP temperature was 120F. The Exhibit C Section 3.5 analysis demonstrates that with the larger SFP heat exchanger and one SFP cooling pump in service that the maximum temperature expected would be 124F. With two pumps and heat exchangers in service this temperature would be lower. The rate of evaporation and thus the need for makeup water is not expected to be changed by the proposed modifications. (Reference 3)

4.0 MECHANICAL, STRUCTURAL, AND MATERIAL EVALUATION

4.1 Description of the Spent Fuel Pool and Racks

The SFP storage facility and systems were described in the FSAR Sections 9.3 and 9.5. Figures 5-9 of this report show the location of the spent fuel storage facility in relation to other plant structures. No changes in the plant structures are planned. Exhibit C Figure 3.1-1 shows the typical 7x8 rack structure. Exhibit C figure 3.3-1 shows the normal unit cell configuration. The neutron absorber shown will be in the form of Boraflex, previously authorized by the NRC for use at the Point Beach facility. (Reference 2)

4.2 Structural and Mechanical Analyses

Exhibit C, Section 3.4 describes the mechanical and structural analysis for the new spent fuel racks.

Exhibit D contains the structural evaluation for the spent fuel storage pool facility. That report addresses the following areas:

1. Loads, load combinations, and acceptance criteria
2. Method of analysis and computation of design loads
3. Evaluation results and conclusions.

4.3 Small Pool Protective Cover

A report describing the structural evaluation for the small pool protective cover was provided by letter, L O Mayer (NSP) to Don K Davis, Acting Chief ORB#2 (NRC) dated April 14, 1977. This report provided a system description and discussion of loadings, allowable stresses and design procedure, and design results. This report is included as Exhibit E.

4.4 Material Evaluation

The stainless steel sheath provides support for and protection of the Boraflex. Gases that might be generated by irradiation of the Boraflex are vented to prevent bulging and swelling of the stainless steel shrouds.

No evidence has been determined that indicates any significant deterioration of Boraflex through a cumulative irradiation in an excess of $1x10^{11}$ rads gamma occurs to effect the suitability of Boraflex as a neutron shielding material

Testing to date of Boraflex in a neutron and gamma field indicates that Boraflex is suitable for the proposed use.

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All permanent structural material exposed to the spent fuel pool environment that is used in the fabrication of the spent fuel storage racks is 304 stainless steel with the exception of the leg adjusting bolts which are 17-4Ph stainless steel. These materials were chosen for compatibility with the spent fuel pool water.

At the design operating temperature of $120^{o}F$, there is no deterioration or corrosion of stainless steel in this environment. There is also no corrosion problem at temperatures up to and including pool boiling. All other structural components in the spent fuel pool system, such as the pool liner, cooling system pipe connections, etc., are made of stainless steel.

The high density spent fuel storage racks to be used at PINGP will utilize Boraflex sheets within an inner and outer stainless steel clad. The cells will be vented to prevent bulging and swelling of the structural steel. The stainless steel clad is Type 304 meeting the requirements of ASTM A 240. The Boraflex sheets will be demonstrated, at a 95% confidence level, to have a minimum B^{10} content of 0.04 gm/cm^2 of sheet surface area. The Boraflex is designed to operate in a 2100 ppm nominal boric acid concentration, normal pool temperature 80F-140F and total radiation exposure of 10^{11} Rads (gamma).

In summary, the pool liner, rack lattice structure, and cell exteriors are all stainless steel, which has demonstrated good corrosion resistance in PWR spent fuel pool environments. The design, material selection, and the NDE program provide a high degree of assurance that the integrity of the fixed absorber material will be maintained. The material used in the new spent fuel storage racks is similar to present components and does not affect or alter previous evaluations.

4.5 Neutron Absorber Verification Programs

Close control and verification of the material properties utilized in the manufacture of the Boraflex is assured through the manufacturer's Quality Assurance Program and is documented on appropriate material certification reports. Prior to inserting the Boraflex sheets into the finished cell configuration, each sheet is identified in order to allow traceability to the end product. Records are generated for each cell identifying the plates installed in that cell by serial number, thereby providing positive assurance that the required plates are in place.

During rack fabrication, additional care is exercised to prevent damage to the stainless steel cladding of the fuel storage cells. Packaging and shipping will be done in accordance with approved procedure to minimize the possibility of degrading the quality of the racks during transit. A thorough receipt inspection at PINGP is performed to assure no damage has occurred.

Documentation is maintained on all testing and surveillance performed on the fuel storage cells as well as material certification reports on all materials used in the construction of the cell.

4.6 In-Pool Surveillance Program

Surveillance specimens are provided to allow for surveillance over the lifetime of the fuel storage racks. The purpose of these specimens is to provide assurance that no unexpected corrosion is occurring which could compromise the integrity of the Boraflex. The surveillance specimens are in the form of removable stainless steel clad Boraflex sheets, which are typical of the fuel storage cells. These specimens can be removed and examined.

4.7 Conclusions

The fuel pool structures are structurally adequate to withstand additional loads that would result from the proposed modification in which the present racks will be replaced by high density racks.

The Boraflex material will provide the required neutron absorption and based on test data to date will perform satisfactorily in the intended environment. The stainless steel used in the modification will provide the required structural integrity.

5.0 RADIOLOGICAL EVALUATION

We have evaluated the radiological impact of this modification. This section addresses considerations of

1. Radioactive Waste
2. Radiation Exposure

and describes the systems used to reduce the likelihood of spent fuel radionuclides reaching the environment and the systems used to monitor radionuclides and radiation levels associated with the spent fuel storage.

5.1 Impact on Wastes

5.1.1 Solid Waste

The thirteen (13) existing spent fuel racks will be removed from the spent fuel pool and disposed as low level radwaste at licensed disposal sites. Based on prior experience, the crevices in the racks do not make it practical to decontaminate the structures to levels low enough to recycle the stainless steel. The total solid radwaste, generated by this modification, is expected to be 15050 ft^3, most of which is the old spent fuel racks. The radioactivity content has been conservatively estimated to be less than 270 Ci. The total estimated weight of the racks and associated structural waste is 310,000 lbs.

With an estimated radioactivity of less than 40 Ci, approximately 132 ft^3 of solid radioactive waste are produced by operation of the spent fuel facility annually. There is not expected to be a significant increase due to this modification. The solid waste generated by the spent fuel pool cleanup system represents approximately 2% of the total volume shipped from the plant. In 1978, approximately 6870 ft^3 of solid radioactive waste was shipped from all sources at the facility.

Typically the frequencies for replacing cleanup system filtration components are -

SFP Filter	1/yr
SFP Skimmer Filter	5/yr
SFP Demineralizer	2/yr

5.1.2 Liquid Waste

The only radioactive liquids resulting from the construction activity are the evaporator distillates from processing the mop water and SFP resin sluicing. This is estimated to be less than 3000 gallons, which after further processing through a demineralizer, will have a radioactivity content of less than 12 uCi and less than 1.25 Ci tritium.

There should not be an increase in the normal liquid wastes due to this modification since the spent fuel cooling system operates as a closed system with removal of soluble ionic and insoluble particulates by the cleanup system. Water originating from cleanup of SFP floors and sluicing of SFP resins are estimated to be less than 1000 gallons per year. The associated radioactivity content is less than 4 uCi and less than 0.4 Ci tritium.

Existing liquid effluent technical specifications provide assurance that liquid releases are enveloped by the radioactive liquid waste design basis.

5.1.3 Gaseous Wastes

No radioactive gases are expected to be generated while performing this modification that are over and above those associated with normal use of the SFP.

Gaseous waste from the SFP during operation is primarily tritium evaporated from the pool surface and released through the normal SFP ventilation system. In 1978, the total plant gaseous tritium release was 143.4 Ci, most of which can be assumed to come from the SFP.

The only significant noble-gas isotope remaining in the spent fuel is Kr-85 (Reference 7). This modification is not expected to have any significant on-site or off-site consequences due to Kr-85 (references 2, 3). There have not been detectable releases of Kr^{85} from the SFP to date.

On occasion, a small quantity of I-131 has been introduced into the SFP water from makeup water and subsequently released to the atmosphere. In the first six months of 1979, this amounted to 1.143 mCi, approximately one percent of the plant design objective for I-131 releases.

5.2 Radiation Exposure

5.2.1 Modification

In the 1977 rerack, the man-rem exposure associated with poolside work, removal and disposal of the old racks, and installation of the new racks was 4.54 man-rem. For this rerack, the overall radiation exposure associated with removal of the old racks, movement of spent fuel currently in the SFP, installation of the new racks, and preparation of the old racks for shipment is conservatively estimated to be less than 40 man-rem. Based on experience with these work tasks, it

is anticipated that the actual accumulated dose will be significantly less than the estimate. This occupational exposure will not be incurred on a continuing basis. The exposure is small compared with the total plant occupational exposure over the useful life of the modification (approximately 200 man-rem annually).

5.2.3 Annual Exposure

Based on experience to date, the expected annual man-rem exposure due to all operations associated with spent fuel pool area related activities, is as follows:

Operation	Man-hours	Total Exposure (Man-Rem)
Fuel Handling	600	4.20
Equipment Checkout & Maintenance	140	0.54
System Maintenance	60	1.20
Radwaste handling	16	0.80
Cleanup	100	0.20
Total		6.94

5.2.3 Internal Exposure

Operations in the spent fuel pool area have not contributed any measurable internal doses to plant personnel and are not expected to be a significant source of internal doses in the future.

Periodically the health physics staff conducts whole body counting of personnel who may be exposed to airborne radioactivity. Intake of airborne radionuclides is measured by this process.

5.3 Measurements

5.3.1 Radiation Levels

The measured approximate radiation levels near the spent fuel pool are as follows:

Location	Dose Rate (mrem/hr)
On SFP Crane above pool area	10.0
At handrail around pool	2.0
Other areas	1.0

The major radionuclides contributing to this exposure are Co^{58} & Co^{60}.

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The radiation levels experienced at the surface of the spent fuel pool are due primarily to the radioactive contaminents in the pool water. Measurements conducted at the Morris storage facility for PWR fuel, similar to the Prairie Island fuel, are shown in Figure 10. These show that for water levels more than eight feet above the fuel there is no significant change in radiation level due to the fuel. The normal surface level at Prairie Island is 25 feet above the fuel; thus there would have to be a significant level change before any change is expected in the radiation level.

5.3.2 Airborne Radioactivity

Samples from the SFP area usually only contain tritium as the detectable nuclide. Weekly and monthly integrated radioactivity samples are taken from the SFP ventilation system. For 1979, the following radioactivity releases were measured:

Nuclide	Total Activity, Ci	Sample Frequency
$1H^3$	141.3	Monthly
Xe^{133}	35.3	Weekly
Xe^{135}	1.09	Weekly
I^{131}	1.16 E-3	Weekly
Particulate	0	Weekly

To our knowledge, there has not been detectable Kr^{85} in the SFP air. These findings are consistent with those presented in Reference 7, where the NRC reported that experience at the NFS West Valley reprocessing plant showed that almost all of the Krypton was retained in the fuel until its dissolution during the reprocessing operation.

5.3.3 Measured SFP Liquid Activity

The following is a summary of SFP liquid activities (in microcuries per milliliter) during different periods of operation:

Radionuclide	Before[1] Refueling	After[2] Refueling	Recent[3] Data
Cs^{134}	4.08E-04	*	*
Cs^{137}	4.01E-04	*	*
Sb^{124}	4.09E-04	5.04E-03	5.11E-04
Co^{58}	2.25E-03	2.16E-02	5.32E-04
Co^{60}	3.38E-03	4.05E-03	1.98E-03
Na^{24}	4.18E-05	1.13E-04	*
Co^{57}	*	7.76E-05	*
Sb^{125}	*	3.12E-04	*

Notes

1 Data taken 27 March 79 prior to Unit 1 Reload 4 Refueling
2. Data taken 1 May 79 after the Unit 1 Reload 4 Refueling
3. Data taken 29 August 79
* Not detected

Sample activities are determined using a Ge-Li detector with multichannel analyzer (4096 channels).

5.4 Radioactivity Control Systems

Three systems provided at the Prairie Island Nuclear Generating Plant reduce the likelihood of spent fuel radionuclides reaching the environment:

(1) Refueling cavity cleanup system
(2) Spent fuel pool cleanup system
(3) Spent fuel pool ventilation system

5.4.1 Refueling Cavity Cleanup System

During refueling operations, the refueling cavity cleanup system (Figures 11 and 12) is used to remove suspended solids. This system substantially aids in preventing crud buildup in the spent fuel pool so that there currently is no noticeable buildup of crud along the sides of the spent fuel pool.

This system consists of a single pump which takes a suction from low in the refueling cavity and discharges through filters (in a parallel or series-parallel mode) back to a higher elevation in the refueling cavity.

The filters are changed whenever radioactivity levels reach 3 R/hr or the filter dp exceeds 25 psid, conditions which indicate possible reduction in filter efficiency.

5.4.2 Spent Fuel Pool Cleanup System

The spent fuel pool cleanup system is a bypass loop around the spent fuel pool heat exchangers consisting of a demineralizer, several filters, and associated piping, valves and fittings. A skimmer loop augments the cleanup loop and consists of a skimmer intake, strainer, pumps, filter, piping, valves, and fittings. The cleanup and skimmer loops provide turbidity and soluble radioactivity control via the filters and demineralizer, respectively. The systems are shown in Figures 13 and 14.

Two spent fuel pits have combined volume of 355,617 gallons. One pit contains approximately 2-1/2 times the water volume capacity of the other pit. Approximately 60 gpm of pool water is diverted, when necessary, through the cleanup loop. One spent fuel pool water volume can therefore be processed in less than 100 hours.

The spent fuel pool filter is a cartridge type filter designed to retain 98% of particles of sizes down to as small as 5 microns.

The filter is designed for a maximum differential pressure of 5 psi at rated flow. Local pressure indicators are located upstream and downstream of the filter to monitor loop pressure.

Two parallel pre-filters, normally operated singly, located upstream of the demineralizer, are designed to reduce the effect of insoluble particles on the demineralizer operation.

The demineralizer is a deep bed H-OH type containing 20 cubic feet of resins. Spent resins are flushed to a resin storage tank of the waste disposal system. Demineralizer resins are changed when pool water samples indicate reduced decontamination effectiveness.

The spent fuel pool water is sampled to determine radioactivity levels. The purification loop is placed in service whenever a significant concentration of radioactive materials exist. The purification loop is also placed in service during any refueling operations.

Radioactivity and turbidity in the fuel pool water are caused by the release of crud buildup on spent fuel assemblies. Operating experience has shown that the greatest quantity of crud is released when loose deposits are dislodged during movement of spent fuel assemblies. Once the handling of spent fuel assemblies is finished the addition of materials and radioactivity to the pool water is greatly diminished.

Failed fuel, if present, may add some fission product contamination to the spent fuel pool. This contamination is proportional to the amount of failed fuel present. This addition of contamination is much smaller than the crud added to the pool by refueling operations. Consequently, a change in the capacity of the fuel pool does not overburden the cleanup system which has been sized to remove the impurities resulting from refueling operations.

No changes or equipment addition to the cleanup loop are necessary for the augmented storage facility. As no change in refueling frequency is anticipated, the frequency of operation of the cleanup loop is not expected to change.

The redesign of the SFP racks increases only the storage capacity of the pool and not the frequency or the amount of the core to be replaced for each fuel cycle. Thus, the amount of corrosion product nuclides released into the pool during any year will be about the same regardless of the length of time or number of assemblies stored in the pool.

5.4.3 Spent Fuel Pool Ventilation System

The normal and special spent fuel ventilation systems are described in Section 9.6 of the FSAR. The fuel handling accident, (described in Section 14.2.1 of the FSAR), which forms the design basis for the special ventilation system, is not affected by the expansion of the fuel pool. The normal ventilation system has no safeguards functions and is designed to isolate upon detection of high radiation in the fuel pool area.

5.5 Radiation Monitors

The radiation monitors used at Prairie Island are described in Section 11 of the FSAR. The following is a summary of those process and area radiation monitors appropriate to this modification.

5.5.1 Process Radiation Monitors

Radiation process monitors (R-25 and R-31) are installed in the SFP normal ventilation exhaust ducting. Upon sensing high radionuclide concentrations in the air leaving the spent fuel pool area, the SFP normal ventilation system (Figure 15) shuts down and SFP special ventilation system starts which discharges through the spent fuel special and inservice purge PAC filters to the shield building exhaust stack (Figure 15).

5.5.2 Area Radiation Monitors

A criticality radiation monitor (R 28) is installed on the spent fuel/new fuel storage facility operating deck. This monitor would alarm in the unlikely event of criticality in the area.

Six additional radiation area monitors are installed in close proximity to the spent fuel storage facility that could alert the operations staff that abnormal radiation conditions exist. All of these radiation monitors, alarm locally and in the control room, so that corrective action may be taken promptly. These monitors are R5, R8, R29, R32, R33, and R34. R5, the spent fuel storage area monitor, is mounted on the operating deck of the storage facility. R8 is mounted in the waste gas valve gallery area below the storage facility.

R29, 32, 33, and 34 are mounted in the shipping and receiving area and the three floors of the radwaste building. G-M and scintillation detectors provide diversity in gamma detection.

If any of these area or process radiation monitors alarm, there is an individual qualified in radiation protection procedures on site to direct the appropriate corrective actions.

All of these alarms provide defense-in-depth to a design which assures that Keff $\leq$ 0.95 even if pure water and new fuel are used in the pool.

5.5.3 Continuous Air Monitor (CAM)

A portable CAM unit is normally located in the area near the SFP ventilation system to detect airborne particulates and iodine. This unit is normally monitored by the Auxiliary Building operator during routine shift equipment checks.

1877 063

5.6 Radiation Protection Practices

The PINGP radiation protection program is formulated to maintain radiation exposures ALARA (As Low As Reasonably Achievable). Work in the controlled area, of which the SFP is a part, is governed by radiation work permits. The radiation work permit (RWP) is used to identify protective requirements, such as Anti-C clothing, respiratory equipment, dosimetry, special instructions, etc., for work in the controlled area. The radiation protection specialist prepares the RWP for jobs performed in the controlled area and attempts to assure that the radiation exposures will be ALARA. In addition, all plant personnel are recommended to look at their work at the plant in order to maintain exposures ALARA.

Utilizing these principles has helped and will continue to help in ensuring that the man-rem exposures associated with this modification are ALARA.

5.7 Technical Specifications

The PINGP technical Specifications, Section 3.9, impose limits on releases of radioactive effluents. These specifications include releases associated with the operation of the SFP storage facility and provide additional assurance that there will not be any significant impact on offsite radiation exposures as a result of this modification.

5.8 Conclusions

Sections 5.1-5.7 describe the review of the radiological impact of this modification. As a result, we conclude that there will not be a significant impact on radiation exposures either onsite or offsite.

6.0 Nonradiological Impact Evaluation

6.1 Nonradiological Effluents

There will be no change in the chemical or biocidal effects from the plant as a result of the proposed modifications.

Thermal impact of this modification was addressed in Section 3 of this safety evaluation.

6.2 Impact on the Community

The new racks will be fabricated offsite and shipped to the plant. No environmental impact on the community is expected to result during or after completion of this modification.

7.0 Accident Evaluation

7.1 Heavy Loads Analysis

The existing technical specifications preclude handling of any heavy loads over or in either spent fuel pool when fuel is stored in that pool. The consequences of fuel handling accidents are unchanged from those presented in the FSAR. For the purpose of completing the modification, the footnote added at the bottom of page TS.3.8-2 allows movement and placement of loads described in the installation procedures for this modification as described in Exhibit C. The small pool covers are designed to sustain a heavy load drop (see Exhibit E) without affecting the fuel in the small pool.

7.2 Fuel Handling Accidents

The FSAR Section 14.2.1 addresses fuel handling accidents. The conclusions presented in that report are unchanged by this modification. Fuel assembly drop accidents are also addressed in Section 3.3.4 of Exhibit C.

7.3 Cask Drop Accidents

The FSAR section 9.5 provides a description of the cask drop accident. Cask drop is also described in Section 3.3.4 of the Exhibit C report.

7.4 Conclusions

The existing technical specifications and analyses provide assurance that this modification will not adversely affect the public health and safety.

8.0 Procedural Impact Evaluation

The fuel rack installation sequence and a summary procedure are described in section 3.7 of the Exhibit C report. The fuel handling procedures as previously reported in the FSAR Section 9.5 will not need to be changed as a result of this modification.

As noted in the separate safety evaluation for the proposed change in Technical Specification 3.8, the plant Operations Committee will review the specific fuel rack installation procedures for this modification.

9.0 Summary

Reference 1 described the information required for the NRC to conduct a review of license amendment requests involving spent fuel storage facility modifications. This report has addressed the two general areas of staff review - (1) Safety Evaluation Report, (2) Environmental Impact Appraisal.

Evaluation of the nuclear, thermal-hydraulic, mechanical, material, structural, and environmental aspects of this modification that are presented in this report provide assurance that there will be no significant effect on the public health and safety. This report addresses Sections III through V of Enclosure 1 to Reference 1. This spent fuel storage capacity increase license amendment request must be approved on a timely basis (September 1980) to assure continued operation of the Prairie Island Nuclear Generating Plant.

10.0 References

1. Letter, B K Grimes (NRC) to All Power Reactor Licensees, dated April 14, 1978. Enclosure 1 "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications".

2. Letter and Enclosures, A Schwencer (NRC) to S Burstein (WEP) authorizing amendments 35 (DPR-24) and 41 (DPR-27), dated April 4, 1979.

3. Letter and Enclosures, A Schwencer (NRC) to J Dolan (I & MEC) authorizing amendments 32 (DPR-58) and 13 (DPR-74) dated October 16, 1979.

4. Letter and Enclosures, K Goller (NRC) to L O Mayer (NSP) authorizing amendments 22 (DPR-42) and 16 (DPR-60), dated August 16, 1977.

5. Battelle Pacific Northwestern Laboratory Report PNL-3065, Commentary on Spent Fuel Storage at Morris Operation, K J Eger and G E Zima, July 1979.

6. Letter, L O Mayer (NSP) to Victor Stello (NRC), dated November 24, 1976 Exhibit A.

7. Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel, NUREG 0575, August 1979.

1877 069

11.0 Tables

TABLE 1

PRAIRIE ISLAND SPENT FUEL STORAGE

HISTORICAL DATA AND FORECAST

PRESENT SPENT FUEL STORAGE ARRANGEMENT

REFUELING DATE	UNIT II ASSEMBLIES	UNIT I ASSEMBLIES	TOTAL TO DATE	TOTAL NO.[6] REMAINING SPACES	% OF TOTAL[6] SPACES OCCUPIED
1980	40	40	360	327	52
1981	40	41[1]	441[2]	245	64
1982	41[1]	40	522[3][4]	165	76
1983	40	40	602	85	87
	Full Core Storage for 121 Assemblies is no Longer Available				
1984	40	40	682	5	99

PROPOSED SPENT FUEL STORAGE MODIFICATION

(to be complete in late 1981)

REFUELING DATE	UNIT II ASSEMBLIES	UNIT I ASSEMBLIES	TOTAL TO DATE	TOTAL NO. REMAINING SPACES	% OF TOTAL SPACES OCCUPIED
1982	40	40	522	857	38
1983	40	40	602	777	44
1984	40	40	682	697	49
1985	40	40	762	617	55
1986	40	40	842	537	61
1987	40	40	922	457	67
1988	40	40	1002	377	73
1989	40	40	1082[5]	297	78
1990	40	40	1162	217	84
1991	40	40	1242	137	90
1992	40	40	1322	57	96
1993	40[7]	40	1402		100

NOTES

(1) Exxon delivers 201 fuel assemblies for each unit on their 5-year contract.
(2) Maximum contemplated capacity of Pool 1 with high density spent fuel storage racks - 462 assemblies. The modification can be made during Summer and Fall 1981 with all fuel stored in Pool 1.
(3) Pool 2 capacity - 555 assemblies.
(4) The next refueling discharge will require storage of spent fuel in Pool 1. Presently our license does not allow handling heavy objects above a pool storing irradiated fuel. Pool 1 is in the crane movement pathway and also is required for shipping spent fuel.
(5) Pool 2 modified capacity - 1120 assemblies. Crane modification required to move heavy loads over Pool 1 after next refueling. (See Note 4)
(6) Pool 1 and 2 total modified capacity - 1582 assemblies. However, four modules must be removed from Pool 1 for spent fuel shipping cask handling. Therefore, only 1379 storage locations are available for normal storage.
(7) After the first refueling in 1993, normal storage capacity will be exhausted.

12.0 Figures

1877 072

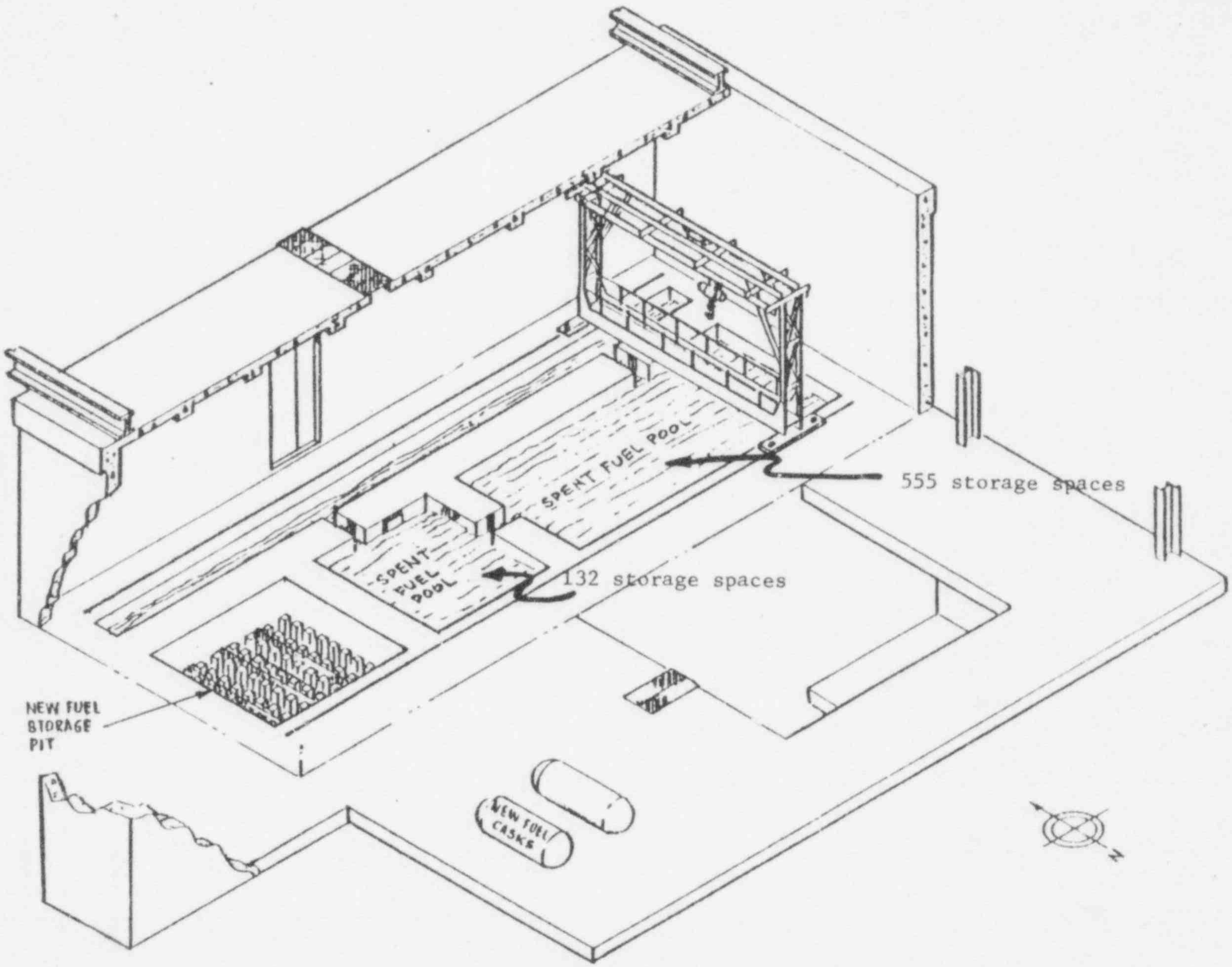


FIGURE 1: PINGP FUEL STORAGE FACILITY

1877 073

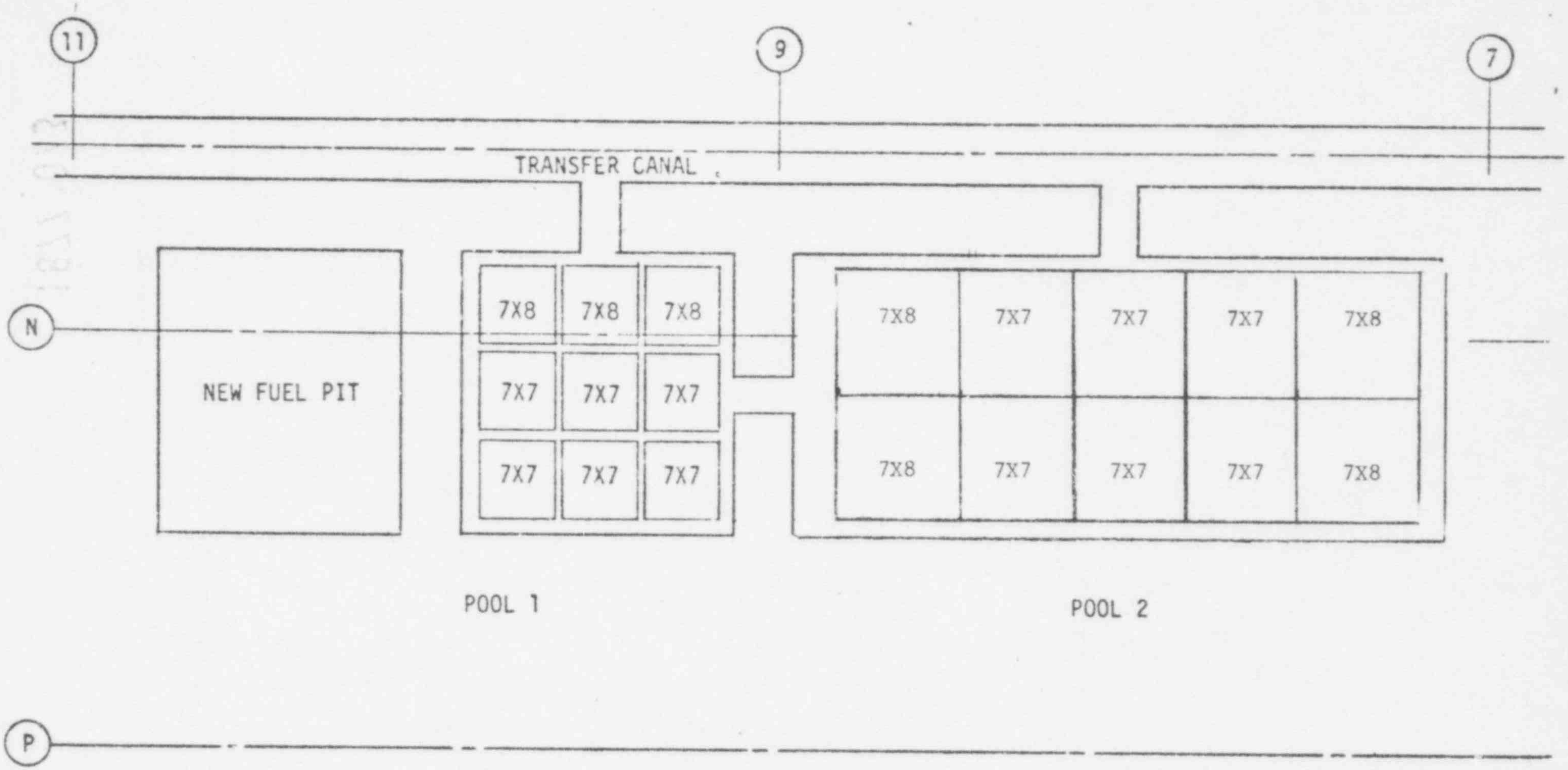
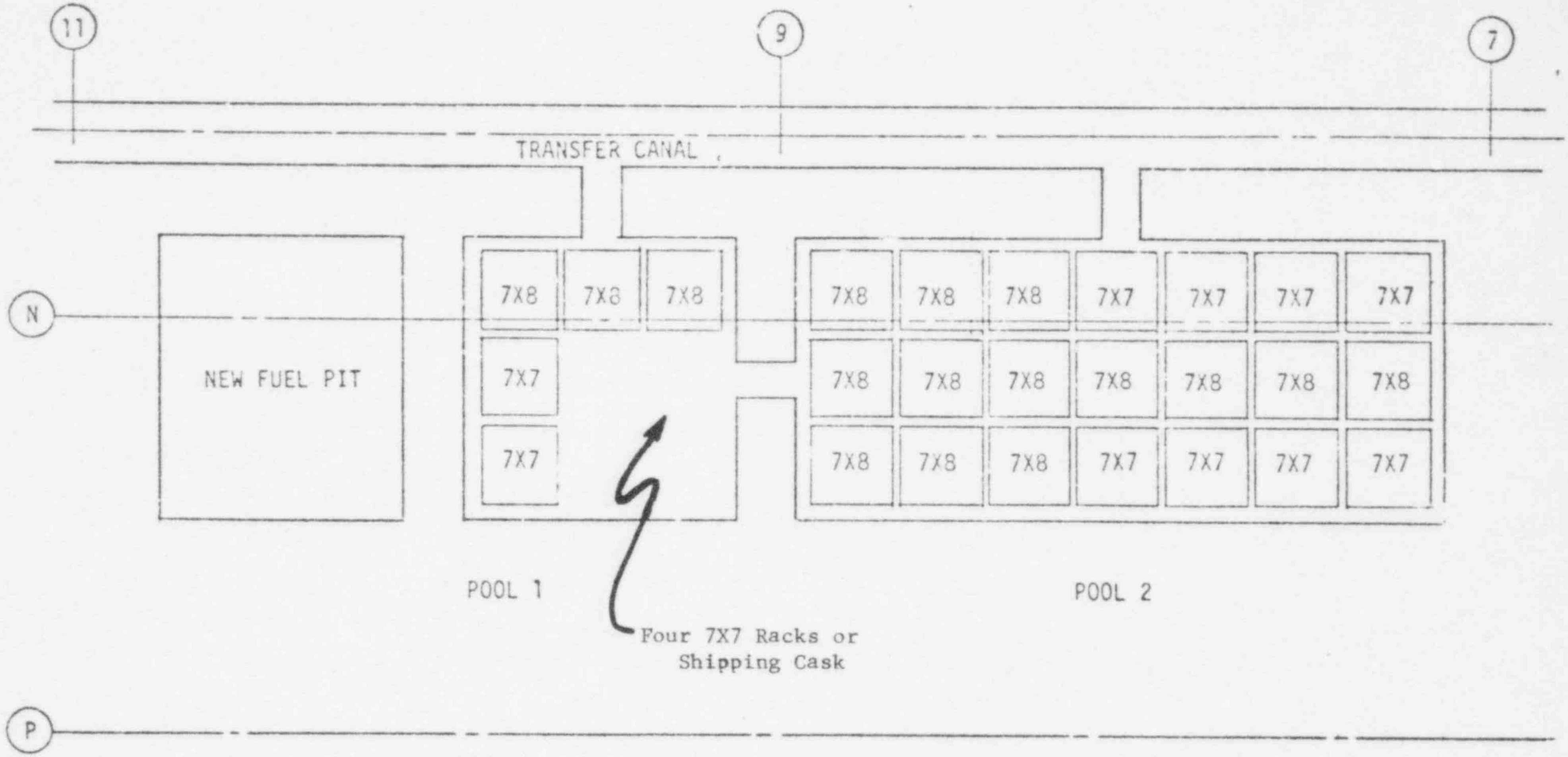


FIGURE 2: INTERIM STORAGE CONFIGURATION (Early 1981- after rerack of Pool 1)

1877 074



1386 STORAGE SPACES

1877 075

FIGURE 3 FINAL STORAGE CONFIGURATION

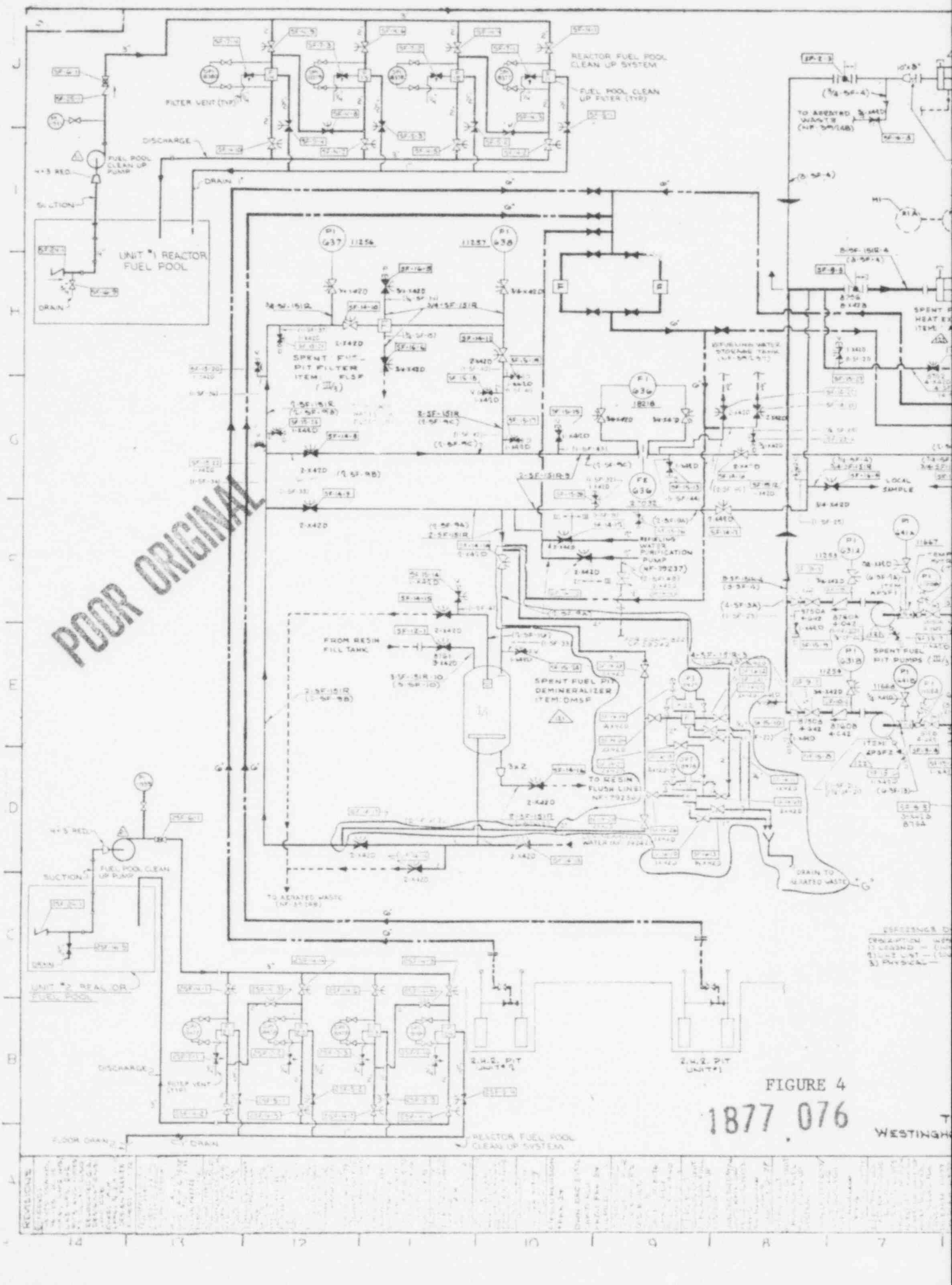
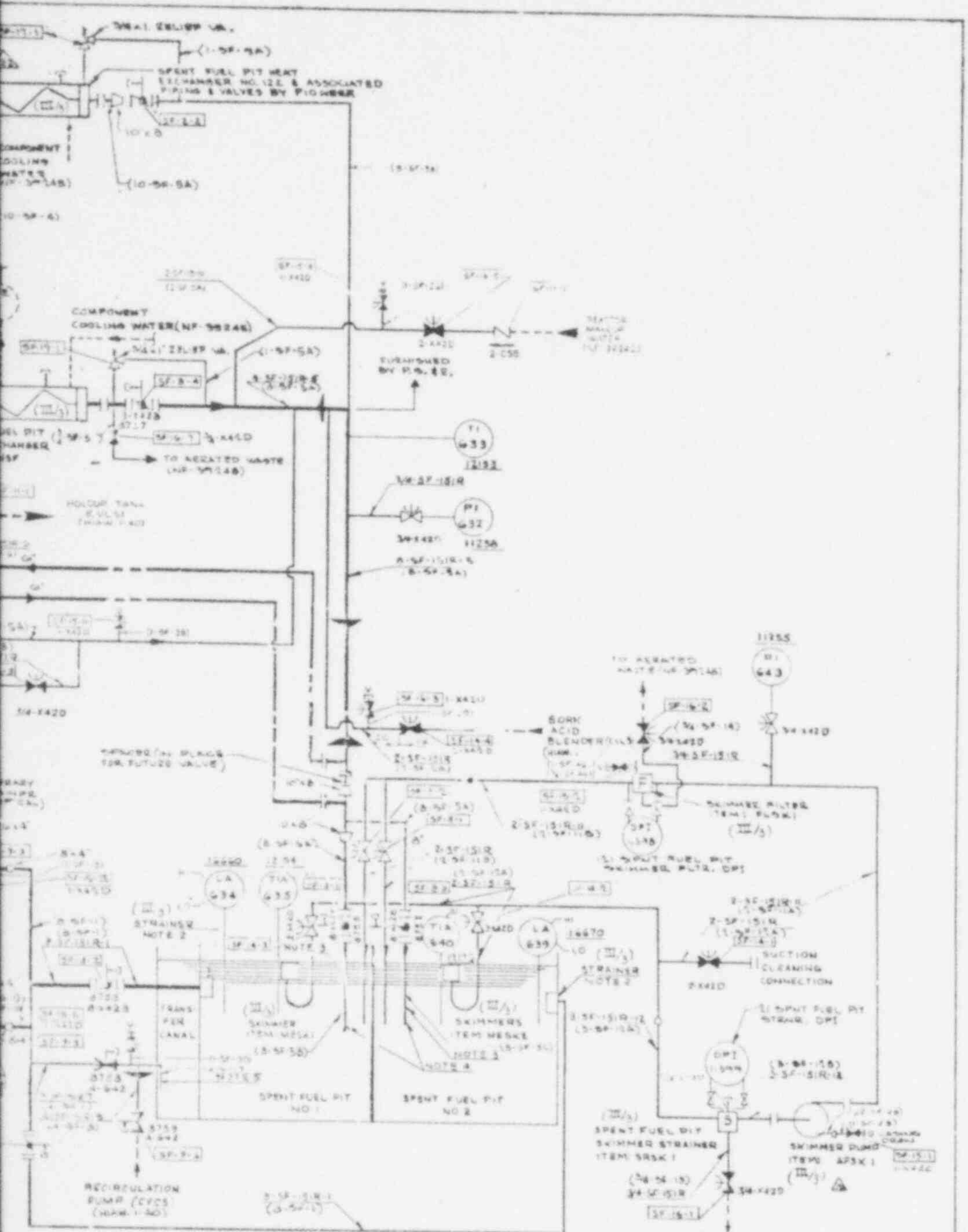


FIGURE 4

1877 076



NOTE :

1. ALL ITEM NOS INCLUDE PREFIX NSPAC.
2. LOCATE STRAINER 4 FEET BELOW NORMAL WATER LEVEL
3. DRILL 1/2 INCH HOLE 4 FEET BELOW NORMAL WATER LEVEL
4. TERMINATE PIPE 10 FEET BELOW NORMAL WATER LEVEL.
5. TERMINATE PIPE NEAR BOTTOM OF TRANSFER CANAL.
6. ALL PIPE ON THIS DWG TO BE Q.I. TYPE IC UNLESS OTHERWISE NOTED.
7. ALTERNATE PIPING BY PIONEER, INDICATES POSSIBLE FUTURE INSTALLATION IF REQUIRED.
8. NUMBERS IN PARENTHESIS DENOTE P.S. & E. LINE NUMBERS.
9. DENOTES CATEGORY I VENTILATION CONN.
   CATEGORY I
10. P - DENOTES PURGE CONN.
11. DENOTES EQUIPMENT CAT/TYPE/DESIGN CLASS

THIS DIAGRAM REPLACES
P.S.&E. DWG. NO. 110E009 REV. 6

SIGNIFICANT NO.	8630		35	100	4410	1490
GROUP	1	2	3	4	5	6

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
UNIT 1 & 2 - RED WING, MINNESOTA

FLOW DIAGRAM UNITS 1 & 2
SPENT FUEL PIT COOLING SYSTEM

Designed By
Pioneer Service & Engineering Co.

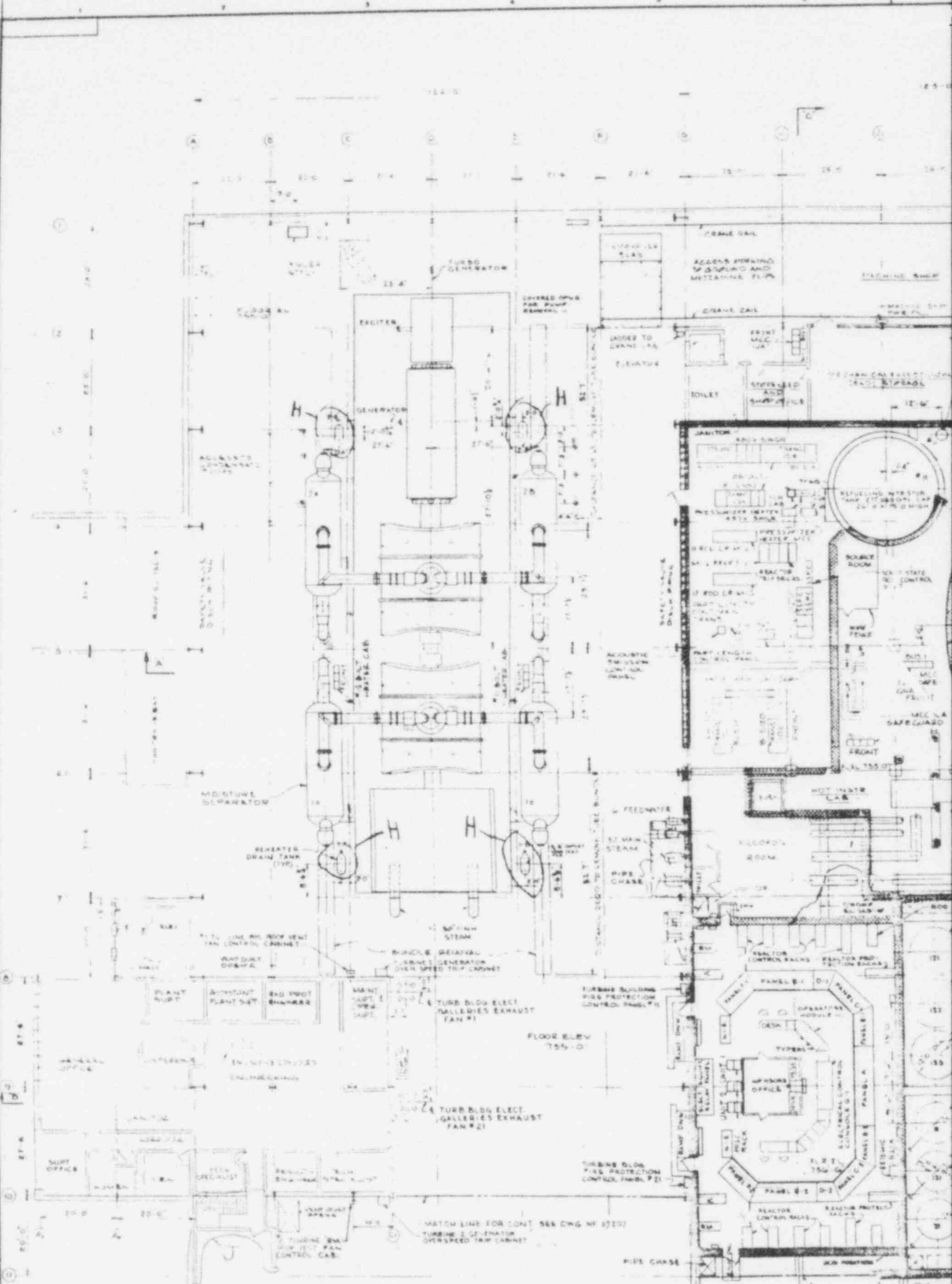
NORTHERN STATES POWER COMPANY

PROJECT NO. & DWG. NO.
21 6197

34

POOR ORIGINAL

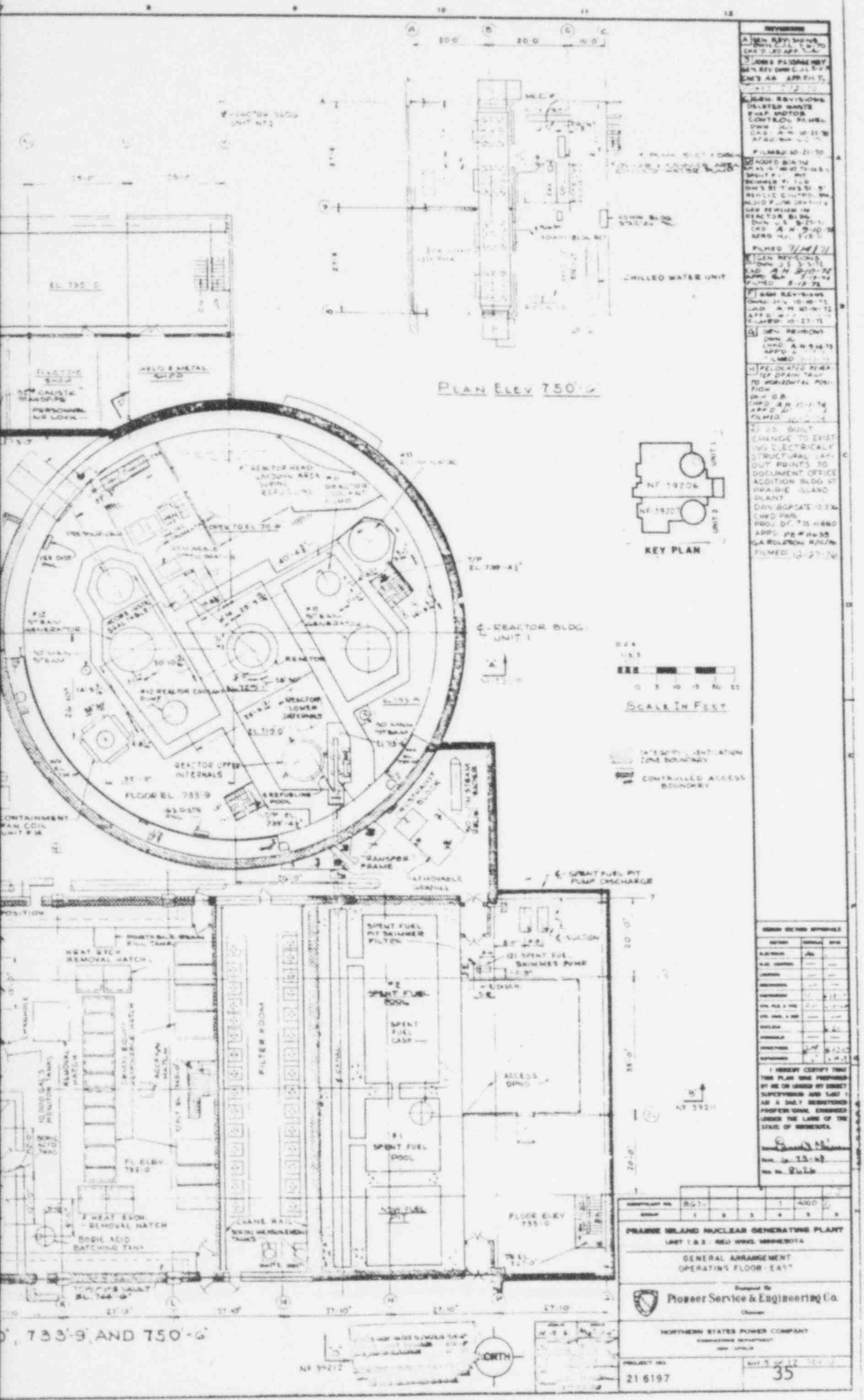
1877 077



1877 078

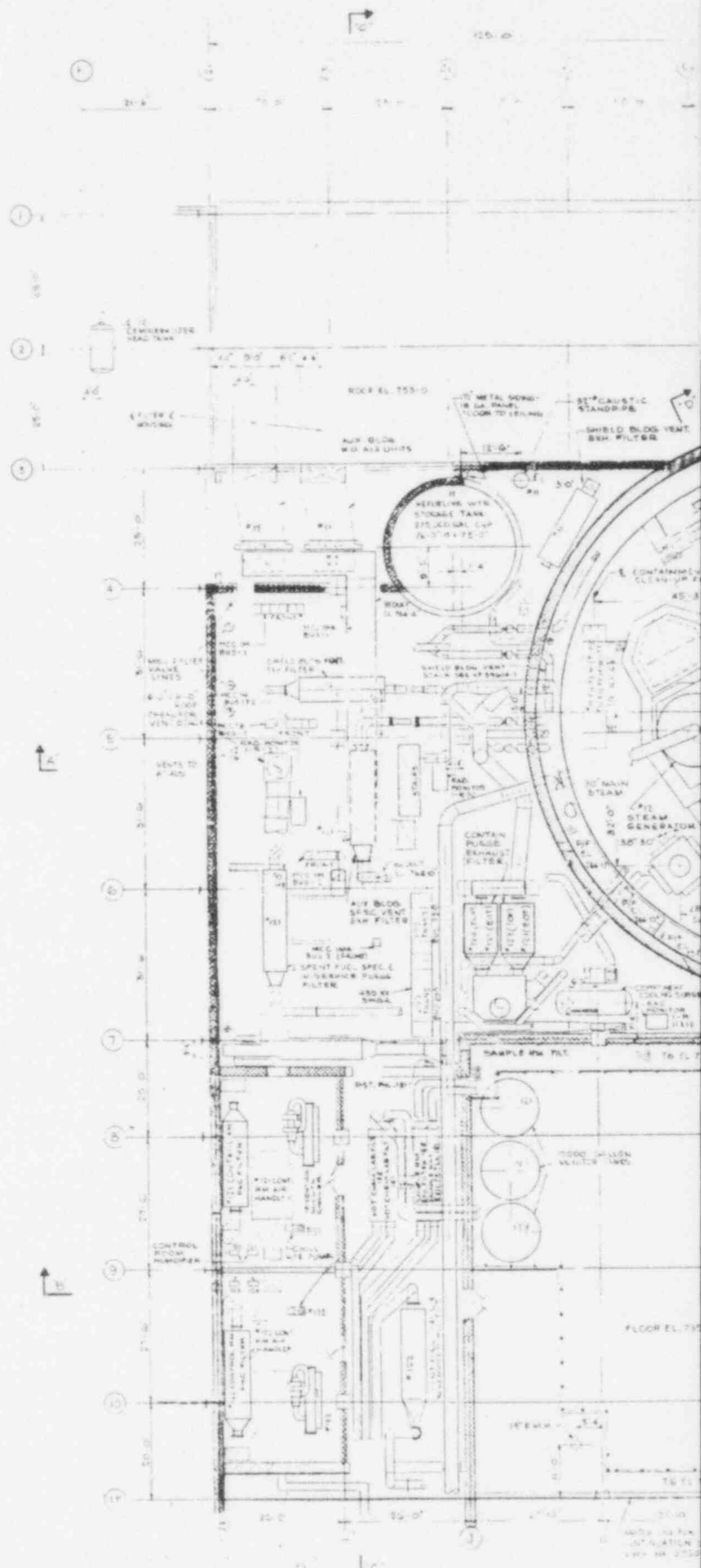
FLOOR PLAN ELEV. 735-0

FIGURE 5



POOR ORIGINAL

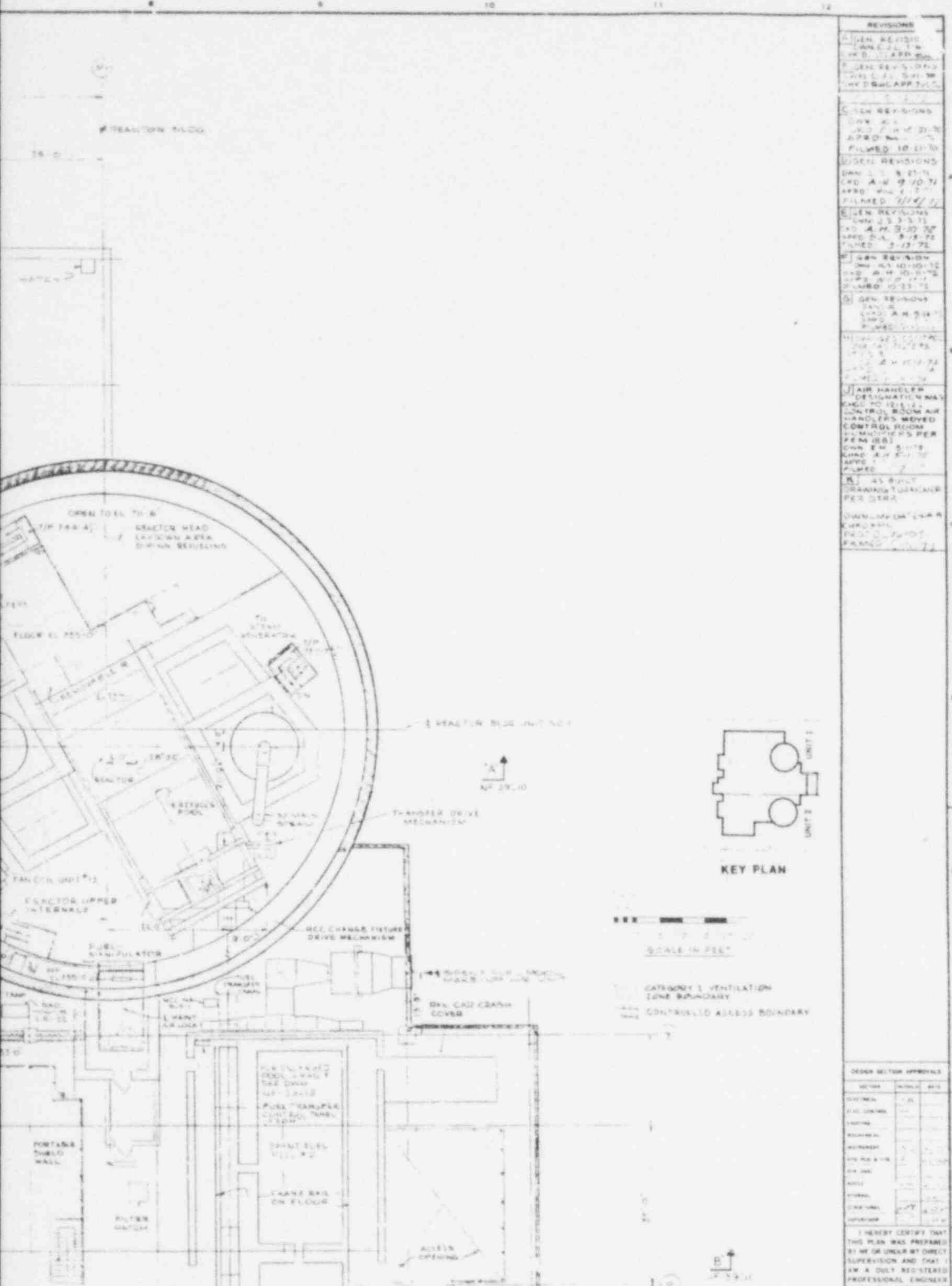
1877 079



1877 080

FLOOR PLAN EL

FIGURE 6



POOR ORIGINAL

1877 081

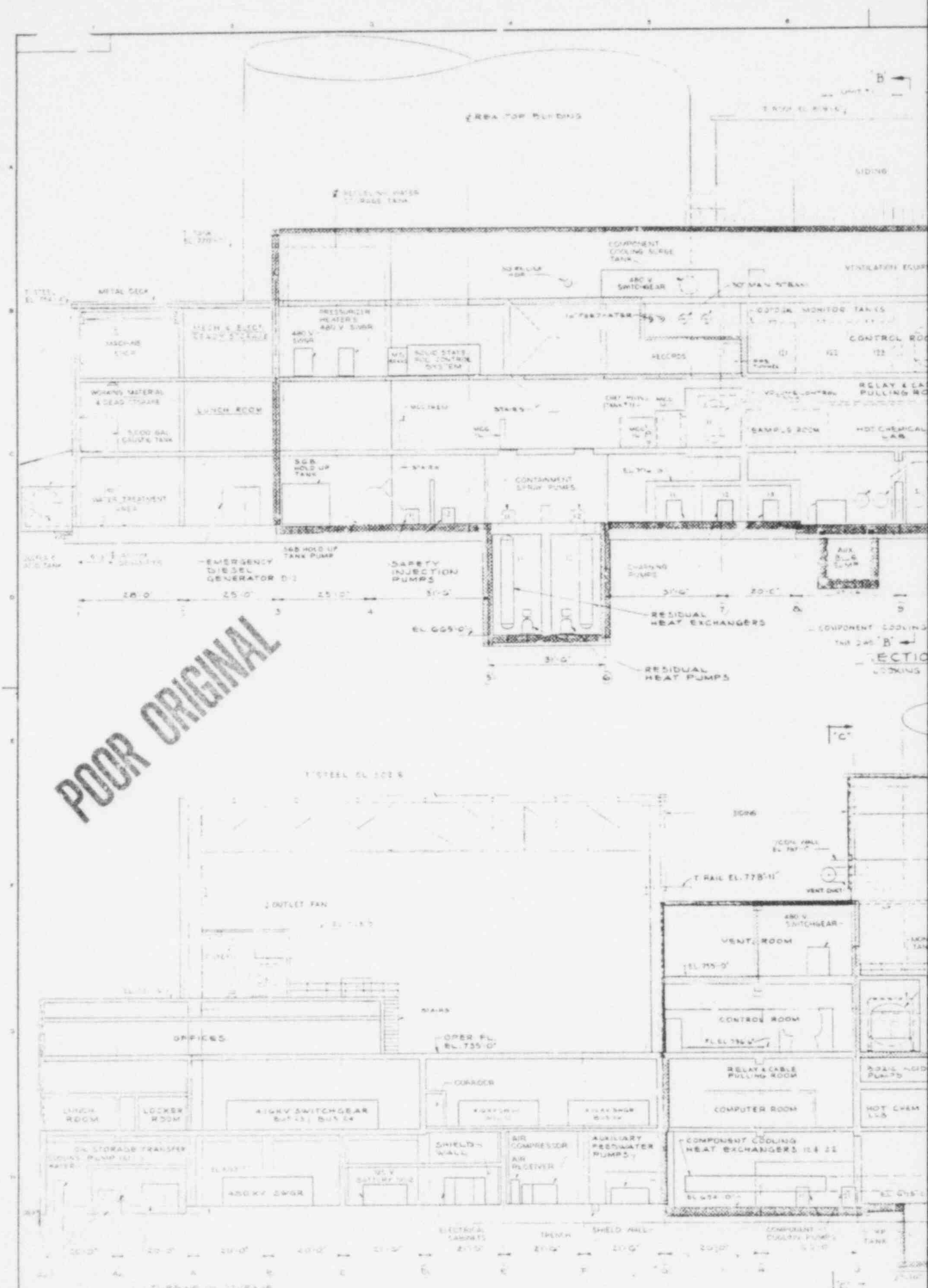
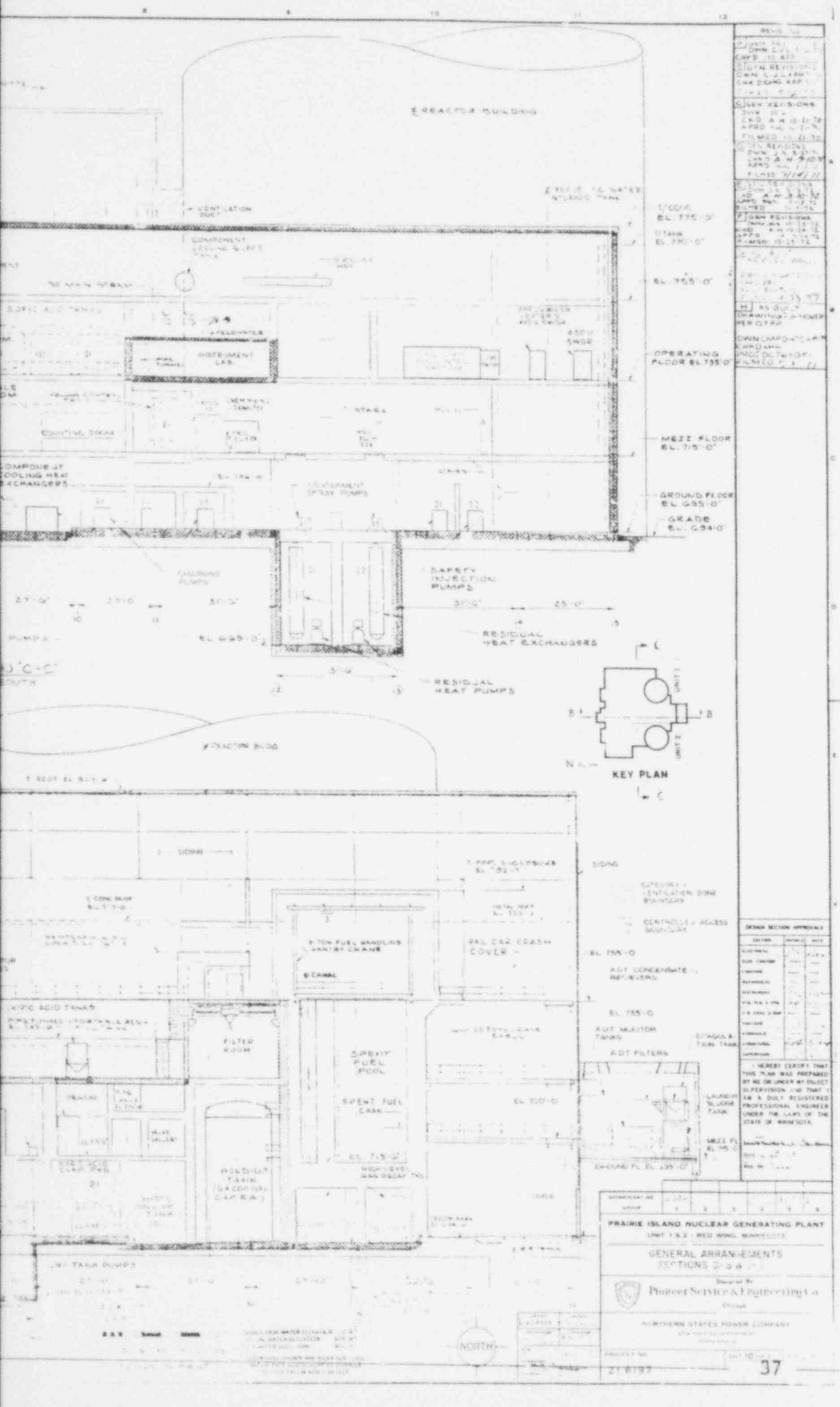


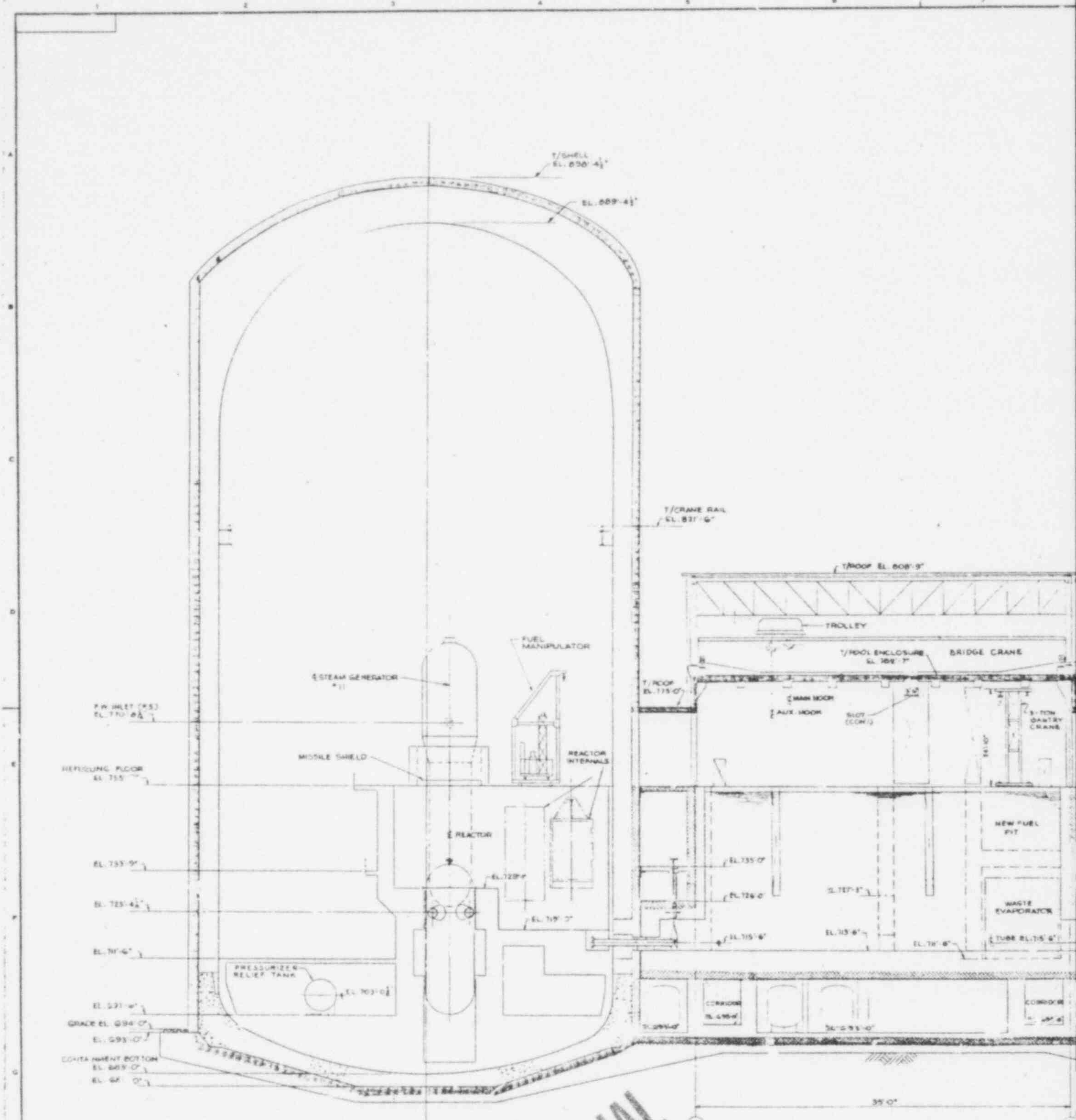
FIGURE 7

1877 082



POOR ORIGINAL

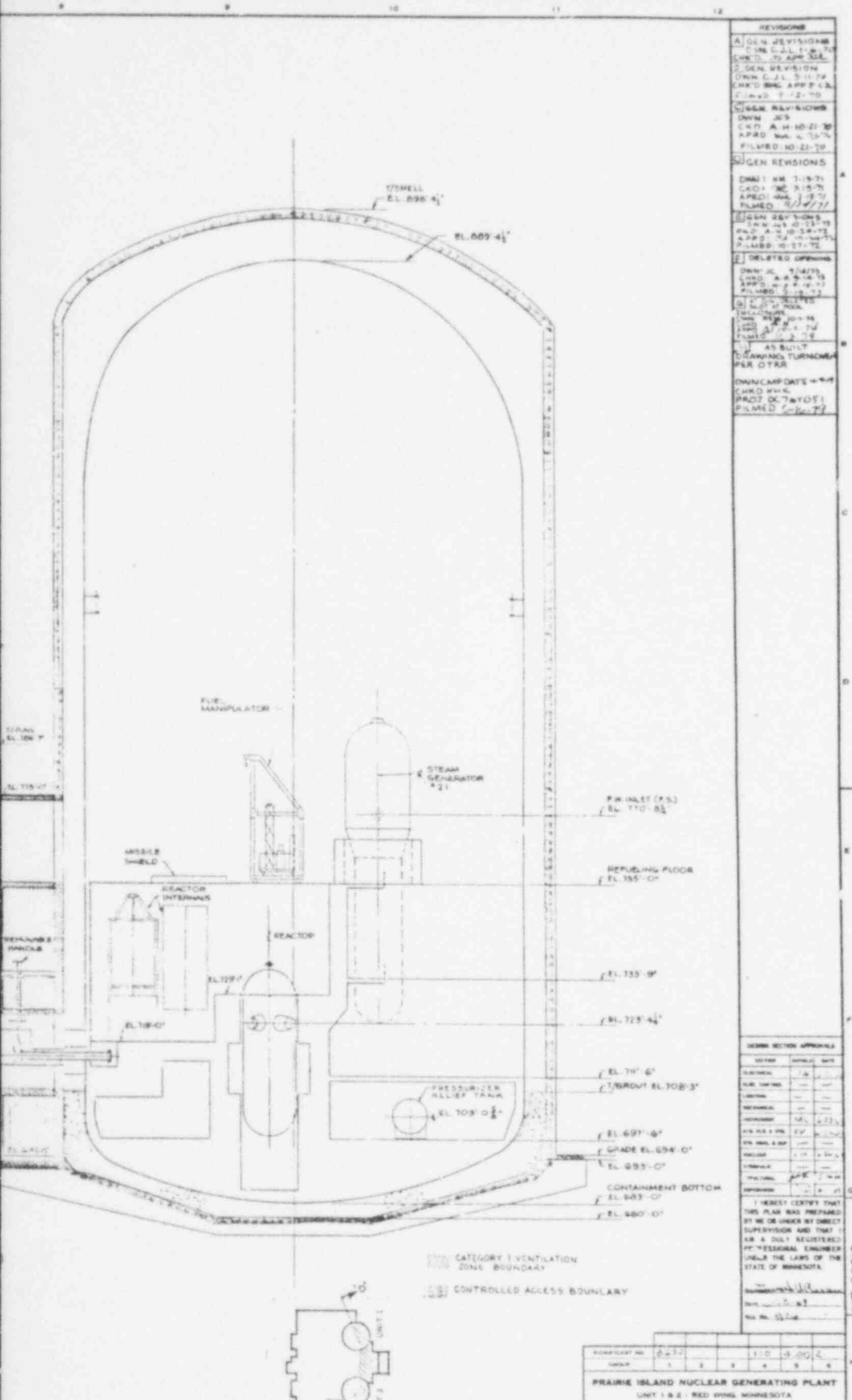
1877 083



SECTION "D-D"
LOOKING SOUTH

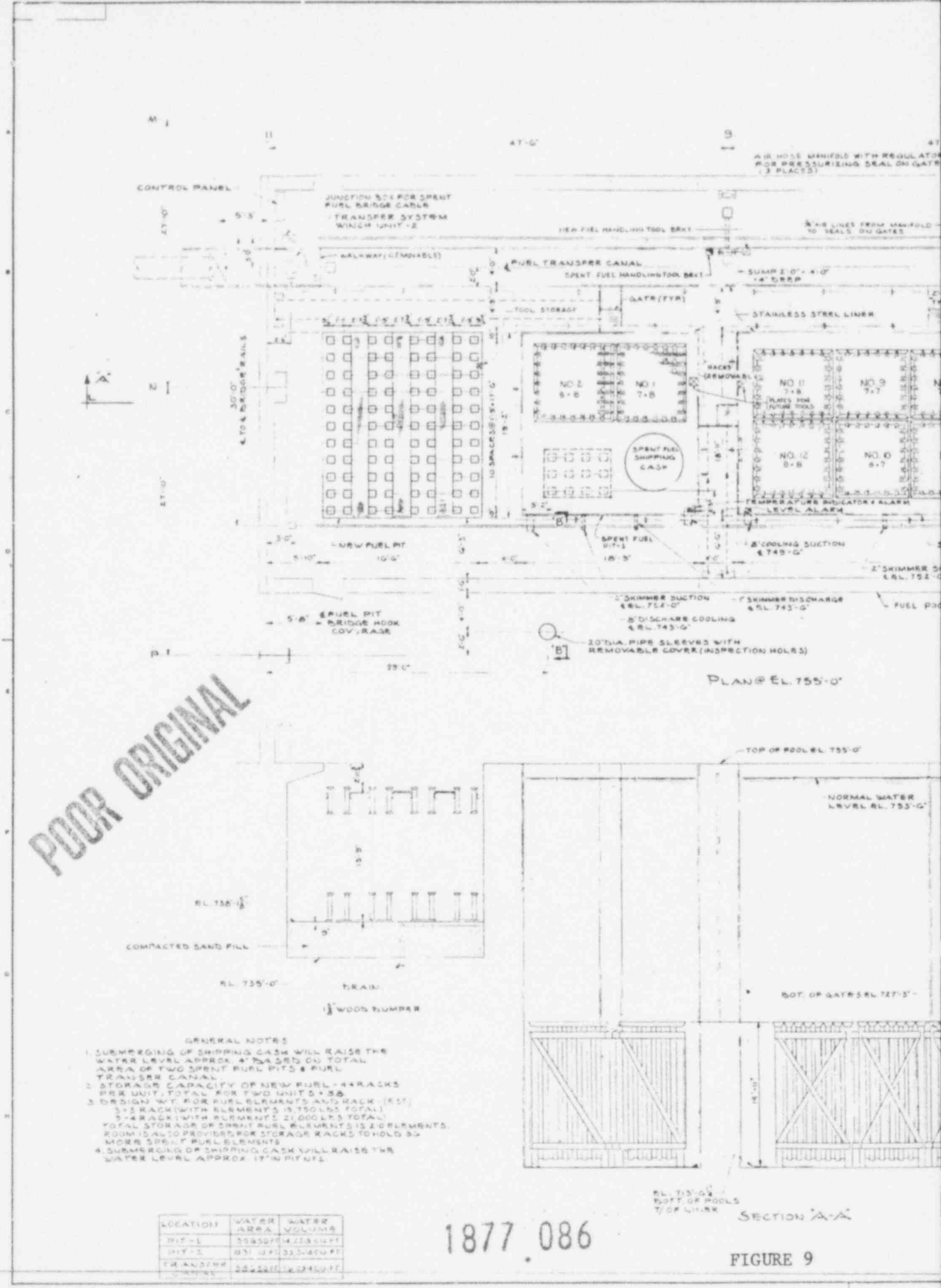
FIGURE 8

1877 084



POOR ORIGINAL

1877 085



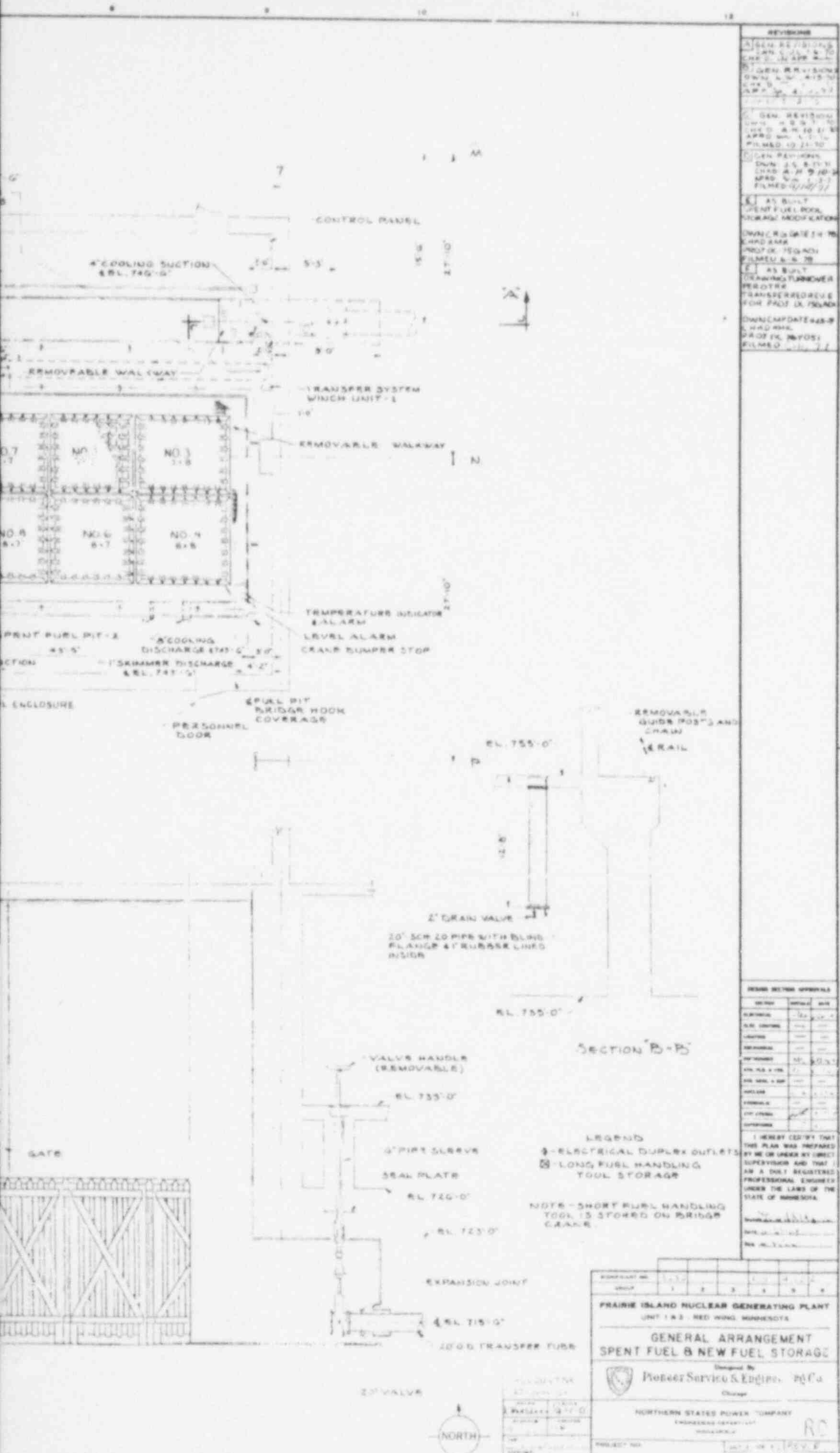
GENERAL NOTES

1. SUBMERGING OF SHIPPING CASK WILL RAISE THE WATER LEVEL APPROX. 4" BASED ON TOTAL AREA OF TWO SPENT FUEL PITS & FUEL TRANSFER CANAL.
2. STORAGE CAPACITY OF NEW FUEL - 44 RACKS PER UNIT, TOTAL FOR TWO UNITS = 88.
3. DESIGN WT. FOR FUEL ELEMENTS AND RACK (EST.)
   3x3 RACK (WITH ELEMENTS 15,750 LBS TOTAL)
   3x4 RACK (WITH ELEMENTS 21,000 LBS TOTAL)
   TOTAL STORAGE OF SPENT FUEL ELEMENTS IS 210 ELEMENTS. ROOM IS ALSO PROVIDED FOR STORAGE RACKS TO HOLD 35 MORE SPENT FUEL ELEMENTS.
4. SUBMERGING OF SHIPPING CASK WILL RAISE THE WATER LEVEL APPROX. 17" IN PIT NO. 1.

LOCATION	WATER AREA	WATER VOLUME
PIT-1	539 SQ FT	14,228 CU FT
PIT-2	831 SQ FT	33,346 CU FT
TRANSFER CANAL	352 SQ FT	12,374 CU FT

1877 086

FIGURE 9



POOR ORIGINAL

1877 087

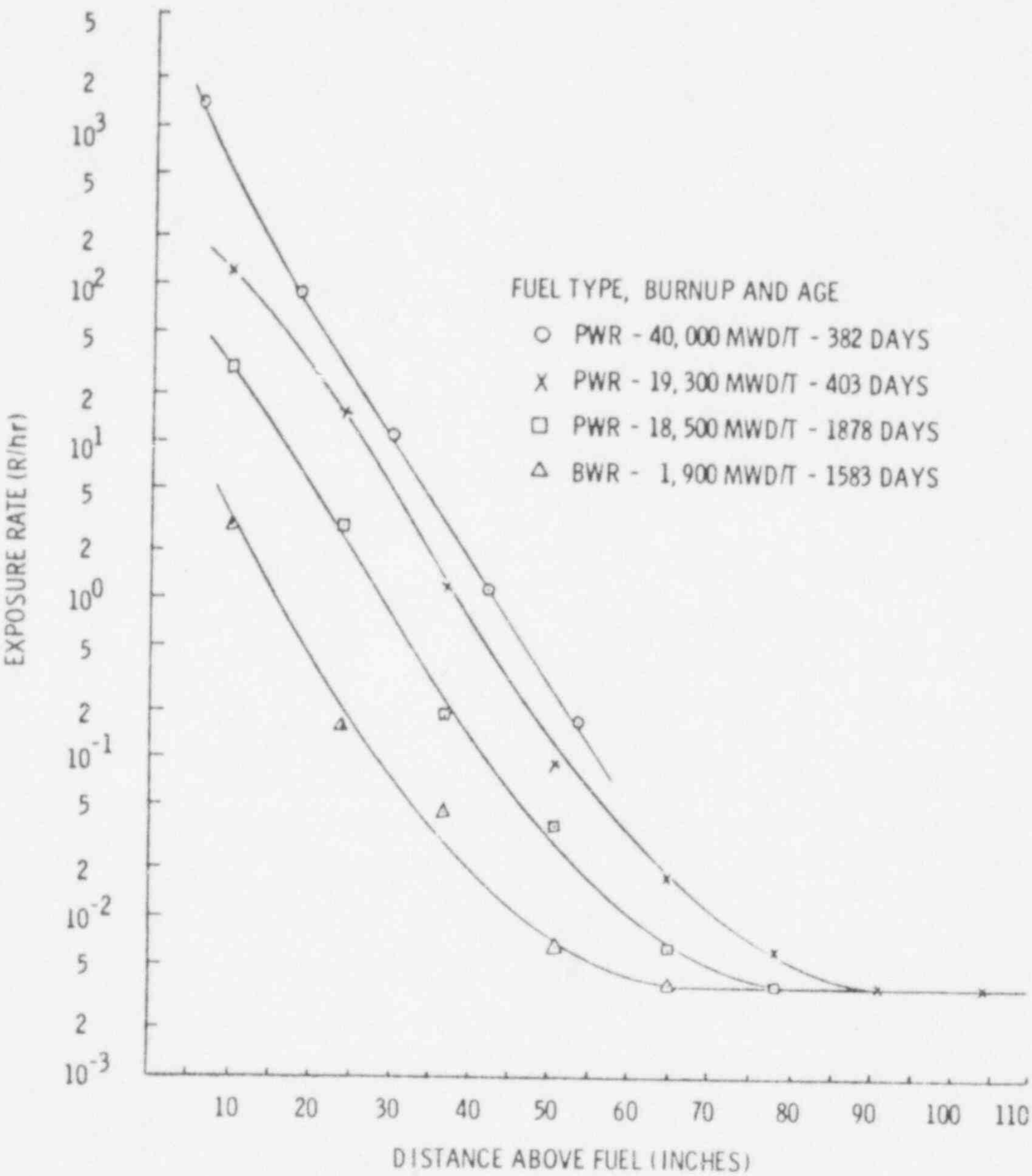


FIGURE 10 Gamma Exposure Rates Above Spent Fuel

1877 088

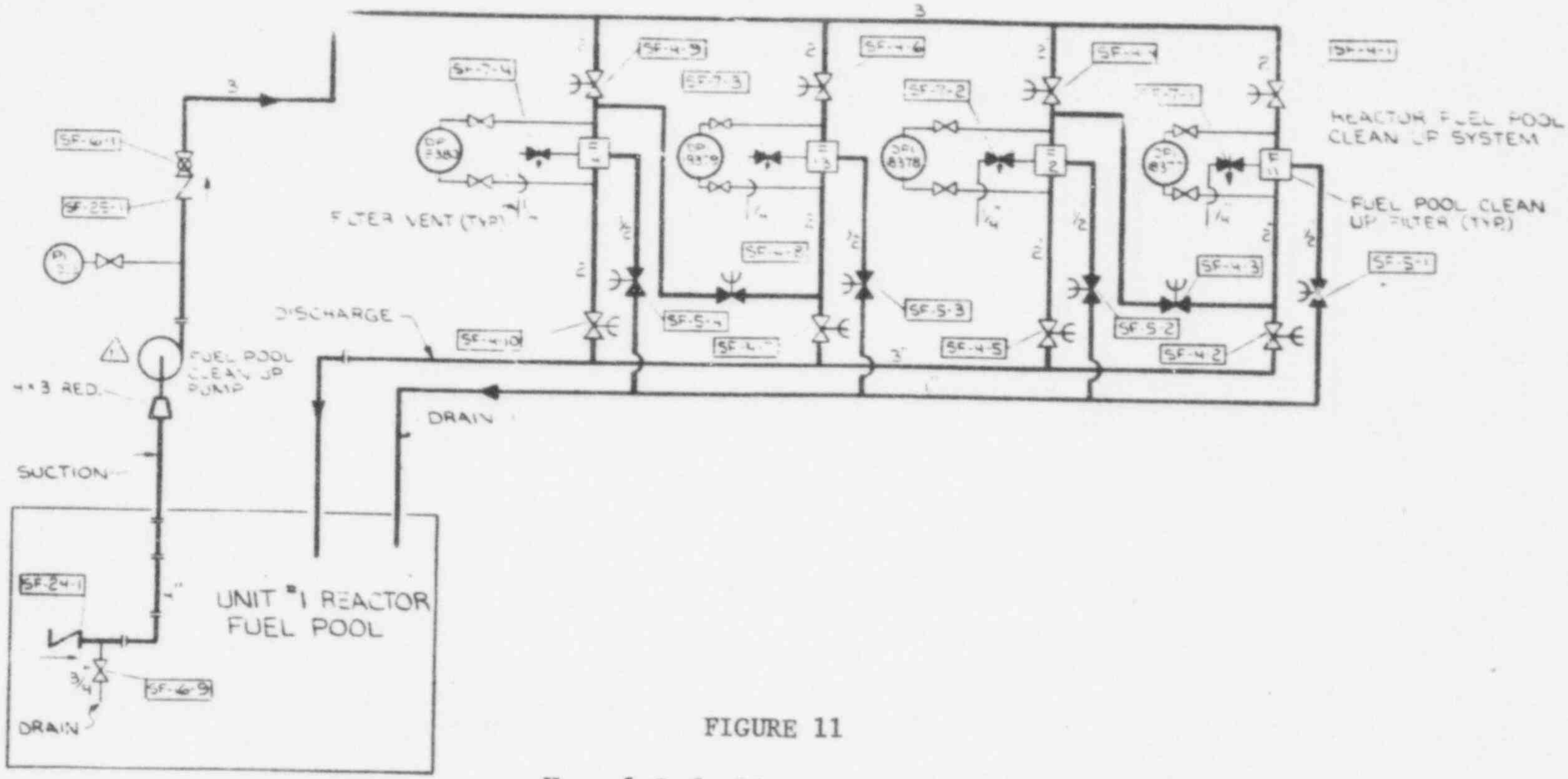


FIGURE 11

Un 1 Refueling Cavity Cleanup System

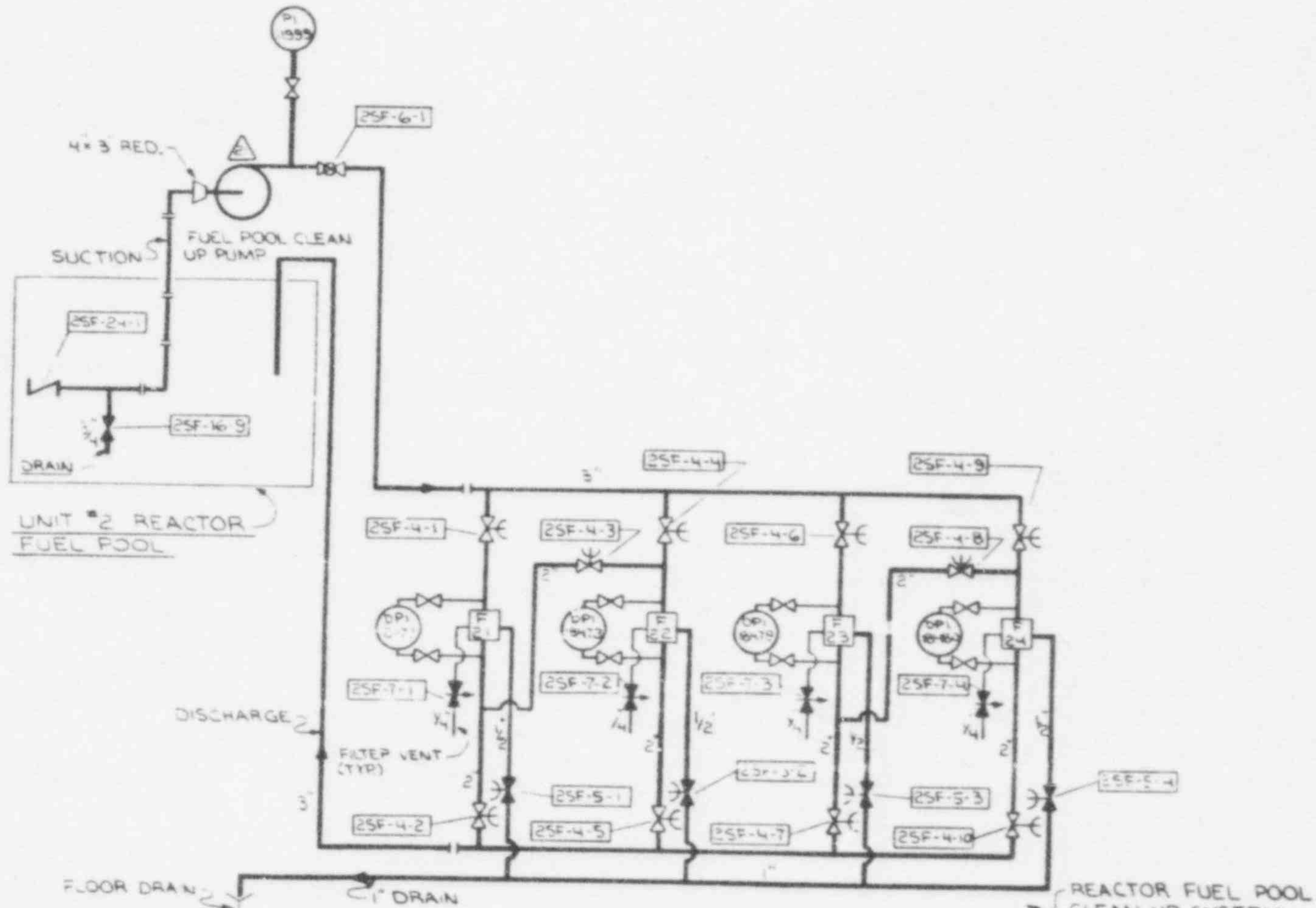


FIGURE 12

Unit 2 Refueling Cavity Cleanup System

1877 090

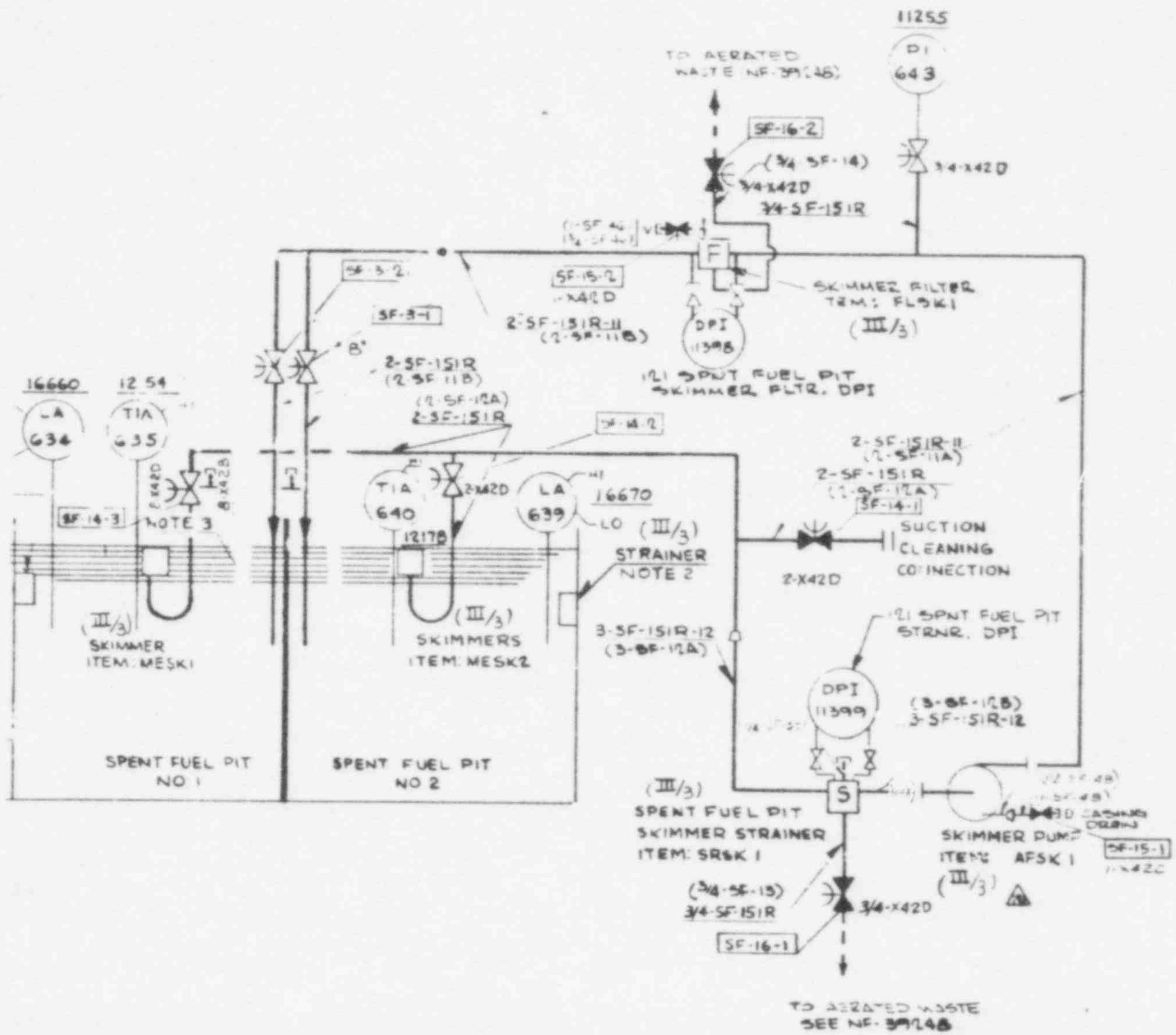


FIGURE 13

Spent Fuel Pool Skimmer System

1877 091

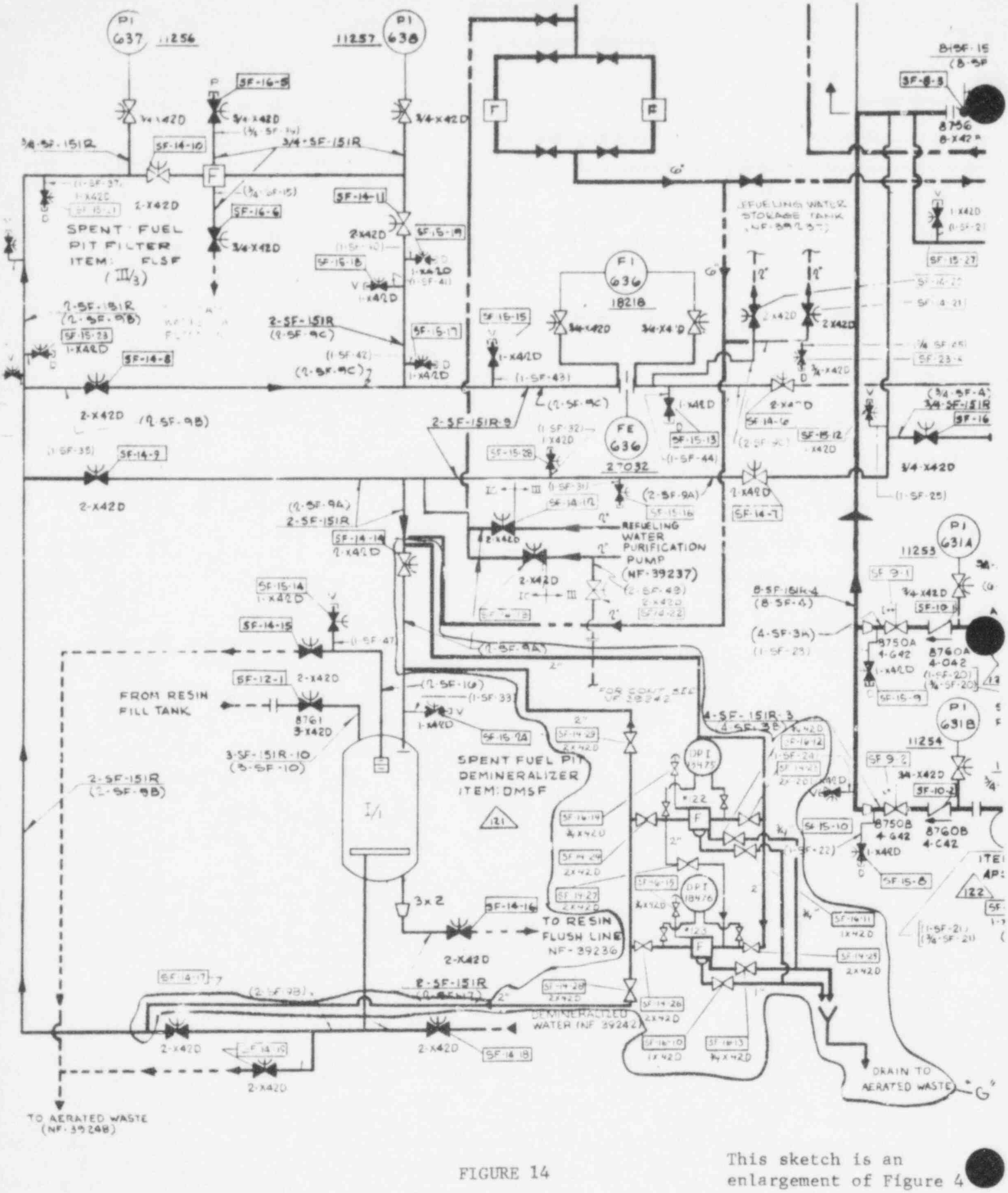
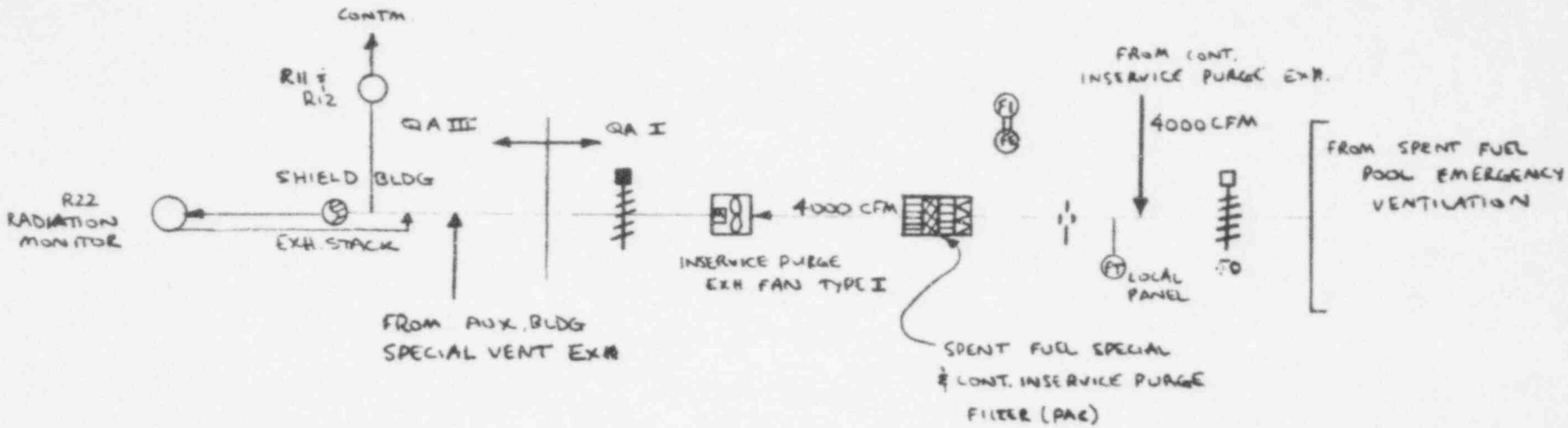


FIGURE 14

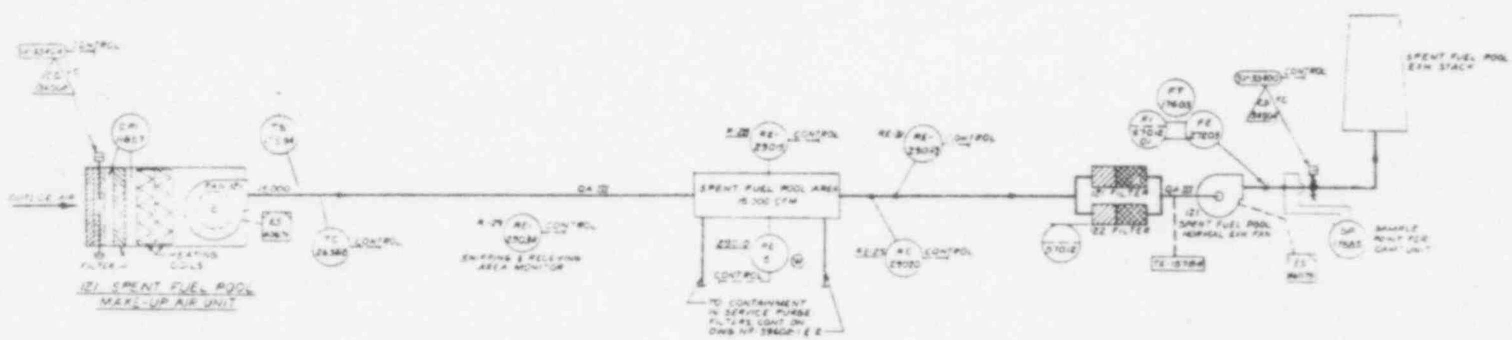
SPENT FUEL POOL CLEANUP SYSTEM

This sketch is an enlargement of Figure 4

1877 092



Spent Fuel Special Ventilation System
(typical of 2)



Spent Fuel Normal Ventilation System

FIGURE 15

Normal and Special SFP Ventilation

1877 093

EXHIBIT B

License Amendment Request dated January 31, 1980

Docket Nos. 50-282	License Nos. DPR-42
50-306	DPR-60

Exhibit B consists of revised pages of the Prairie Island Nuclear Generating Plant Technical Specifications, Appendix A, as listed below:

Pages (TS-)

3.8-2
5.3-1
5.6-1

1877 094

6. Direct communication between the control room and the operating floor of the containment shall be available whenever changes in core geometry are taking place.

7. No movement of irradiated fuel in the reactor shall be made until the reactor has been subcritical for at least 100 hours.

8. The radiation monitors which initiate isolation of the Containment Purge System shall be tested and verified to be operable immediately prior a refueling operation.

B. During fuel handling operations, the following conditions shall be satisfied:

1. No heavy loads will be transported over or placed in either part of the spent fuel pool when irradiated fuel is stored in that part.*

2. Prior to spent fuel handling in the auxiliary building, tests shall be made to determine the operability of the spent fuel pool special ventilation system inlcuding the radiation monitors in the normal ventilation system that actuate the special system and isolate the normal systems.

3. Prior to fuel handling operations, fuel-handling cranes shall be load-tested for operability of limit switches, interlocks, and alarms.

4. When the spent fuel cask contains one or more fuel assemblies, it will not be suspended more than 30 feet above any surface until the fuel has decayed more than 90 days.

C. If any of the specified conditions in 3.8.A or 3.8.B above are not met, refueling or fuel-handling operations shall cease. Work shall be initiated to correct the violated conditions so that the specifications are met, and no operations which may increase the reactivity of the core shall be performed.

*For the purpose of completing the fuel storage pool modification, the movement and placement of loads shall be in accordance with the installation procedures approved by the plant on-site review committee.

1877 095

5.3 REACTOR

A. Reactor Core

1. The reactor core contains approximately 48 metric tons of uranium in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods. The reactor core is made up of 121 fuel assemblies. Each fuel assembly contains 179 fuel rods.

2. The average enrichment of the reload core is a nominal 2.90 weight per cent of U-235. The highest Uranium-235 loading is a nominal 39 grams of U-235 per axial centimeter of fuel assembly (average).

3. In the reactor core, there are 29 full-length RCC assemblies that contain a 142-inch length of silver-indium-cadmium alloy clad with stainless steel.[2]

B. Reactor Coolant System

1. The design of the reactor coolant system complies with all applicable code requirements.[3]

2. All high pressure piping, components of the reactor coolant system and their supporting structures are designed to Class I requirements, and have been designed to withstand:

   a. The design seismic ground acceleration, 0.06g, acting in the horizontal and 0.04g acting in the vertical planes simultaneously, with stressed maintained within code allowable working stresses.

   b. The maximum potential seismic ground acceleration, 0.12g, acting in the horizontal and 0.08g acting in the vertical planes simultaneously with no loss of function.

3. The nominal liquid volume of the reactor coolant system, at rated operating conditions, is 6100 cubic feet.

C. Protection Systems

The protection systems for the reactor and engineered safety features are designed to applicable codes, including IEEE-279, dated 1968. The design includes a reactor trip for a high negative rate of change of neutron flux as measured by the excore nuclear instruments.[4] The system is intended to trip the reactor upon abnormal dropping of more than one control rod[4]. If only one control rod is dropped, the core can be operated at full power for a short time, as permitted by Specification 3.10.

References

(1) FSAR, Section 3.2.3
(2) FSAR, Sections 3.2.1 and 3.2.3
(3) FSAR, Table 4.1-9
(4) FSAR, Section 7

5.6. FUEL HANDLING

A. Criticality Consideration

The new and spent fuel pit structures are designed to withstand the anticipated earthquake loadings as Class I (seismic) structures. The spent fuel pit has a stainless steel liner to ensure against loss of water. (1)

The new and spent fuel storage racks are designed so that it is impossible to insert assemblies in other than the prescribed locations. The fuel is stored vertically in an array with the center-to-center distance between assemblies sufficient to assure $k_{eff} \leq 0.95$ even if unborated water were used to fill the pit. In addition, fuel in the storage pool shall have a U-235 loading of $\leq$ 39.0 grams of U-235 per axial centimeter of fuel assembly (average).

The spent fuel storage pit is filled with borated water at a concentration to match that used in the reactor cavity and refueling canal during refueling operations or whenever there is fuel in the pit.

B. Spent Fuel Storage

The spent fuel storage facility is a two-compartment pool that may contain up to 1582 storage locations for spent fuel assemblies. The pool is enclosed with a reinforced concrete building having 12- to 18-inch thick walls and roof (1)

The pool and pool enclosure are Class I (seismic) structures that afford protection against loss of integrity from postulated tornado missiles. The storage compartments and the fuel transfer canal are connected by fuel transfer slots that can be closed off with pneumatically sealed gates. The bottoms of the slots are above the tops of the active fuel in the fuel assemblies which will be stored vertically in specially constructed racks.

EXHIBIT C

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

License Amendment Request dated January 31, 1980

Docket Nos. 50-282
50-306

License Nos. DPR-42
DPR-60

Exhibit C consists of the Nuclear Services Corporation document:

QUAD-1-79-509

"Licensing Report for Prairie Island Nuclear Generating Plant Units 1 and 2 Spent Fuel Storage Modification"

QUAD-1-79-509
JOB NO. NOR-0174

LICENSING REPORT

FOR

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

UNITS 1 AND 2

SPENT FUEL STORAGE MODIFICATION

NSP PROJECT NUMBER E-78Y075

Prepared For:

NORTHERN STATES POWER COMPANY
Minneapolis, Minnesota

By:

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REV. NO.	ENGINEER	REVIEWER	PROJECT QA ENGINEER	PROJECT ENGINEER	ISSUING MANAGER	DATE OF APPROVAL

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1.0 INTRODUCTION

This report is occassioned by a proposed modification to the Spent Fuel Storage System for the Prairie Island Nuclear Generating Plant Units 1 and 2. The proposed modification will provide 1582 spent fuel storage spaces for Unit 1 and Unit 2 combined. Of this 1582 storage spaces, 1386 spaces will be available for normal storage of fuel and 121 can be used for a full core discharge. The remaining 75 spaces will be used only during installation of the new racks. This layout is described more completely in Section 3.1.

The two units of the Prairie Island Plant share two interconnected fuel pools for spent fuel storage. The proposed modification applies to both of those fuel pools.

2.0 PURPOSE

The purpose of this report is to examine from a technical standpoint those aspects of the proposed modification which may have a bearing on the storage of spent fuel in the Prairie Island spent fuel pools.

The nuclear, structural and thermal aspects of the proposed modification have been examined and those aspects of the modification are presented in the following sections of this report.

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3.0 SUMMARY OF DESIGN MODIFICATIONS AND ANALYSES

3.1 General Description

The Prairie Island Spent Fuel Storage Facility consists of two storage pools. The first is a small fuel storage pool (pool 1) which is used for fuel storage and for loading of fuel into the shipping cask. The other pool (pool 2) is a larger pool which is used only for fuel storage. The arrangement of these two pools is shown in Figure 3.1-1.

In order to use a spent fuel shipping cask in pool 1, it will be necessary to remove the four spent fuel racks located in the southeast corner of that pool. Therefore, only the five remaining racks in pool 1 can be used for normal fuel storage. This results in the availability of 266 normal storage spaces in pool 1. The racks in the southeast corner of pool 1 can be used for a full core discharge, since it is not necessary to use a shipping cask during a full core discharge.

The spent fuel pool structure and supports have been analyzed and found to be acceptable for the additional load imposed by the increased fuel storage capacity. A description of this analysis and the results are presented in Report Q-1-79-558.

3.2 Mechanical Design

Figure 3.1-1 shows that two sizes of spent fuel racks will be used; a 7 x 7 space rack and a 7 x 8 space rack. The 7 x 8 rack is shown in Figure 3.2-1. In this design, upper and lower grids are used to interconnect the storage tubes. These grids also ensure proper location of the storage tubes on 9.5 inch pitch in both directions. The upper and lower grids are tied together by vertical and diagonal members. These members will transmit seismic and handling loads from the tubes to the rack base.

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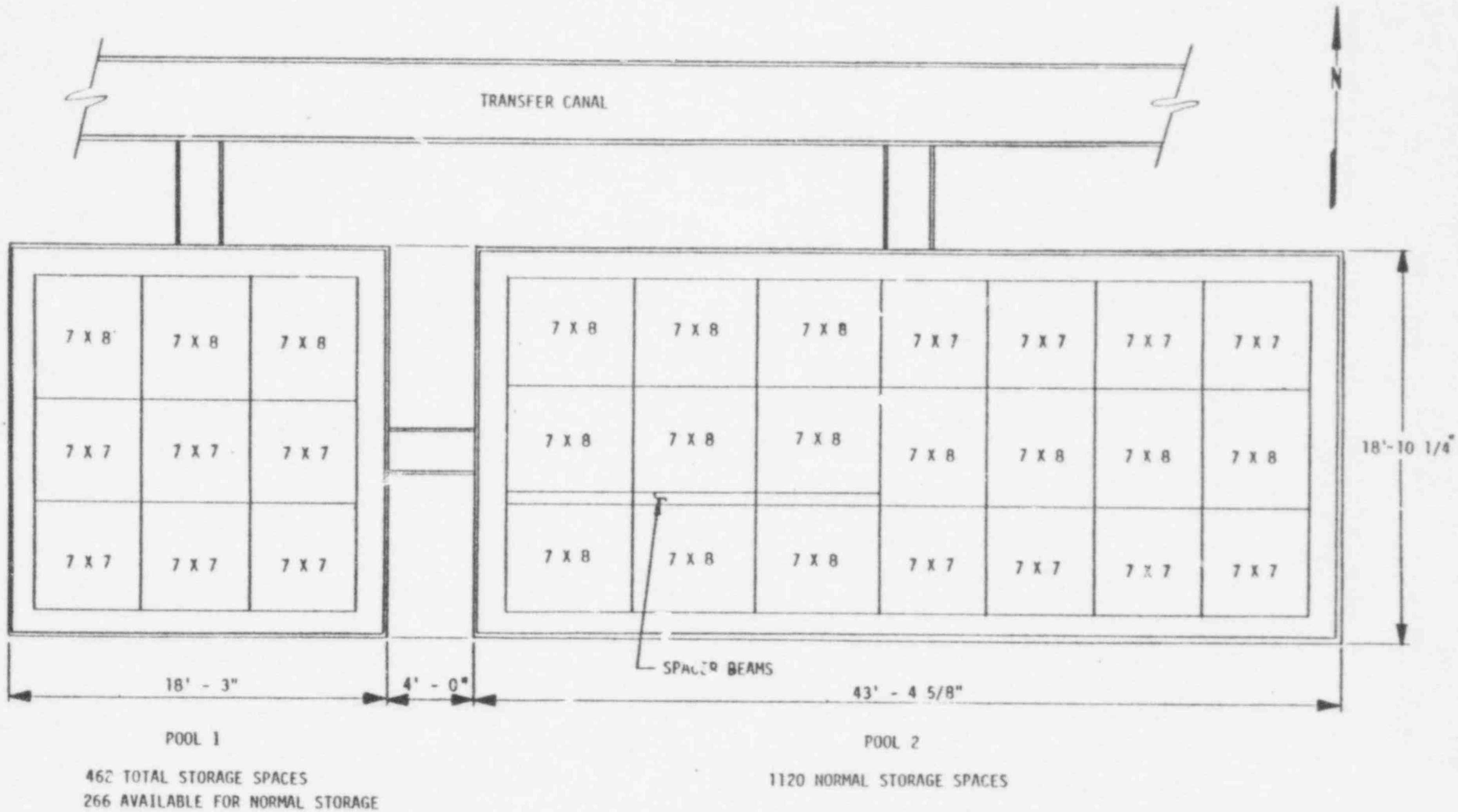

FIGURE 3.1-1: PRAIRIE ISLAND SPENT FUEL STORAGE FACILITY

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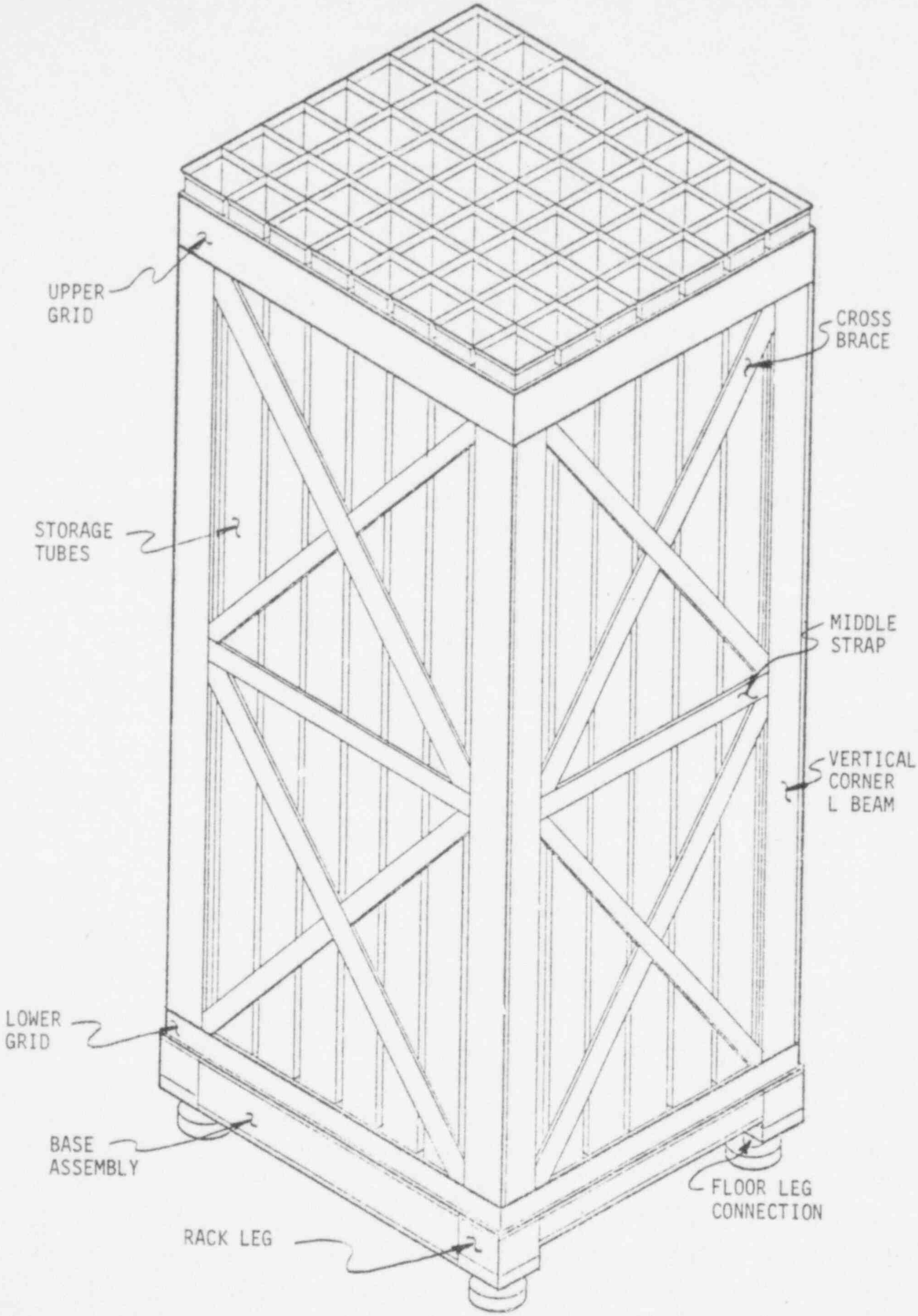

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FIGURE 3.2-1: 7X8 SPENT FUEL RACK

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The rack base is composed of heavy box beams connected at the four corners to box section legs with adjustable feet. These adjustable feet will provide adjustment during installation to ensure that the storage tubes are vertical. The box beams of the base are elevated above the pool floor to allow flow of cooling water below the rack and up into the storage tubes.

Reactivity control is provided by the 9.5 inch storage tube pitch and by the storage tube material. Each storage tube consists of three components: an inner stainless steel tube, a layer of neutron absorbing material, and an outer skin of stainless steel.

The inner tube is of adequate length to extend from below the bottom of the fuel assembly to above the top of a stored fuel assembly with the control cluster in place. Two support bars are welded into the bottom of this inner tube, and the stored fuel rests on these support bars.

The layer of neutron absorber is located on the four outer surfaces of each inner tube. The neutron absorber is in the form of solid sheets of material provided by Brand Industrial Services Company. This type of material has been previously licensed by the USNRC for use in spent fuel racks. The material is composed of a silicon polymer base material with sufficient boron in the form of boron carbide to result in an area density of 0.04 grams/square centimeter of boron-10. The neutron absorber extends the full axial length of the active fuel region. Quality assurance procedures for the neutron absorber fabrication will ensure to a 95% confidence level that the boron-10 area density is a minimum of 0.04 g/cm^2.

The outer skin is a thin sheet of stainless steel which covers the neutron absorber and holds the absorber in place. This outer skin will completely enclose the neutron absorber, except that some

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regions at the joint between of the outer skin and the inner tube will not be welded. This will result in a venting of the neutron absorbing region and will prevent pressure buildup in this region which might result from gas generation or hydrostatic pressure buildup.

With the exception of the above described neutron absorber and the screws in the ajustable feet, all material used for rack construction will be type 304 stainless steel which meets the appropriate ASTM or ASME material specification. The adjustable foot screws will be 17-4 Ph stainless steel in accordance with ASTM A564. These stainless steel materials are the same materials used in the fuel racks presently in use at Prairie Island.

Each rack sits on the pool floor liner. There are no bolted or welded connections between the rack and pool. Vertical loads are transmitted in bearing from the rack feet directly to the floor. Horizontal loads are transmitted from the feet to the floor in shear by friction only. There are no connections between adjacent racks, nor are there any supports to the fuel pool walls.

Spacer beams are located as shown in Figure 3.1-1 to preclude the insertion of a fuel assembly between racks.

3.3 Nuclear Analysis

A nuclear analysis has been performed for the proposed spent fuel rack to determine the k_{eff} for the rack while storing the fuel.

3.3.1 Design Criteria

The acceptance criteria established for this spent fuel storage rack is as follows:

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TABLE 3.3-1

Fuel Design Parameters

	Fuel Type		
	Westinghouse	Exxon	Future
Rod Array	14 x 14	14 x 14	14 x 14
No. of Fuel Rods	179	179	179
No. of Water Holes	17	17	17
Rod Pitch	0.556"	0.556"	0.556"
Pellet O.D.	0.3659"	0.3565"	0.3444"
Clad O.D.	0.422"	0.424"	0.400"
Clad Thickness	0.0243"	0.0300"	0.0243"
Clad Material	Zircaloy 4	Zircaloy 4	Zircaloy 4
Pellet Density, % T.D.	94	94	94
U_{235} Loading	39 g/cm	39 g/cm	39 g/cm
Nominal Active Fuel Length	144"	144"	144"

When immersed in clean, unborated water and completely filled with new fuel as described in Table 3.3-1, the calculated k_{eff} for this rack shall be ≤ 0.95. This shall include all probable uncertainties and anticipated pool water temperatures.

This spent fuel rack is designed in accordance with USNRC Standard Review Plan 9.1.2. Revision 2.

Included in the conditions to be analyzed are: pool water temperatures from 40°F to 212°F, eccentric position of fuel in the proper location, fuel tube and rack assembly tolerances, and single fuel assemblies outside the storage racks.

3.3.2 Analysis Method

The value of k_{eff} is determined as follows:

$$k_{eff} \leq k_0 + \Delta k_1 + \Delta k_2 + (\Delta k_3^2 + \Delta k_4^2 + \Delta k_5^2 + \Delta k_6^2 + \Delta k_7^2)^{1/2}$$

where

k_0 = nominal calculated k (2-D diffusion theory)
Δk_1 = transport correction
Δk_2 = method bias
Δk_3 = fuel location effect
Δk_4 = storage tube pitch tolerance effect
Δk_5 = uncertainty in methods bias (95% confidence level)
Δk_6 = neutron absorber boron-10 tolerance effect
Δk_7 = storage tube dimensional effect

Verification of k_{eff} is obtained using a two dimensional (X-Y) diffusion theory computer code calculation. Fuel neutron cross sections are developed for a four group energy range using the

CHEETAH code (Ref. 1) which is an adaptation of the LEOPARD-CINDER code. XSDRN (Ref. 2), which is a one dimensional discrete ordinates spectral averaging code, was used for two calculations:

1. Fuel rod cell, 123 group; collapsed to 27 groups.
2. Rack supercell, 1-D (cylindrical) approximation, to collapse from 27 to 4 groups.

Cross section sets for all non-fuel regions were obtained from XSDRN. The output of CHEETAH and XSDRN are used for a diffusion theory calculation using the CITATION code (Ref. 3) to establish the k_{eff}.

In order to verify the accuracy of the CHEETAH-XSDRN-CITATION calculation, a comparison was made with two critical experiments as shown in Table 3.3-2. The actual fuel used in the critical experiments was placed into CHEETAH-XSDRN-CITATION calculation with the results shown for comparison. These results may be compared with the measured k_{eff} and with the value of k_{eff} calculated using the KENO code (Ref. 4). As a further check the k_{eff} for the proposed fuel rack has been calculated with the KENO code. The data provided in Table 3.3-2 show that the CHEETAH-XSDRN-CITATION analysis performed for the proposed racks provides a conservative result for k_{eff}. The methods bias and uncertainty in the methods presented in the following section are based on the comparison to KENO for the proposed design.

Based on the values of k_{eff} calculated with CHEETAH-XSDRN-CITATION and KENO for the proposed design, the transport theory correction factor is conservatively taken as zero.

The following assumptions were used in the calculations:

TABLE 3.3-2

CASE	EXPERIMENTAL	CHEETAH-XSDRN-CITATION	KENO
(1) Single BORAL blade in array of 9x9 2.35 w/o UO_2 fuel	1.0018	1.0174	1.01680 ±.00550
(2) Bare U rods with depleted U block and BORAL sheet, water reflected	1.000	1.01321	1.01748 ±.00634
Prairie Island rack with Westinghouse fuel	---	0.89998	0.85377 ±.00624

(1) Critical experiment data from BNWL-1379

(2) Critical experiment data from YAEC-1090, Run 105

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a. Pool is filled with fuel at the highest enrichment stored in an infinite array.

b. The water in the fuel storage pool is clean and unborated.

c. The pool temperature is 40°F (4.4°C).

d. No credit is taken for the fuel assembly support structure.

e. Neutron absorption in fuel assembly grids is excluded.

f. Axial neutron leakage is included.

g. No credit is taken for U^{234} and U^{236} in the fuel.

h. The neutron absorber boron-10 area density is 0.04 g/cm^2.

The analysis is based on nominal stainless steel thicknesses, because the Boron-10 content is taken at the minimum 95% confidence level value and the Boron-10 is the dominant neutron absorber.

The above assumptions are considered as a conservative base for the calculations.

3.3.3 Analysis Results

The nominal value of k_{eff} and the applicable uncertainties are presented in Table 3.3-3 for each of the fuel assemblies to be stored.

The results of the k_{eff} calculations for potential storage and handling conditions are listed as follows:

		K_{eff}		
CONDITION	Fuel Type	Westinghouse	Exxon	Future
1. Normal positioning in the spent fuel array See Figure 3.3-1		.89998	.90210	.93414

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TABLE 3.3-3

Nuclear Analysis Results

Term	Fuel Type W	Fuel Type Exxon	Fuel Type Future	Method
k_0	0.89998	0.90210	0.93414	Calculated for cell
Δk_1	0	0	0	Comparison with KENO
Δk_2	< 0	< 0	< 0	KENO and critical experiments
Δk_3	< 0	< 0	< 0	Fuel moved to corner
Δk_4	0.00059	0.00059	0.00065	.060" displacement in X and Y
Δk_5	0	0	0	KENO and critical experiments 95% confidence level
Δk_6	0	0	0	Analysis is based on minimum B^{10} content with 95% confidence
Δk_7	.00808	.00812	.00801	Analysis based on worst case tube dimensions
k_{eff}	.90808	.91024	.94218	

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where:

$$k_{eff} = k_0 + \Delta k_1 + \Delta k_2 + (\Delta k_3^2 + \Delta k_4^2 + \Delta k_5^2 + \Delta k_6^2 + \Delta k_7^2)^{1/2}$$

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TABLE 3.3-3 (Continued)

NOTE: From the data presented in Table 3.3-2, it can be seen that the combination of methods bias and uncertainty results in a negative value. These correction factors are here conservatively assumed to equal zero.

2.	Eccentric positioning in the spent fuel storage array with fuel at corners of the channels See Figure 3.3.2	.89062	.89270	.92457
3.	Normal positioning in the spent fuel storage array with the channel offset 0.06" (.152 cm) in both X and Y See Figure 3.3-3	.90057	.90269	.93479
4.	One extra fuel assembly at side of rack See Figure 3.3-4	.90742	.90958	.94056

Sensitivity studies were performed to evaluate the influence on k_{eff} of storage tube pitch and pool water temperature.

The effect of change in pitch on k_{eff} was calculated, and the results are presented in Table 3.3-4.

The reference rack unit cell was analyzed at several temperatures in order to determine the temperature effect on k_{eff}. The fuel pellet, clad, moderator, and rack pitch were assumed to expand normally. The results are shown in Table 3.3-5. The numbers quoted in this report are the higher values for the lowest temperature (40°F).

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TABLE 3.3-4

Effect of Storage Tube Pitch Variation at 40°F

Pitch (inches)	k_{eff}		
	W	Exxon	Future
9.44 x 9.44	0.90837	0.91053	0.94257
9.50 x 9.50 (base case)	0.89998	0.90210	0.93414
9.56 x 9.56	0.89189	0.89397	0.92601

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TABLE 3.3-5

Effect of Fuel Pool Water Temperatures

Temperature (°F)	k_{eff}		
	W	Exxon	Future
40 (base case)	0.89998	0.90210	0.93414
130	0.89000	0.89207	0.92339
212	0.87772	0.87974	0.90896

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3.3.4 Accident Conditions

Potential accident conditions considered in the design of these spent fuel racks include: dropping of a fuel assembly on top of a rack or between a rack and the pool wall; effect of an earthquake on the relative position of fuel racks and fuel storage tubes; loss of all cooling systems. The drop of a fuel shipping cask and tornado effects are not concerns in this modification for the reasons described below.

The effect of a dropped fuel assembly in the most reactive position next to a rack is shown as condition 4 in Section 3.3.3.

The drop of a fuel assembly onto the rack was considered. The fuel rack is designed to prevent any plastic deformation in the fuel region for these loads. Therefore, the reactivity effect of these abnormal loading conditions is insignificant. Consideration was given to a fuel assembly lying on top of a rack. Due to the greater than 18 inch separation of the fuel assembly on top of the rack from active fuel in the rack, the fuel on top of the rack is isolated and k_{eff} is not affected.

The fuel racks have been designed as seismic Class 1 equipment and, as such, can withstand the plant SSE with no damage to the structure which maintains $K_{eff} < 0.95$. Although the racks may move during an earthquake, the dimensions of the fuel rack structure preclude reduction of the fuel storage spacing between adjacent racks to less than the nominal analyzed dimension.

As shown in Table 3.3-5 the fuel rack has been analyzed for the full potential water temperature range. It can be seen that K_{eff} actually decreases as temperature increases. Therefore, loss of the pool cooling systems will not result in $K_{eff} > 0.95$.

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The drop of a fuel shipping cask or other heavy equipment into pool 2 is not a concern because the building crane is physically restricted and cannot be moved over pool 2. During installation a temporary crane will be used to move fuel racks. However, all fuel will be stored in pool 1 while equipment is being lifted in pool 2.

For pool 1 a cover was designed and fabricated during the previous fuel storage modification. At any time that pool 1 is used for fuel storage, this cover will be installed following completion of fuel handling. The use of this cover has been reviewed under the application for the previous fuel storage modification. When fuel is stored in pool 1, the cover will be used and no fuel shipping cask will be inserted into pool 1 until further licensing action is completed.

Because of the above, no consideration in the nuclear analysis has been given to the drop of a fuel shipping cask into the pool.

The building in which the fuel pools are housed has been designed for the maximum anticipated site tornado. Therefore, no tornado effect on the fuel storage equipment is anticipated.

3.3.5 Conclusion

The proposed design meets the specified criteria of $k_{eff} \leq 0.95$ for all conditions included in the postulated accidents. Therefore, storage of the fuel described in Table 3.3-1 presents no safety problems with respect to the nuclear aspects of the design.

In order to ensure that the fuel rack will contain the analyzed amounts of neutron absorber, a fabrication quality assurance program will confirm and document that each sheet of absorber is in location. Inspection and analysis of the absorber material will confirm to a 95% confidence level that the boron-10 content is equal to or above the level assumed in this analysis.

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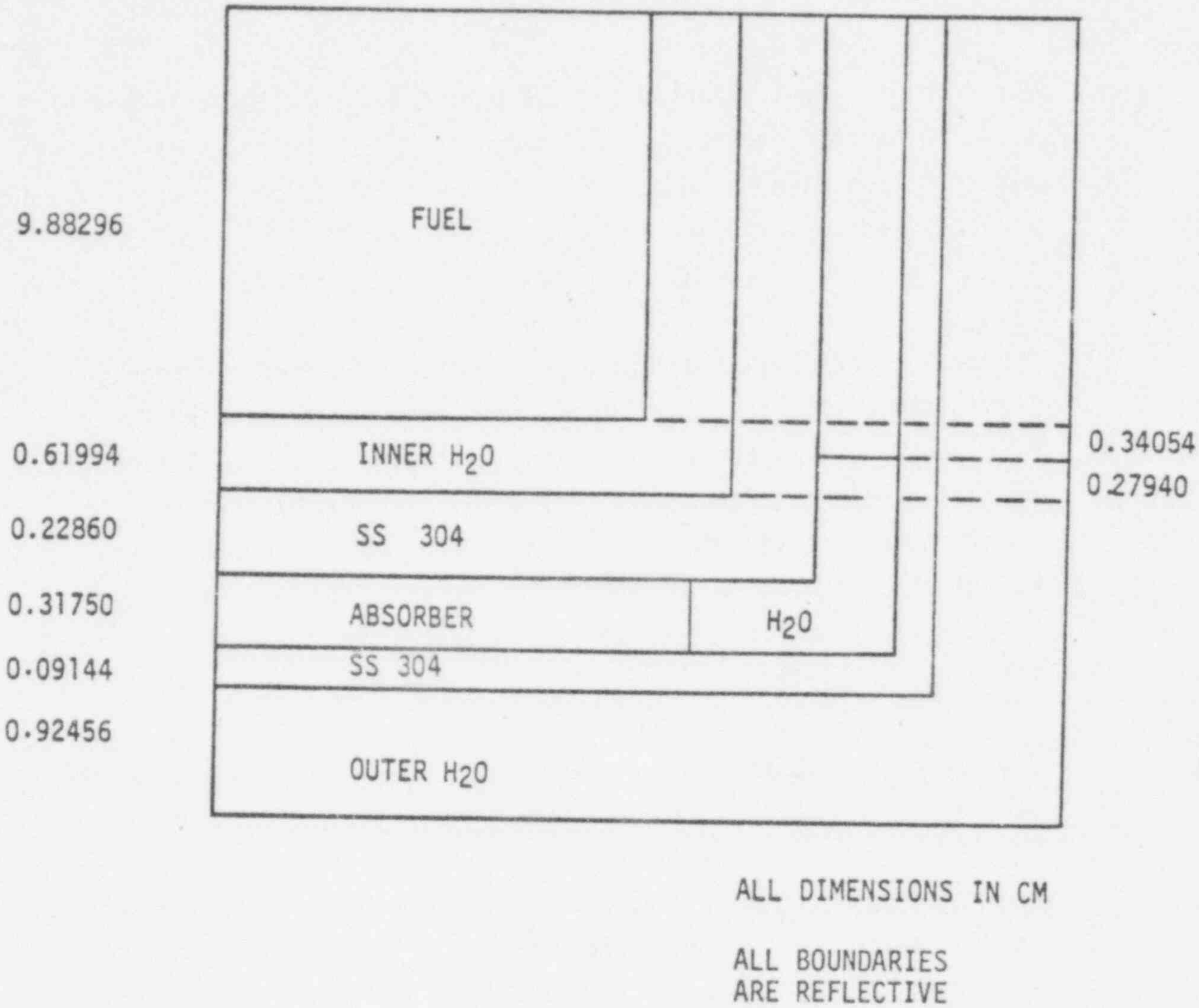
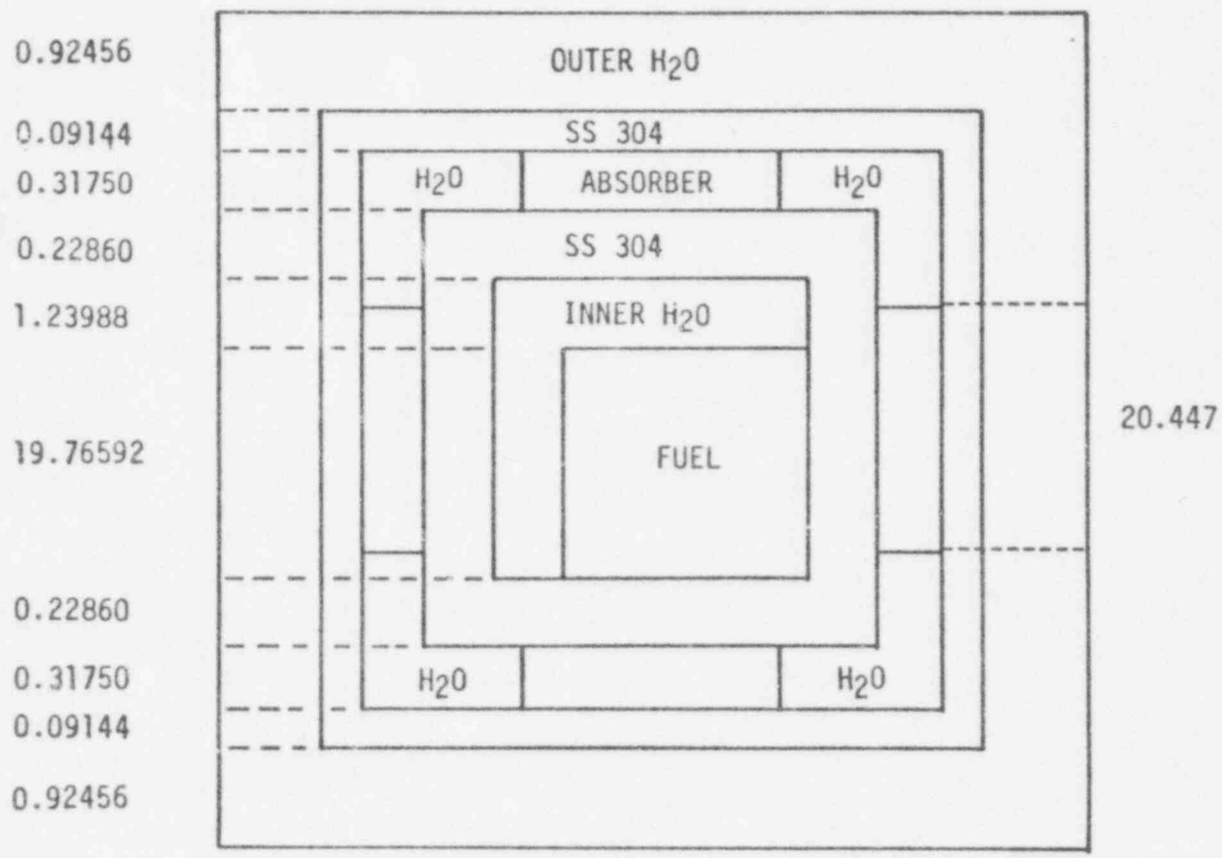


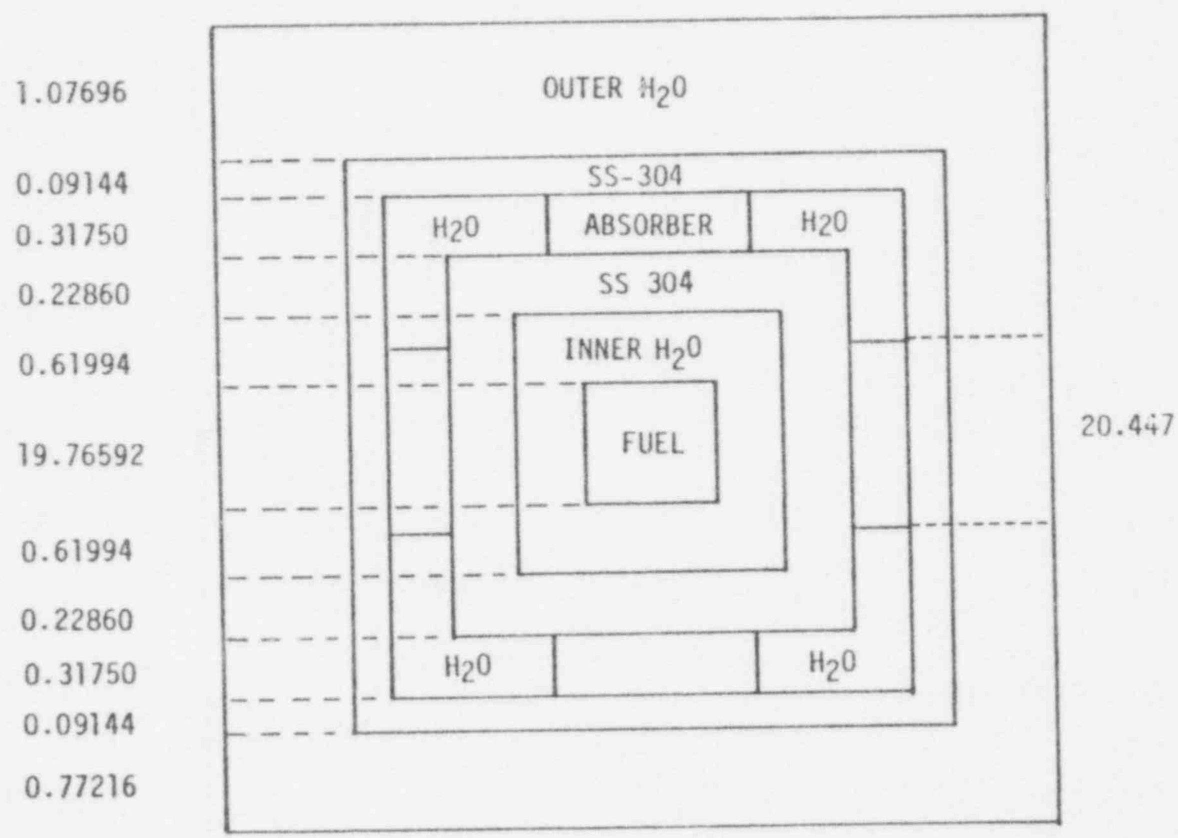
Figure 3.3-1: Normal Unit Cell Configuration



ALL BOUNDARIES ARE REFLECTIVE

ALL DIMENSIONS IN CM

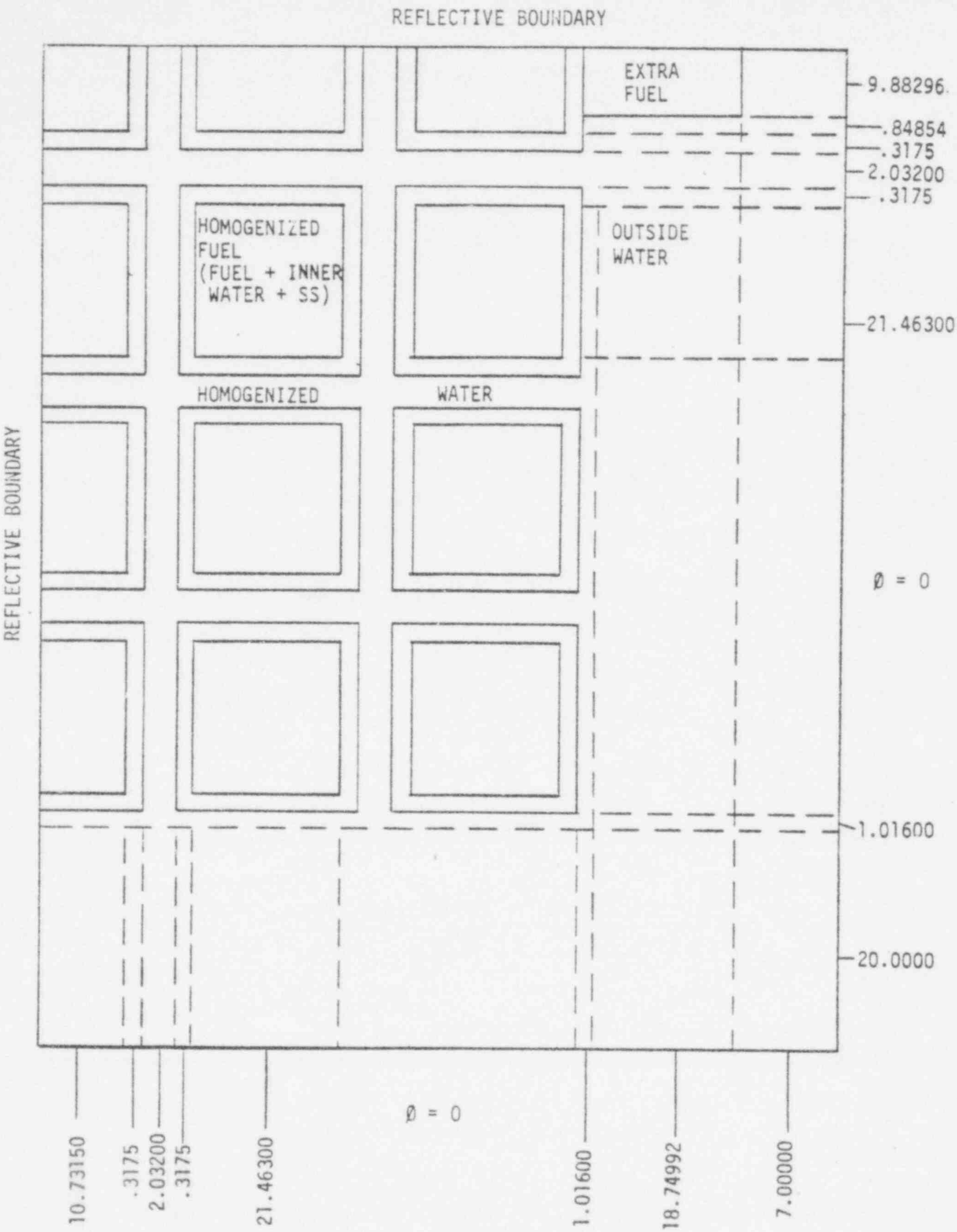
Figure 3.3-2: Fuel Offset



ALL BOUNDARIES ARE REFLECTIVE

ALL DIMENSIONS IN CM

FIGURE 3.3-3: STORAGE TUBE OFFSET



ALL DIMENSIONS IN CM

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FIGURE 3.3-4: ONE EXTRA FUEL ASSEMBLY AT SIDE OF RACK

As a further backup each storage tube will be inspected after installation in the pool to ensure that all neutron absorber sheets are in position.

3.4 Structural Analysis

3.4.1 Rack Structural Description

The proposed high-density spent fuel racks are free-standing type, i.e., these are placed on the pool floor without any floor attachment or lateral wall support.

Two different sizes of spent fuel racks have been used in the proposed pool arrangement. These are designated as 7 x 8 and 7 x 7 size racks. Configurations for these two rack sizes are similar. Each rack consists of a frame assembly, an upper grid, a lower grid, a base assembly, four leg assemblies, and the tubes supported on the lower grid and base assembly and welded to the top grid. The structural components of the rack assembly are shown in Figure 3.2-1 and listed in Table 3.4-1.

3.4.2 Loads, Load Combinations, and Evaluation Criteria

3.4.2.1 Loads

The following loads are considered in the design and evaluation of the proposed racks in their installed condition in the pool:

D = Deadweight: Weight of the rack assembly including the spent fuel bundles.

B = Buoyance: The effects of the buoyancy on the submerged racks and fuel bundles.

TABLE 3.4-1
MEMBER SIZES FOR RACK ASSEMBLY (1)

RACK COMPONENT	PLANE (2)	MEMBER SIZE
UPPER GRID	OUTER PLANE E&5	10"X3/4"
	INNER PLANE A&3	8"X1/2"
	INNER PLANE B,C,D,2&4	6"X1/2"
LOWER GRID	PLANE A,B,C,D,E AND 2,3,4,5	4"X1/2"
VERTICAL CORNER L BEAM	ALL 4 CORNERS	8"X8"X1/2"
CROSS BRACES	PLANE A&3	4"X1/2"
	OUTER PLANE E&5	4"X3/4"
MIDDLE STRAP	OUTER PLANE E&5	4"X3/4"
BASE ASSEMBLY CROSS (BOX) BEAMS	PLANE 5	10 1/2" X 4"
	PLANE A,C,E	6" X 6" X 3/4"
RACK LEG	ALL 4 CORNERS	9" X 9" X 3/4"
FLOOR LEG CONNECTION	ALL 4 CORNERS	2 3/4" Ø SOLID ROUND BAR

(1) See Figure 3.2-1; (2) See Figure 3.4-7

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E = Operating Basis Earthquake (OBE): Loads from the two horizontal and the vertical component of the OBE using pool floor OBE vibratory motion from Reference 14. The mass of the water inside the rack and hydrodynamic mass effects of the water surrounding the racks are also considered.

E' = Safe Shutdown Earthquake (SSE): SSE loads were conservatively assumed to be twice the OBE loads.

M = Fuel Bundle Drop: Loads resulting from the accidental drop of a spent fuel bundle from a height of 18 inches.

Q = Thermal Gradient: Loads resulting from thermal gradient due to a single "hot" spent fuel bundle being placed in a rack cavity with the adjacent cavities empty.

T_o, T_a = Pool Water Temperature: Loads resulting from the increase in fuel pool water temperature during normal operation (T_o) and during accident condition (T_a). However, for free-standing racks with no floor attachment, stresses due to T_o and T_a are insignificant and are not considered.

U = Grapple Load: Loads that might occur if a fuel assembly or a fuel handling grapple were to jam accidentally in the racks during removal. The design vertical and horizontal loads for this condition are considered to be 7,000 and 3,500 pounds, respectively. However, the resultant of the simultaneously applied forces is taken to be 7,000 pounds.

3.4.2.2 Load Combinations and Evaluation Criteria

In accordance with USNRC Regulatory Guide 1.29 (Reference 5) and ANSI Standard N210 (Reference 11), the proposed racks were classified as Seismic Category I and Safety Class 2 structures. Structural adequacy of the racks was verified for the applicable loading combinations and stress allowables listed in USNRC Standard Review Plan Section 3.8.4 (Reference 6). Basic allowable stress (S) values were taken from Table I-7.2 of ASME Boiler and Pressure Vessel Code Section III (Reference 7). Elastic working stress

method of analysis and AISC (Reference 17) method of evaluation were used for all loads except for impact loads, in which case an elasto-plastic method of analysis was used to evaluate the consequences.

Table 3.4-2 shows the loading combinations and the respective structural evaluation criteria for which the proposed spent fuel racks were to be evaluated. The basic acceptance criteria in the evaluation of the spent fuel rack is that the rack configuration shall always maintain the nuclear criticality coefficient K_{eff} less than 0.95. For the proposed racks this was ensured by using the stress limits set by USNRC in Standard Review Plan Section 3.8.4, and shown in Table 3.4-2.

Since the proposed racks are free-standing and are not tied to the pool floor, these may or may not slide during a seismic event and so may or may not impact on each other. This would depend on the seismic intensity to which the racks may be subjected, and the coefficient of friction between the rack and the pool floor. For evaluating the consequence of such impact as well as the consequence of accidental drop of a fuel bundle on the rack, the criticality criteria was translated to the following equivalent structural criteria: The resulting deformation state shall be such that the structure, which maintains the fuel spacing in the active fuel region, remains within the elastic limit.

Free-standing racks have the potential for overturning during severe seismic events. To ensure the stability of these racks, an acceptable factor of safety needs to be established as a minimum. Following USNRC Standard Review Plan 3.8.5 (Reference 12), a factor of safety of 1.5 for OBE and 1.1 for SSE has been used.

3.4.2.3 Seismic Input

Structural evaluation of the proposed racks was based on the seismic data provided in Reference 14 in which the spectra cor-

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TABLE 3.4-2

LOADS, LOAD COMBINATIONS AND STRUCTURAL ACCEPTANCE CRITERIA(4)

Load Combination(1)	Allowable Stress(2)
1. D + B	S
2. D + B + E(3)	S
3. D + B + Q	1.5S
4. D + B + Q + E	1.5S
5. D + B + E'(3)	1.6S
6. D + B + E' + Q	1.6S
7. D + B + U	(4)
8. D + B + M	(4)

Notes:

1. These are applicable loading combinations generally in accordance with USNRC Standard Review Plan 3.8.4.

2. These allowable stresses are in accordance with USNRC Standard Review Plan 3.8.4. The 'S' value for stainless steel shall be from the applicable appendix of ASME Boiler and Pressure Vessel Code, Section III, Division 1, 1977.

3. Factors of safety against overturning shall not be less than 1.5 for OBE and 1.1 for SSE (USNRC Standard Review Plan 3.8.5).

4. The general acceptance criterion for all the loading conditions is that the final configuration of the rack array shall maintain a $K_{eff} < 0.95$. This criticality criteria is translated to the following structural criteria: The resulting deformation state shall be such that the structure which maintains the fuel spacing in the active fuel region remains within elastic limit.

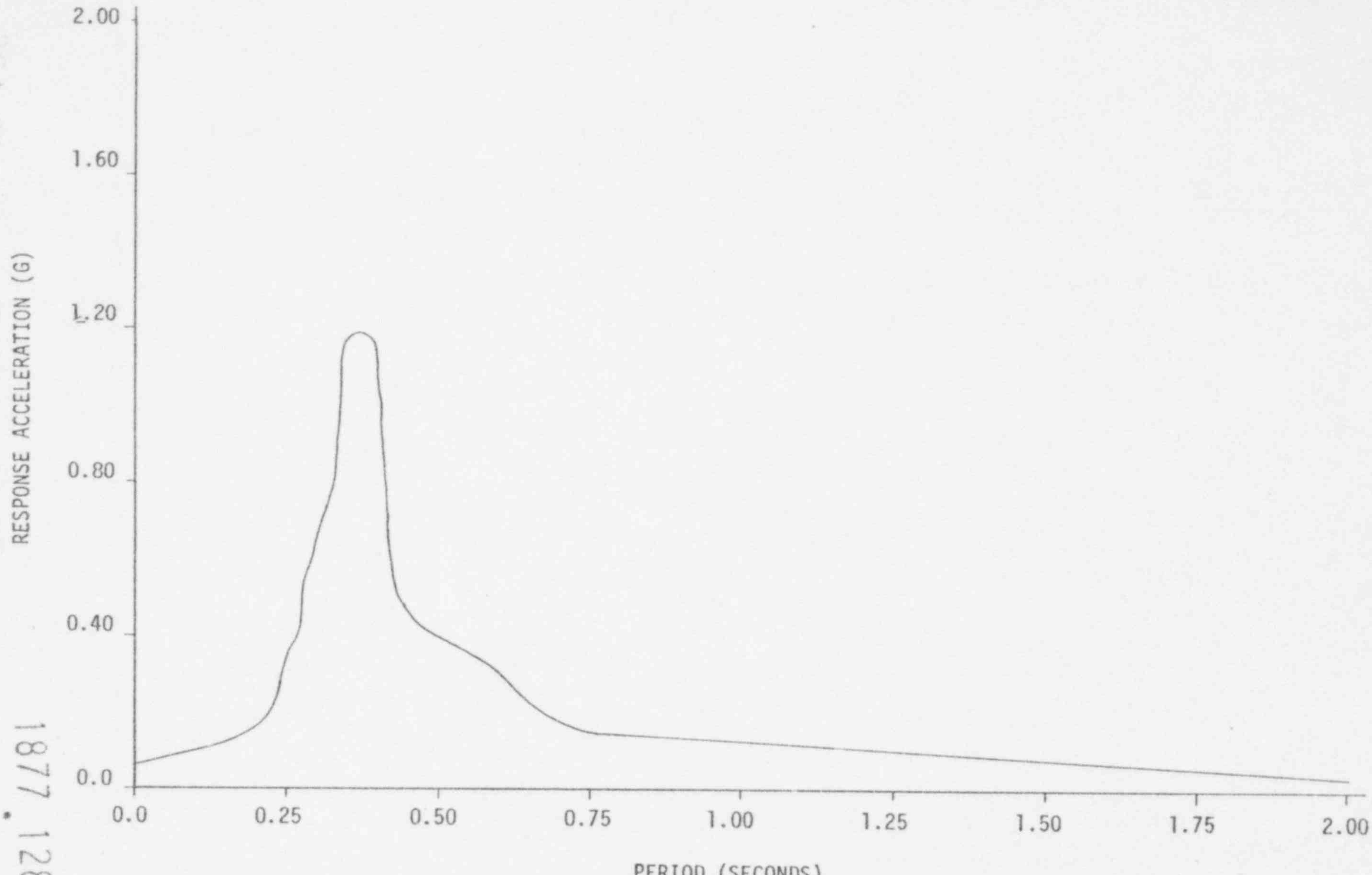

FIGURE 3.4-1: RESPONSE SPECTRUM OF OBE HORIZONTAL MOTION AT THE POOL FLOOR LEVEL
(DAMPING= 1% OF CRITICAL)

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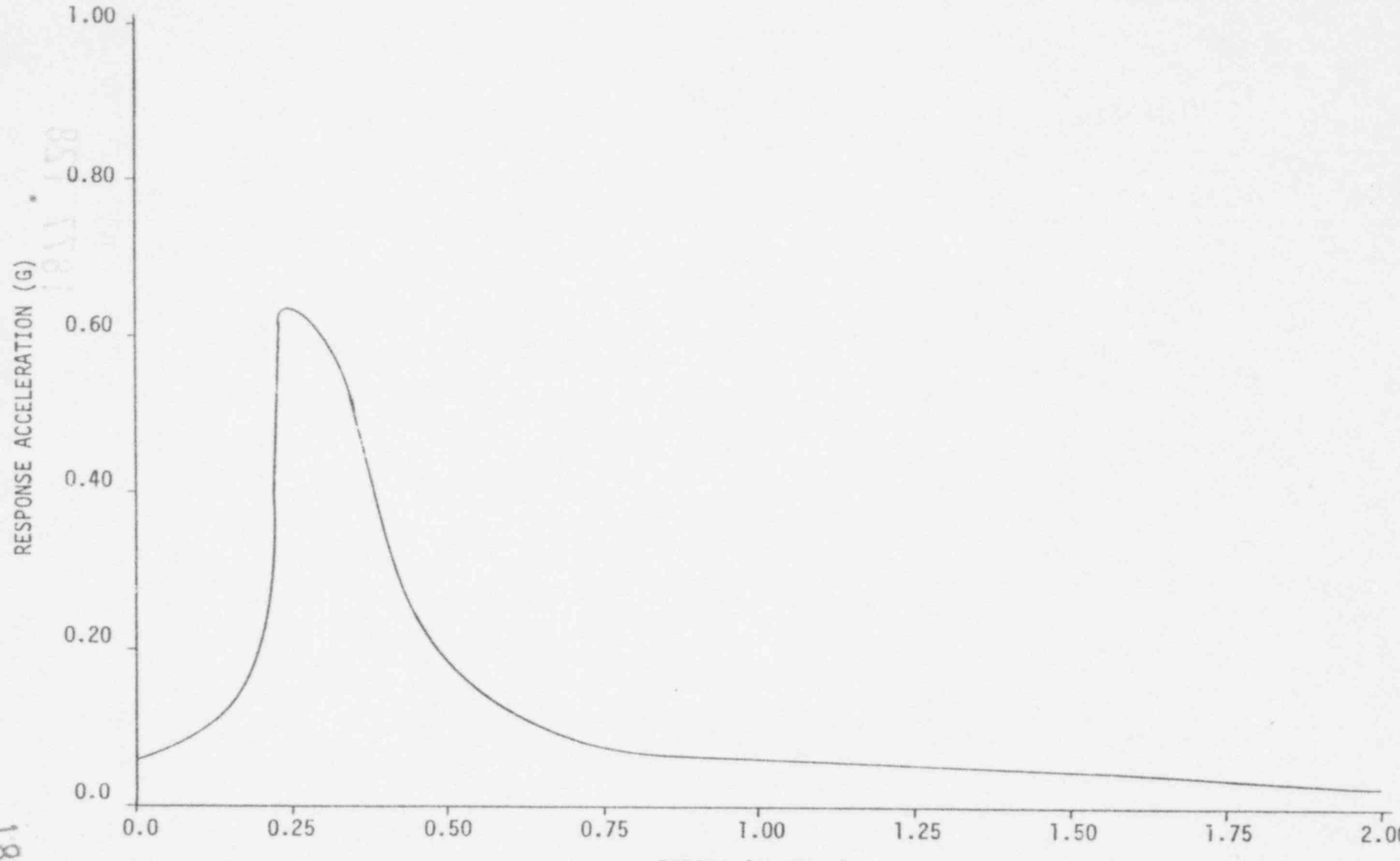

FIGURE 3.4-2: RESPONSE SPECTRUM OF OBE VERTICAL MOTION AT POOL FLOOR LEVEL
(DAMPING ≈ 1% OF CRITICAL)

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responding to Mass Point 25 represents those for the spent fuel pool floor. Figure 3.4-1 and 3.4-2 show the horizontal and vertical OBE response spectra at the pool floor. The spectrum values were multiplied by appropriate scaling factors to account for eccentricities (Reference 14). The final values representing the design OBE response spectra are listed in Tables 3.4-3 and 3.4-4.

For nonlinear sliding analysis, SSE time history of floor horizontal seismic motion was used as input. Such a time history was generated from SSE response spectrum using NSIC's proprietary computer program NSCTH (Reference 15). The SSE response spectrum for horizontal motion was developed by multiplying the OBE design response spectrum values shown in Table 3.4-3 by 2. This is shown in Figure 3.4-3. The generated acceleration time-history is shown in Figure 3.4-4. Figure 3.4-5 shows its compatibility to the target floor response spectrum in the frequency range of interest.

3.4.3 Method of Analysis

Four major types of analyses were performed on the proposed racks to evaluate their structural adequacy. Since the spent fuel racks would rest freely on the pool floor, it was necessary to determine the maximum horizontal movement and velocity of racks relative to the pool floor when subjected to the vibratory motion of the most severe postulated earthquake, i.e., SSE. This was computed by performing a nonlinear sliding analysis. To ensure that stresses in the spent fuel racks, when subjected to different combinations of loads, are within the allowable stress limits, elastic finite element stress analyses were performed for D, B, Q, U and E loadings. For a free-standing rack without any floor attachment, the stresses resulting from pool water temperature (T_o and T_a) are negligible. Hence, no analysis for T_o and T_a loadings was necessary. Since, SSE response spectrum input was not available; no separate analysis was performed for SSE loading condition. Conservatively, SSE

TABLE 3.4-3
RESPONSE SPECTRUM FOR OBE HORIZONTAL MOTION
(1% DAMPING)

Period (sec)	Frequency (cps)	Acceleration (g's)
.025	40.00	.073
.050	20.00	.083
.075	13.33	.094
.100	10.00	.104
.150	6.67	.130
.200	5.00	.190
.275	3.64	.595
.325	3.08	.822
.375	2.67	1.227
.425	2.35	.599
.600	1.67	.308
.800	1.25	.164
1.000	1.00	.140
1.500	.67	.092
2.000	.50	.043

TABLE 3.4-4
RESPONSE SPECTRUM FOR OBE VERTICAL MOTION
(1% DAMPING)

Period (sec)	Frequency (cps)	Acceleration (g's)
0.025	40.00	0.061
0.050	20.00	0.065
0.100	10.00	0.095
0.150	6.67	0.138
0.200	5.00	0.210
0.238	4.20	0.635
0.250	4.00	0.635
0.400	2.50	0.340
.500	2.00	.190
.600	1.67	.120
.800	1.25	.068
1.000	1.00	.060
1.500	.67	.042
2.000	.50	.025

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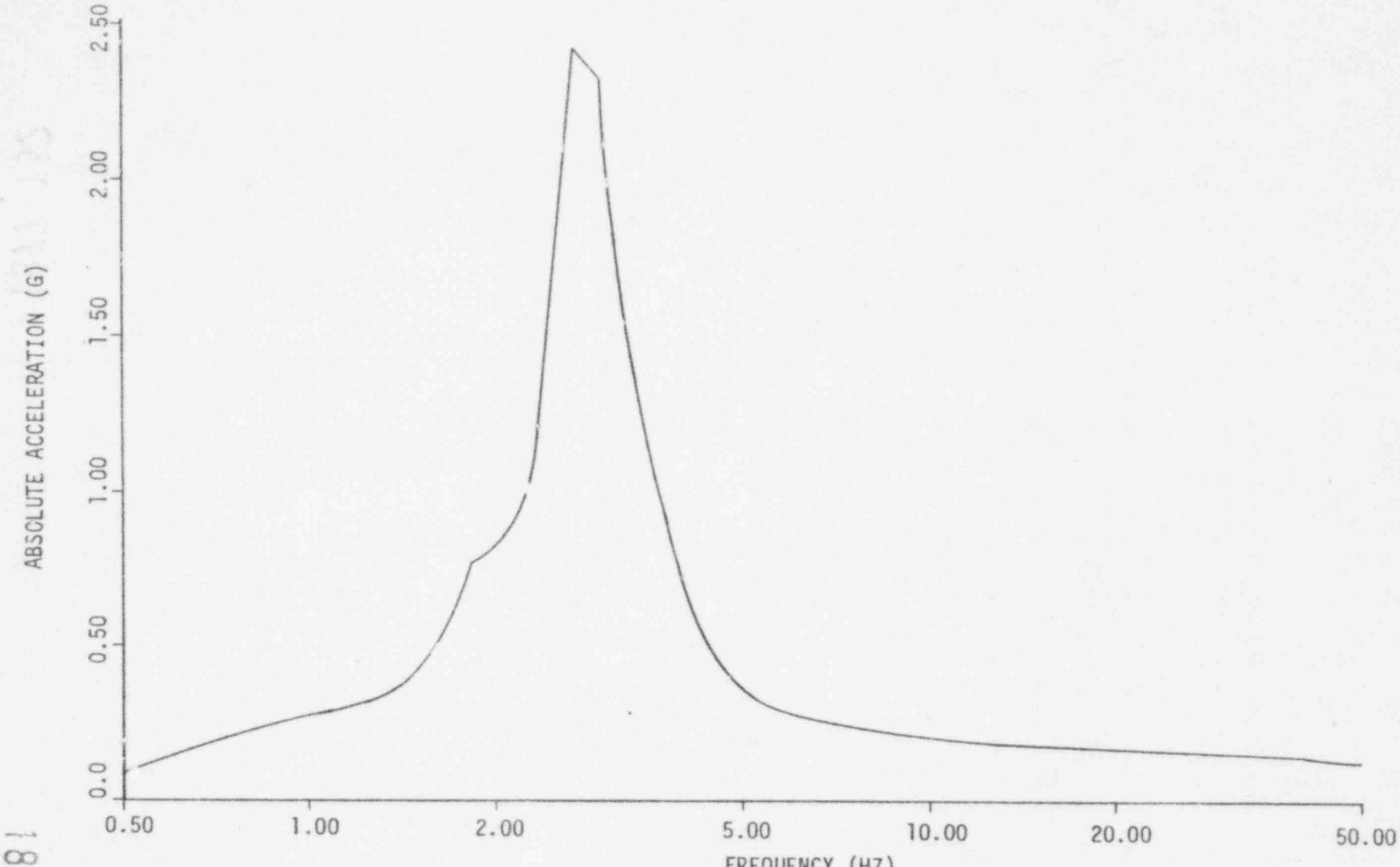

FIGURE 3.4-3: RESPONSE SPECTRUM FOR SSE HORIZONTAL MOTION AT POOL FLOOR LEVEL
(DAMPING = 1% OF CRITICAL)

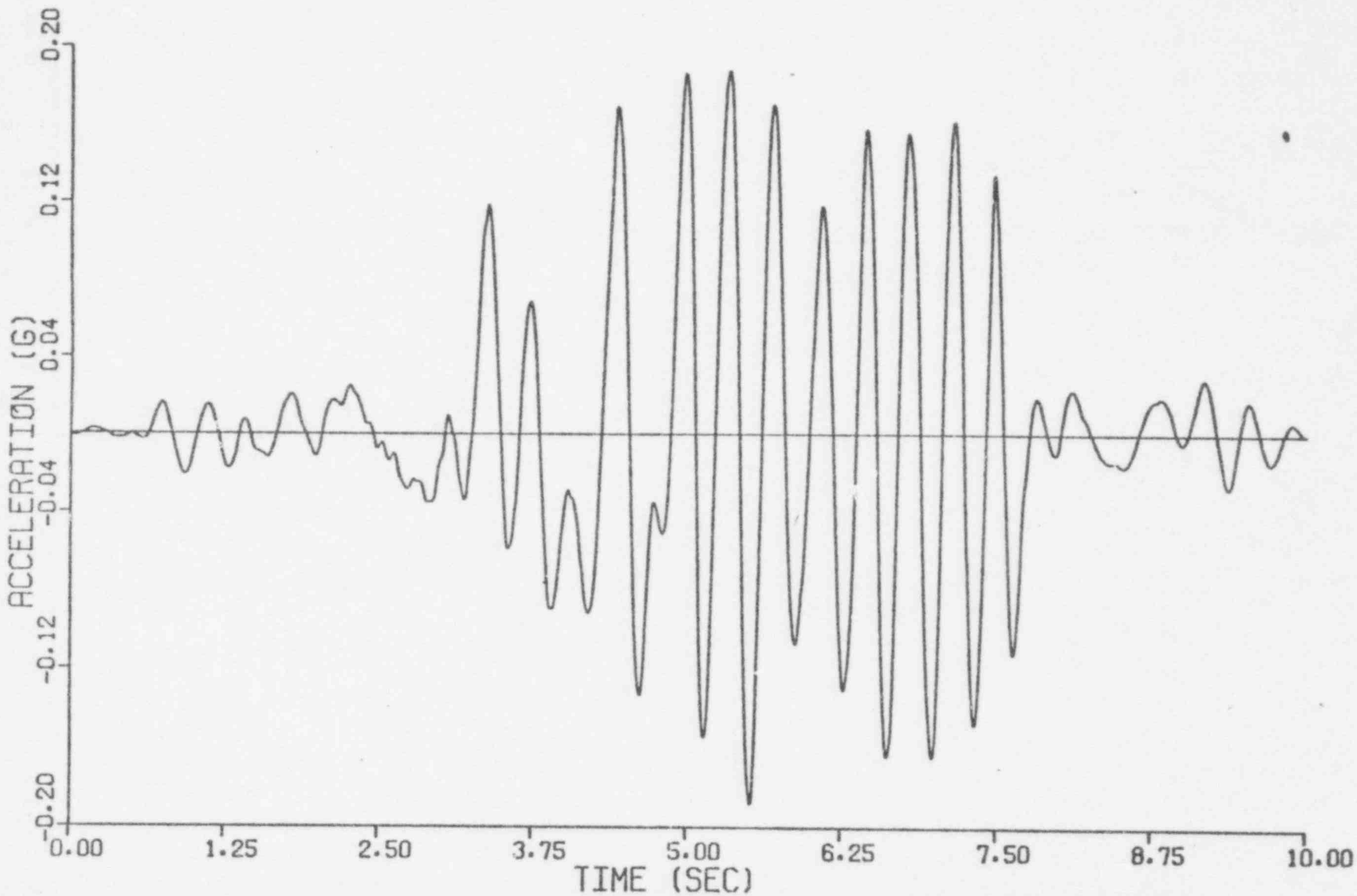

FIGURE 3.4-4: TIME HISTORY COMPATIBLE TO SSE HORIZONTAL RESPONSE SPECTRUM

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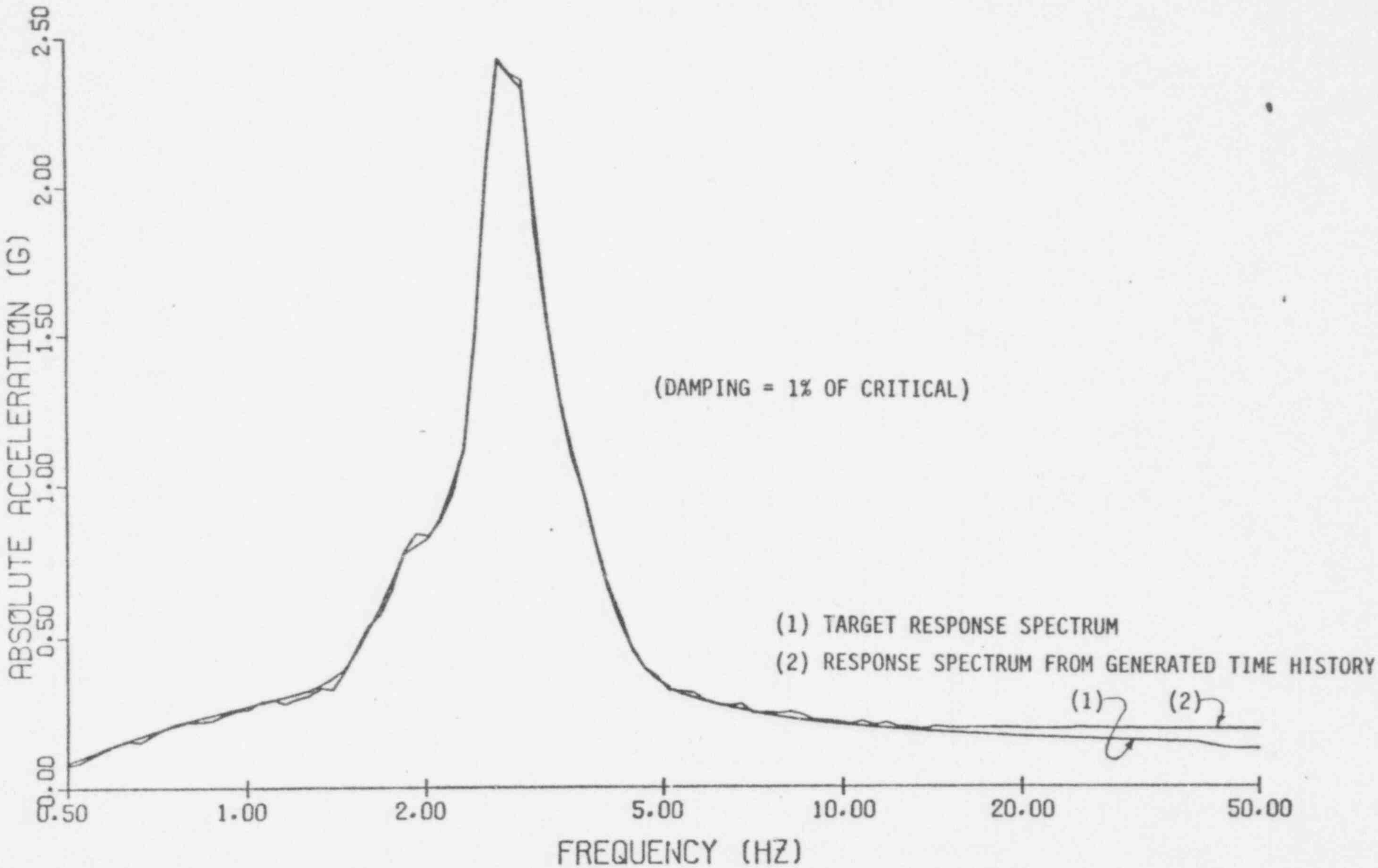

FIGURE 3.4-5: COMPARISON OF SSE HORIZONTAL RESPONSE SPECTRUM FROM GENERATED TIME HISTORY AND TARGET RESPONSE SPECTRUM.

stresses were taken as two times OBE stresses. Stability analyses of racks were performed to determine the factor of safety against overturning due to the action of SSE. Lastly, the consequences of accidental dropping of a fuel bundle (M) and the impact between two racks during an SSE event were evaluated.

The configurations of 7 x 8 and 7 x 7 racks are almost identical and their sizes are also not too different. Hence, it was judged that their structural behavior would be very much similar, and both rack sizes can be qualified structurally by analyzing only the critical rack size in detail. From past experience the 7 x 8 rack was judged to be the most critical (see Section 3.4.5). Section 3.4.4 describes the various structural analyses performed on 7 x 8 size racks. Evaluation of the 7 x 7 size rack has been performed approximately, by conservatively extrapolating the results of the 7 x 8 size rack. This is described in Section 3.4.5.

3.4.4 Analysis of 7 x 8 Rack

3.4.4.1 Sliding Analysis

The objective of this analysis was to determine the maximum displacement and velocity of the rack relative to the pool floor under the action of SSE vibratory motion. Upper bounds of these values were determined for a very conservative combination of various parameters affecting the movement. The conservative use of these parameters in the nonlinear sliding analysis is described below:

1. The coefficient of friction between the rack and the pool liner (i.e., between stainless steel and stainless steel in a wet condition) was assumed to be 0.2. Judging from the test results reported by Professor Rabinowicz of the Massachusetts Institute of Technology (Reference 8), this value is very

conservative. Results of his 199 tests under simulated condition had shown a mean value of 0.503, upper limit of 0.753 (mean plus two times the standard deviation), and a lower limit of 0.253 (mean minus two times the standard deviation).

2. The effect of vertical components of SSE was considered conservatively by assuming a constant upward acceleration on the rack equal to the peak vertical SSE acceleration. The frequency of the rack in the vertical direction was computed to be more than 21 cps for which the peak floor acceleration is 0.13g. However, a conservative value of 0.16g was applied to the rack in the upward direction thereby reducing the frictional resistance against sliding.

3. In computing the mass properties of the mathematical model, the water inside the rack was considered as added mass. The hydrodynamic mass effect of the surrounding water was considered conservatively by adding equivalent virtual mass computed in accordance with Reference 10. Due to the participation of larger hydrodynamic mass, inertial force and so the sliding displacement and velocity parallel to the shorter side of the 7 x 8 size rack would be more than those for the other rack size. Hence, sliding displacement and velocity parallel to the shorter side of the 7 x 8 rack were computed for the SSE motion. These values were used for computing the energy of impact between the racks.

4. For the sliding analysis, even though the hydrodynamic mass was considered in modeling the inertial properties of the rack, the associated damping was ignored, and only structural damping (2 percent) was used. For a structure vibrating in a water medium, the use of 2% damping is judged to be conservative.

Non-linear time-history analyses using the computer code ANSYS (Reference 9) were performed for the fully-loaded rack and for the empty rack. The mathematical model used in this analysis is shown in Figure 3.4-6. Equivalent stiffness properties of the members were obtained by matching the structural behavior of the stick model with that of a detailed finite element model of the actual rack (see Section 3.4.4.2). Input time-history is shown in Figure 3.4-4. A damping value equivalent to 2% modal damping was used in the analysis. Direct integration method of analysis was employed using time-step size of 0.01 second.

3.4.4.2 Dead Load, Buoyance, and Seismic Analyses

3.4.4.2.1 Mathematical Model

For the purpose of dead load (D), buoyance (B) and OBE seismic (E) analyses, the 7 x 8 rack was represented as an assemblage of finite elements. Figure 3.4-7 shows the finite element model used in the analysis. All the stress analyses on this model were performed using the computer program STARDYNE (Reference 16).

3.4.4.2.2 Stress Analysis for Deadload and Buoyance Loads

Stresses resulting from deadload and buoyance were evaluated simultaneously using the finite element mathematical model described in Section 3.4.4.2.1. These loads included the following:

1. Buoyant weight of the rack body.

2. Buoyant weight of the fuel bundles.

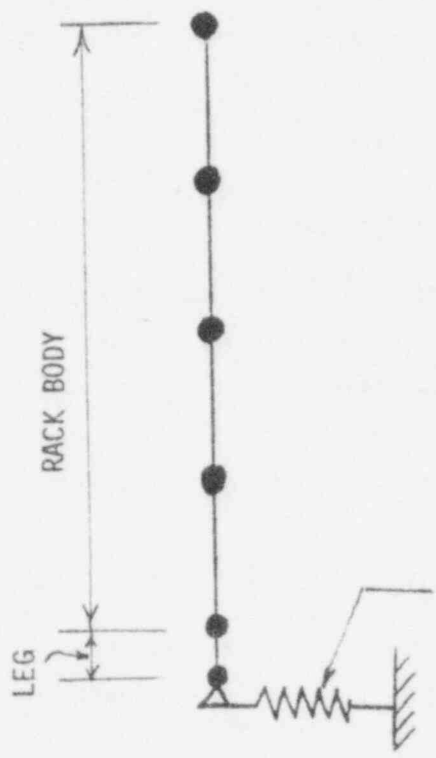

FIGURE 3.4-6: MATHEMATICAL MODEL FOR NONLINEAR SLIDING ANALYSIS

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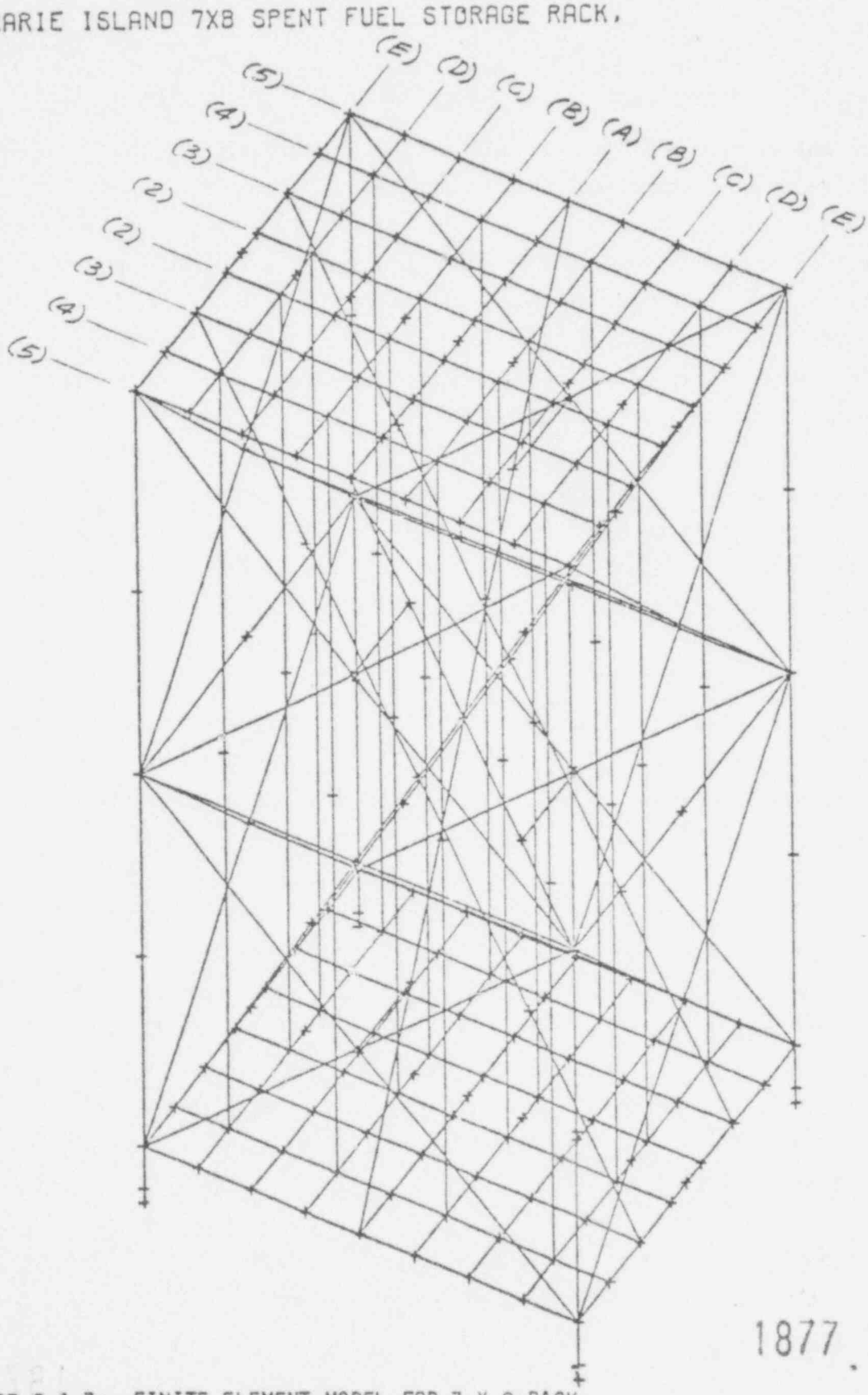

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FIGURE 3.4-7: FINITE ELEMENT MODEL FOR 7 X 8 RACK

3.4.4.2.3 Seismic Response and Stress Analysis

Modal analysis of the 7 x 8 loaded rack was performed using the finite element mathematical model described in Section 3.4.4.2.1.

Eighty-six dynamic modes of vibration were extracted. In performing these analyses, the mass of the water inside the rack was considered as added mass. The hydrodynamic mass effect of the water in between the racks was included in the dynamic mathematical model. This was computed in accordance with Reference 10.

Once the eigenvalues and the eigenvectors were extracted, the stresses due to three OBE components were computed by the response spectrum analysis method. For the horizontal and vertical components, OBE horizontal and vertical response spectra (Tables 3.4-3 and 3.4-4) were used.

In these analyses it was assumed that the friction coefficient is such that the rack would not slide and tilt. But during a postulated seismic event, the racks, having no floor attachment nor lateral supports, may or may not tilt depending on the level of seismic excitation. In the worst possible case, each rack may be supported on one or two legs for a very short duration in a dynamic state of equilibrium. Since the racks are free to slide, the probability for the rack to be tilting on one leg is judged to be very small. However, for the purpose of stress evaluation, the 7 x 8 rack was also analyzed for this extreme condition (i.e., when supported on one leg) in addition to the normal storage condition (i.e., when resting on four legs). For the convenience of reference, these two conditions will be termed as:

Configuration I: Rack supported on four legs
Configuration II: Rack supported on one leg

To simulate Configuration I, the lower nodes of the members representing the four corner legs were restrained from translation. Seismic re.ponse and stress analyses described in the previous paragraphs assumed rack Configuration I. Stress analysis performed using Configuratic.ı II is described in the next paragraph.

During a postulated seismic event, when the rack is supported on one leg for a very short duration, the equilibrium is maintained by inertia loads. For this condition, the external loads acting on the rack, which constitute dead load and seismic loadings were simulated by trial and error method to determine the condition which causes incipient tilting motion of the rack on one leg. The rack structure was then analyzed for the resulting loadings. In general, these loadings caused stresses higher than those obtained from Configuration I, and are considered as upper bound loadings.

3.4.4.3 Thermal Stress Analysis

Thermal stresses due to loads resulting from temperature gradient due to a single hot spent fuel bundle being placed in a rack cell with adjacent cavities empty (Q) are self-relieving, and so can be neglected in accordance with USNRC Standard Review Plan 3.8.4 (Reference 6). However, as a conservative approach these stresses were considered in the present analysis.

These stresses were evaluated for a temperature differential as shown in Figure 3.4-8 and using a the finite element model of the rack.

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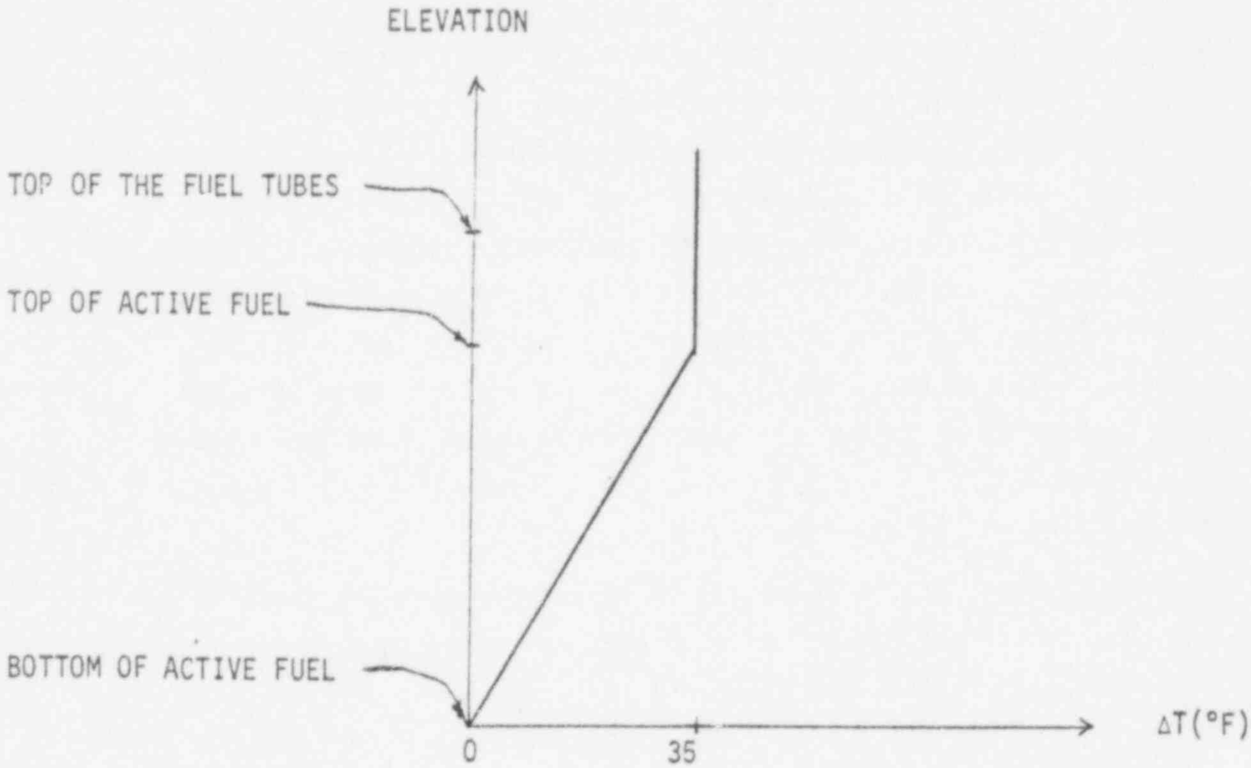

FIGURE 3.4-8 THERMAL GRADIENT

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3.4.4.4 Grapple Load Analysis

The forces that might occur if a fuel assembly were to jam in the racks during removal are conservatively estimated at: 7000 lbs. in the vertical direction and 3500 lbs. in the horizontal direction; however, the resultant of the simultaneously applied forces not to exceed 7000 pounds. Stresses due to these accidental U loads were computed by an elastic analysis.

3.4.4.5 Analysis of Impact Between Two Adjacent Racks

In the event of an SSE, the racks may potentially slide and impact on each other. In this analysis the potential damage to the rack due to this postulated impact is evaluated. The following acceptability criteria was used:

> Any structural part of the rack which is necessary to maintain $K_{eff} < 0.95$, should remain elastic. Thus, any local plastic deformation caused by the above postulated impact must be limited to that portion of the rack which is not required to maintain fuel assembly spacing.

The rack construction is such that the base assembly is projected outward from the rack storage tubes and frame. Thus, in the event of an impact, the two adjacent base assemblies will impact on each other. A portion of the impact energy may be absorbed by the local deformation of the base assembly. The remaining energy, if any, will be spent by a combination of the following:

a) Bottom of the racks will rebound moving away from each other.

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b) The tops of the racks may tip towards each other due to inertia, in which case, a part of the impact energy will be spent in tilting the rack.

c) If there is any more of the energy left, the racks may impact on each other at the top corners, in which case the remaining energy will be spent in deforming the top corners of the racks.

The above-mentioned phenomena were analyzed to determine the following:

a) The energy spent in deforming the base assembly structure.

b) The energy spent in tilting one rack to a position in which its top corner would just touch the top corner of the identically-tilted adjacent rack.

The energy of impact between two racks was assumed to be equal to the kinetic energy of the racks. The latter was computed using the peak sliding velocity of the rack determined from the nonlinear sliding analysis described in Section 3.4.4.1. The kinetic energy was computed conservatively using the mass value of the heaviest rack (i.e., 7 x 8 size rack).

The energy spent in tilting the rack to a position in which its top corner would just touch the top corner of the adjacent rack was computed as the work done in this process. In this computation it was conservatively assumed that the bottom base assembly of the adjacent racks are still in contact while the top of the racks are tilting toward each other. The energy required to tilt the rack was deducted from the initial impact energy to determine the energy

with which the top of the racks may impact each other. Thus, any energy absorbed in the elastic impact of the base assembly and, more significantly, in the rebounding of the bottom of the racks are neglected.

3.4.4.6 Analysis of the Impact Resulting From the Drop of a Fuel Bundle

This section describes the analysis performed to determine the consequences of an accidental drop of a fuel bundle onto the rack from a height of 18 inches above the top of the rack.

The objective of this analysis was to ensure that, in the accidental event of dropping a fuel bundle on the proposed rack at any location, the deformed configuration of the rack would still maintain the criticality coefficient $k_{eff} < 0.95$. This criticality criteria was translated to the following equivalent structural criteria: The resulting deformation state shall be such that the structure which maintains the fuel spacing in the active fuel region remains within the elastic limit.

Using energy balance methods, an elasto-plastic analysis of the rack was performed to determine the maximum length of the rack that might be stressed beyond elastic (yield) limit in the event of a postulated 18 inches drop of a fuel bundle at the most critical location on the rack. No credit was taken to account for the fluid energy dissipation during the drop phase, nor was any credit taken for the energy absorbed due to the crushing of the fuel bundle.

3.4.4.7 Overturning Analysis

To ensure that the proposed racks are laterally stable against overturning during an SSE, stability analysis has been performed very conservatively using an energy method which assumes the following phenomenon of sliding and overturning:

> At the instant the rack attains the maximum sliding velocity, it has the maximum kinetic energy. At this instant, it is assumed that all the kinetic energy of the rack is suddenly transformed to cause overturning of the rack. During this process when the rack is getting tilted, no sliding motion is assumed. Since the rack can potentially slide during the time required to tilt the rack to an unstable position, and thereby can dissipate a significant part of the energy in overcoming the friction force, this assumption is very conservative.

Other sources of conservatism in the stability analyses are as follows:

a) To compute the kinetic energy of the horizontal motion that contributes to the overturning potential, the sliding velocity of the rack is computed using the minimum coefficient of friction (see Section 3.4.4.1), which produced the largest velocity. On the other hand, this energy was assumed to be suddenly transformed to cause overturning, which is equivalent to an assumption that the translatory motion of the rack is stopped suddenly because of infinite friction coefficient. Thus, the overturning potential computed by this method is likely to be an upper bound value, i.e., the method is very conservative.

b) Viscous drag force of water during sliding and overturning has not been considered.

c) Effect of vertical earthquake is considered conservatively by assuming that a constant vertical acceleration equal to the vertical floor acceleration acts constantly upward, thereby increasing the overturning potential. Since, it is unlikely that the vertical component of the earthquake motion will peak at the same time as the horizontal motion, the above assumption is conservative.

Using the above-listed conservative assumptions, the factor of safety against overturning was computed as the ratio of the potential energy of the rack at the incipient overturning position (i.e., when its center of gravity is directly above the tip over edge) to the maximum kinetic energy of the sliding rack.

3.4.5 Analysis of 7 x 7 Rack

From past experience it has been observed that, of all the loads listed in Section 3.4.2.1, seismic loads are by far the most dominant loads. It has also been observed that the rack which has the least base width and highest ratio of plan length to plan width, has higher seismic stresses. From these two considerations, the 7 x 8 rack was considered to be more critical than the 7 x 7 rack. So, the 7 x 8 rack was analyzed in detail (Section 3.4.4). Stresses in the 7 x 7 rack are conservatively evaluated in this section extrapolating the stresses from the 8 x 7 rack, when necessary.

Nonlinear sliding analysis was performed for the 7 x 8 rack. The sliding movement is proportional to the horizontal inertia force resulting from the horizontal virtual mass which is equal to the actual mass plus the added hydrodynamic water mass. The force resisting the sliding movement is proportional to the actual mass only. The higher is the ratio of the virtual mass to the actual mass, the higher is the potential for sliding. Since, the 7 x 8

rack has higher plan length to plan width ratio, it has larger virtual mass to actual mass ratio, and so has higher sliding and overturning potential. Thus, the sliding and overturning potentials calculated for the 7 x 8 racks are upper bound values for the 7 x 7 racks. Also, since the mass of the 7 x 8 rack was used to compute the energy of impact between two racks, the factor of safety against having unacceptable damage potential during SSE event computed in Section 3.4.4.5 is also applicable to 7 x 7 size racks.

Stresses due to dead load plus buoyancy loads in 7 x 7 rack will be less than those in 7 x 8 rack since 7 x 8 rack is bigger and heavier.

Stress analyses due to Q and U loads (Sections 3.4.4.3 and 3.4.4.4) and the analysis of the impact resulting from the drop of a fuel bundle (Section 3.4.4.6) were performed on local models which are applicable to both 7 x 8 and 7 x 7 racks.

Thus, it is concluded that the stresses computed for the 7 x 8 rack are upper bound values for the stresses in 7 x 7 racks.

3.4.6 Analysis Results and Design Evaluations

Results of the analyses described in the previous sections showed that the proposed spent fuel racks, when subjected to various possible load combinations shown in Table 3.4-2, would meet the indicated structural acceptance criteria, and would have adequate factors of safety against any deformation state for which K_{eff} can be greater than the permissible value of 0.95.

Results of the upper bound sliding analysis for the loaded rack showed that the maximum sliding was 0.47 inch with a maximum sliding velocity of 4.43 inches/sec. For impact and overturning evaluation, the peak sliding velocity value was conservatively

rounded to 4.5 in/sec. For the empty rack the maximum sliding was 1.30 inch with a maximum sliding velocity of 5.6 inches/sec. Although the sliding velocity is higher for the empty rack, the total kinetic energy of the loaded rack is greater due to the added mass of the fuel. Therefore, the impact analysis was based on the loaded rack.

Results of the eigenvalue analysis for the 7 x 8 rack are summarized in Table 3.4-5.

Results of stress analyses are summarized in Tables 3.4-6 and 3.4-7. The values presented were calculated for the 7 x 8 rack and are upper bound values for the 7 x 7 rack.

Comparison of the computed stresses for various load combinations with the corresponding allowable stresses shows that the stresses are well within the allowable limits.

Results of stability analysis of the fully-loaded and the empty racks, when subjected to SSE motion, are summarized in Table 3.4-8. The resulting factors of safety against possible overturning are more than the minimum required value of 1.1. Since the minimum required factor of safety against overturning during OBE event is 1.5, while the computed minimum factor of safety for SSE loading is 25, no overturning analysis was needed for OBE loads.

Elasto-plastic analysis of the rack for the accidental fuel bundle drop showed that the maximum length of the rack which might be stressed beyond the elastic limit is 3.17 inches, whereas the available length of the racks above the active fuel length is about 17.4 inches.

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TABLE 3.4-5

RESULTS OF FREQUENCY ANALYSIS OF 7X8 RACK

Mode(1) No.	Frequency (cps)	Modal Participation Factors		
		Horizontal		Vertical
		Short Dir.	Long Dir.	
1	4.712	2.115	**	0.06
2	5.738	0.01	6.64	0.22
3	5.829	0.03	5.74	0.23
4	6.789	1.23	0.25	**
5	8.234	0.35	1.21	0.01
6	10.252	1.39	**	**
7	10.253	1.28	**	**
8	10.509	**	1.14	**
9	10.510	**	1.08	**
10	10.527	**	1.07	**
11	21.519	**	**	1.44
12	33.292	0.09	0.31	1.23
13	39.781	**	0.03	1.32

(1) Only the significant modes are listed

** Modal participation factor less than one-thousandth

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TABLE 3.4-6

STRESS EVALUATION FOR CRITICAL ELEMENTS
OF THE SPENT FUEL RACK

CONFIGURATION I

Rack (1) Component	Critical Load Combination	Computed Stress (2) Index	Allowable Stress Index
Upper Grid	D+B+Q+E'	.717	1.000
Lower Grid	D+B+E	.555	1.000
Vertical Corner L Beam	D+B+Q+E'	.197	1.000
Cross Brace	D+B+Q+E'	.399	1.000
Middle Strap	D+B+E'	.609	1.000
Absorber Tube	D+B+U	.974	1.000
Base Assembly Beam	D+B+E	.434	1.000
Rack Leg	D+B+Q+E'	.280	1.000
Floor Leg Connection	D+B+Q+E'	.348	1.000

NOTE: (1) See Figure 3.2-1

(2) Stress Index = $\frac{f_a}{F_a} + \frac{f_{bx2}}{F_{bx2}} + \frac{f_{bx3}}{F_{bx3}}$

Where,

f_a = computed max. axial stress

f_{bx2} = computed max. bending stress about the major axis.

f_{bx3} = computed max. bending stress about the minor axis.

F_a = allowable compressive stress

F_{bx2} = allowable bending stress about the major axis.

F_{bx3} = allowable bending stress about the minor axis.

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TABLE 3.4-7

STRESS EVALUATION FOR CRITICAL ELEMENTS OF THE SPENT FUEL RACK

CONFIGURATION II

Rack Component (1)	Computed Stress (2) Index	Allowable Stress Index
Upper Grid	.979	1.000
Lower Grid	.765	1.000
Vertical Corner L Beam	.678	1.000
Cross Brace	.561	1.000
Middle Strap	.326	1.000
Absorber Tube	.161	1.000
Base Assembly Beam	.966	1.000
Rack Leg	.502	1.000
Floor Leg Connection	.834	1.000

Note: (1) See Figure 3.2-1
(2) See Table 3.4-5 footnotes for explanation of symbols.

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TABLE 3.4-8
RESULTS OF OVERTURNING ANALYSES

Parameters Computed (1)	Fully Loaded Rack	Empty Rack
KE (Kip-inch)	4.64	3.54
PE (Kip-inch)	390.85	87.59
ΔV (inch)	5.76	5.76
F.S	84	25

(1) KE = Maximum Kinetic Energy of the Sliding Rack Due to SSE Motion

PE = Potential Energy when the Rack Mass is Lifted Through ΔV

ΔV = Vertical Displacement of the Center of Gravity of the Rack necessary to change the rack equilibrium from a stable state to critical one

F.S = Factor of Safety against Overturning

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The possibility of two adjacent racks impacting on each other was evaluated using the method described in Section 3.4.4.5. Results of the analysis show that:

a) When two adjacent racks impact on each other at the maximum sliding velocity of 4.5 inches/sec the postulated plastic deformation of the base assembly is very localized and insignificant.

b) Even if the energy absorbed in deforming the base assembly is ignored, the kinetic energy due to seismic motion is less than the potential energy required to tilt the racks to a position where the tops of the racks can impact on each other.

These, together with the fact that the available length of the racks above the active fuel length is about 17.4 inches, show that the racks are safe against postulated impact between each other during seismic events.

3.5 Thermal-Hydraulic Analysis

The decay heat generated by the spent fuel stored in the proposed racks is disipated into the fuel pool water. This water is then cooled by the spent fuel pool cooling system. In order to ensure adequate cooling of the fuel and of the fuel pool water two thermal analyses have been performed. The first considers cooling of an individual fuel assembly by natural circulation of water through the fuel storage tubes. The second considers overall cooling of the spent fuel pool by the Spent Fuel Pool Cooling System (SFPCS).

3.5.1 Fuel Assembly Cooling Analysis

3.5.1.1 Analysis Method

This analysis is performed to ensure that each fuel assembly receives adequate cooling. All cooling of the stored fuel is assumed to be the result of natural circulation caused by the heat generation of the fuel in the storage racks.

Nuclear Services Corporation has developed the computer code CIRCUS for analysis of natural circulation in spent fuel assemblies. The code treats a series of fuel assemblies fed by a single downcomer, through a series of inlet areas. Viscous losses through the downcomer, inlet and fuel channels are balanced with buoyant forces developed by power generation in the fuel channels. The upper pool is assumed to be maintained at a constant temperature. Outlet temperatures are computed based on an energy balance. An iterative procedure is used to balance the forces.

Natural circulation cooling in the spent fuel pool was modeled as shown in Figure 3.5-1. A peak power fuel assembly was assumed to be stored in the center of the pool (position 11) at the end of a row of average power fuel assemblies (fuel assemblies whose decay heat is based on the average fuel assembly power in the core). Flow to this row of fuel assemblies was assumed to follow a path which takes the cooling water from the upper pool, down the side of the pool (between the fuel racks and the pool wall) and under the fuel storage racks. This model gives an upper bound for outlet temperatures, since flow from other directions is neglected.

The flow of water into the corner storage tubes of each rack is more restricted than to the other tubes, since water must flow through the leg assemblies to reach the corner tubes. Because of this restriction, a check was made for the above described model with the peak power fuel assembly stored in position 8 and average power assemblies in the ten other positions.

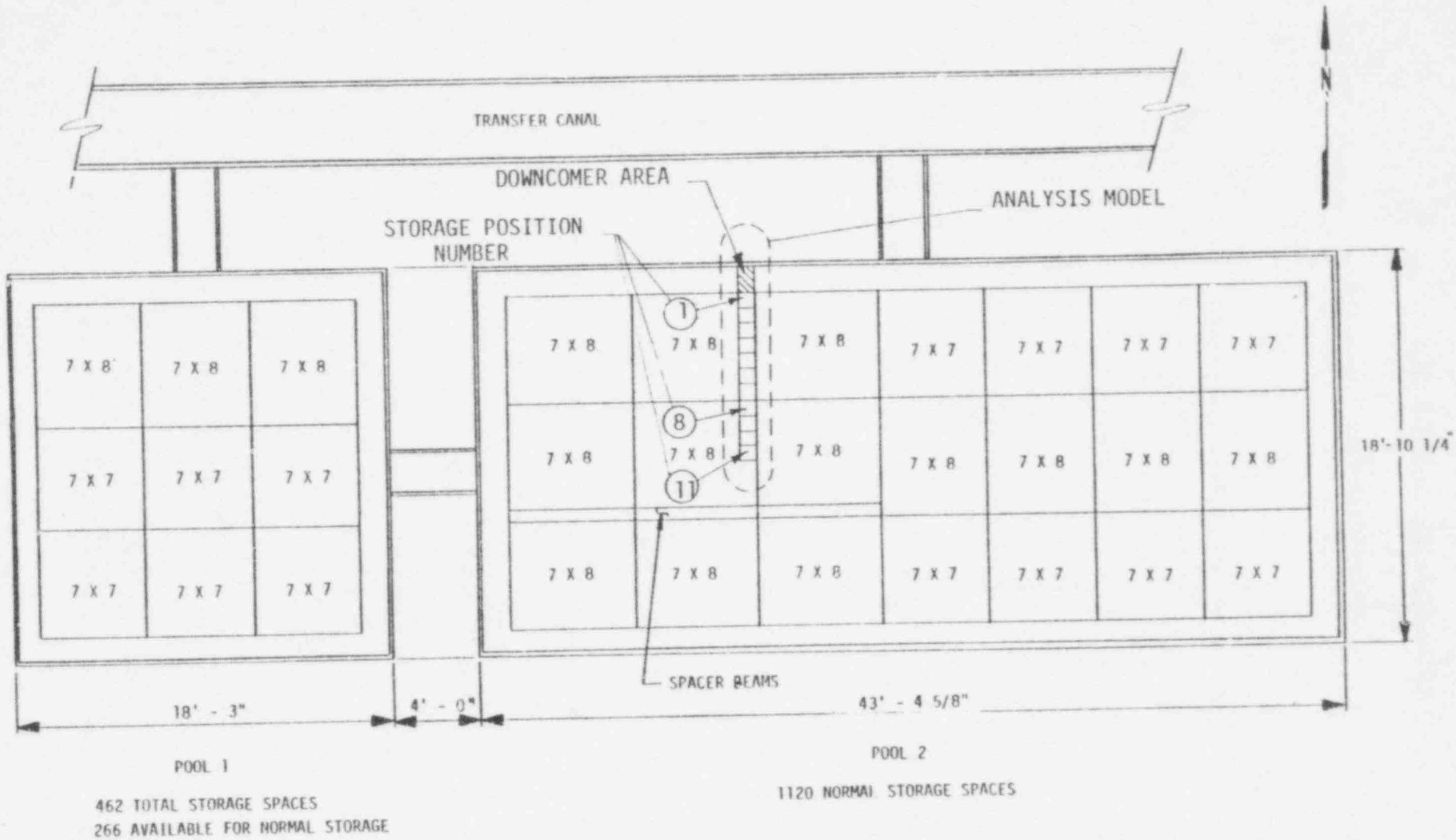

FIGURE 3.5-1: FUEL ASSEMBLY COOLING ANALYSIS MODEL

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Using the models described above, the temperature increase of the water as it passes through the storage tube containing a spent fuel assembly was calculated for a time after reactor shut down of 100 hours.

3.5.1.2 Analysis Results

For the base case with the peak power assembly in position 11, the temperature increase of the water flowing through the storage tube was determined to be 35°F with a corresponding flow rate of 7485 pounds/hour.

The case with the peak assembly in position 8 produced almost identical results of 35.2°F temperature increase and 7454 pounds/hour flow rate.

In both cases the temperature increase for positions with the average power assemblies was less than that described above.

The numbers calculated above are based on a bulk pool temperature of 150°F. However, the bulk pool temperature has a very minor effect on the temperature increase in the fuel storage tube due to water density variation.

3.5.2 Pool Cooling Analysis

3.5.2.1 Analysis Method

The heat generation and pool temperature were calculated using the Nuclear Services Corporation computer code POOLHT. POOLHT calculates fuel decay heat based on Branch Technical Position APCSB 9-2. POOLHT performs an analysis of fuel pool temperature as a function of heat input from spent fuel, heat rejection through the pool

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cooling systems, pool water mass and time. The heat rejection rate in the heat exchangers is calculated based on heat exchanger inlet temperatures, heat transfer coefficient, effective heat transfer surface area, and primary and secondary water flow rates. Finally the time-dependent pool temperature is calculated by an energy balance on the spent fuel pool water.

For this analysis it has been assumed that the first fuel assembly is removed from the reactor and placed into the fuel pool 100 hours after reactor shutdown. The total amount of fuel to be discharged is assumed to be transferred to the fuel pool within 150 hours after reactor shutdown. This same assumption applies to both normal refueling and the full core discharge.

Fuel loading occurs annually for each unit and will be staggered between the two units so that about one-third of a core will be unloaded approximately every six months. Under normal conditions, this will continue until 1362 locations are filled.

As stated earlier a total of 1582 storage positions will be provided. However, 121 positions will be reserved for a full core discharge and 75 positions will be used only during installation of racks in pool 2. With 1362 positions filled from normal refueling the remaining 24 positions are potentially available for normal fuel storage. There are no plans to use these positions, but they are maintained for storage required by unexpected fuel replacements. The increased heat generation due to these 24 assemblies is negligible.

In conformance with the analysis performed for the previous fuel storage modification at Prairie Island the following definitions apply for potential pool cooling conditions:

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Normal	-	Spent fuel pools contain 1362 spent fuel assemblies, leaving room for a full core discharge. One pump and the larger heat exchanger are assumed to be in operation.
Abnormal	-	Spent fuel pools filled with 1483 fuel assemblies, including a recently removed full core. Both pumps and both heat exchangers are assumed to be in operation.
Faulted	-	Loss of operation of a pump from the spent fuel pool cooling system when the spent fuel pools are filled with 1483 fuel assemblies, including a recently removed full core. One pump and the larger heat exchanger are assumed to be in operation.
Accident	-	Loss of all external spent fuel pool cooling when the spent fuel pools are filled with 1483 fuel assemblies, including a recently removed full core.

The "Normal" condition assumes that a routine refueling has just taken place, and that the discharged one-third core results in 1362 spent fuel assemblies stored in the pool. The POOLHT code is then used to determine the heat generation and fuel pool temperature as a function of time after reactor shutdown.

The "Abnormal" condition assumes that after normal operation which results in 1362 spent fuel assemblies in the pool, a full core discharge of 121 assemblies occurs. It is assumed that this full core discharge occurs from one unit 30 days after refueling of the other unit.

The assumed refueling data used in these analyses is shown in Table 3.5-1. The cooling system data used is shown in Table 3.5-2.

TABLE 3.5-1

REFUELING DATA

Batch Number	Number of Assemblies	Total Assemblies In Pool	Normal Refueling Cooling Time	Normal Refueling Irradiation Time	Full Core Discharge Cooling Time	Full Core Discharge Irradiation Time
1	40	40	6022.5	876	6052.5	876
2	40	80	5840	876	5870	876
3	40	120	5657.5	876	5687.5	876
4	40	160	5475	876	5505	876
5	40	200	5292.5	876	5322.5	876
6	40	240	5110	876	5140	876
7	40	280	4927.5	876	4957.5	876
8	40	320	4745	876	4775	876
9	40	360	4562.5	876	4592.5	876
10	40	400	4380	876	4410	876
11	41	441	4197.5	876	4227.5	876
12	41	482	4015	876	4045	876
13	40	522	3832.5	876	3862.5	876
14	40	562	3650	876	3680	876
15	40	602	3467.5	876	3497.5	876
16	40	642	3285	876	3315	876
17	40	682	3102.5	876	3132.5	876
18	40	722	2920	876	2950	876
19	40	762	2737.5	876	2767.5	876
20	40	802	2555	876	2585	876
21	40	842	2372.5	876	2402.5	876

TABLE 3.5.1, REFUELING DATA, Cont'd.

Batch Number	Number of Assemblies	Total Assemblies In Pool	Normal Refueling Cooling Time	Normal Refueling Irradiation Time	Full Core Discharge Cooling Time	Full Core Discharge Irradiation Time
22	40	882	2190	876	2220	876
23	40	922	2007.5	876	2037.5	876
24	40	962	1825	876	1855	876
25	40	1002	1642.5	876	1672.5	876
26	40	1042	1460	876	1490	876
27	40	1082	1277.5	876	1307.5	876
28	40	1122	1095	876	1125	876
29	40	1162	912.5	876	942.5	876
30	40	1202	730	876	760	876
31	40	1242	547.5	876	577.5	876
32	40	1282	365	876	395	876
33	40	1322	182.5	876	212.5	876
34	40	1362	0	876	30	876
35	41	1403	-	-	0	754
36	40	1443	-	-	0	462
37	40	1483	-	-	0	170

Note: All times are given in days.

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TABLE 3.5-2

Spent Fuel Pool Cooling System Data

A. Heat Exchanger 1

Type	Four-pass shell and U-tube
Design Heat Transfer	7.89×10^6 Btu/hr

	Shell	Tubes
Design Temperature	200°F	200°F
Design Flow Rate	1810 gpm	1,415 gpm
Design Inlet Temperature	95°F	120°F
Design Outlet Temperature	104°F	109.6°F
Fluid Circulated	CC wtr	SFP wtr

B. Heat Exchanger 2

Type	Two-pass shell and U-tube
Design Heat Transfer	7.89×10^6 Btu/hr

	Shell	Tubes
Design Temperature	200°F	200°F
Design Flow Rate	1810 gpm	1,415 gpm
Design Inlet Temperature	95°F	140°F
Design Outlet Temperature	104°F	129°F
Fluid Circulated	CC wtr	SFP wtr

Note: The data shown above is the original heat exchanger design data based on the original tube side flow rate of 1,415 gpm. In 1977 the SFPCS pumps were modified to increase the tube side flow rate to 2,200 gpm. The fuel pool calculations are based on this increased flow rate.

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3.5.2.2 Analysis Results

Figures 3.5-2 and 3.5-3 show the heat generation and pool water temperature as a function of time for the normal condition. The maximum pool temperature for this condition is 124°F.

Figures 3.5-4 and 3.5-5 show the heat generation and pool water temperature as a function of time for the abnormal condition. The maximum pool temperature for this condition is 141°F.

Figure 3.5-6 and 3.5-7 show the heat generation and pool water temperature as a function of time for the faulted condition. The maximum pool temperature for this condition is 164°F. This calculation is based on the conservative assumption that only one pump is in operation from the time of reactor shutdown. In actual plant operation, the full core discharge would be delayed if the SFPCS was not in full operation. Therefore, the temperature curve shown in Figure 3.5-6 is an upper bound.

For the accident condition the pool water temperature would rise until boiling at the pool surface began. For this condition the report submitted for the racks presently in service showed that the maximum fuel clad surface temperature for this condition would be 252°F. This is well below the normal fuel operating temperature.

Based on the assumption that all cooling is lost at the peak heat generation point for the abnormal condition, 2.9 hours would elapse before initiation of boiling in pool 1. The maximum evaporation rate would occur at this time and would equal 44.7 gpm. This assumes pool 1 is isolated and the recently discharged full core is located in pool 1. If complete mixing of water in the two pools is assumed, 9.9 hours would elapse before initiation of boiling.

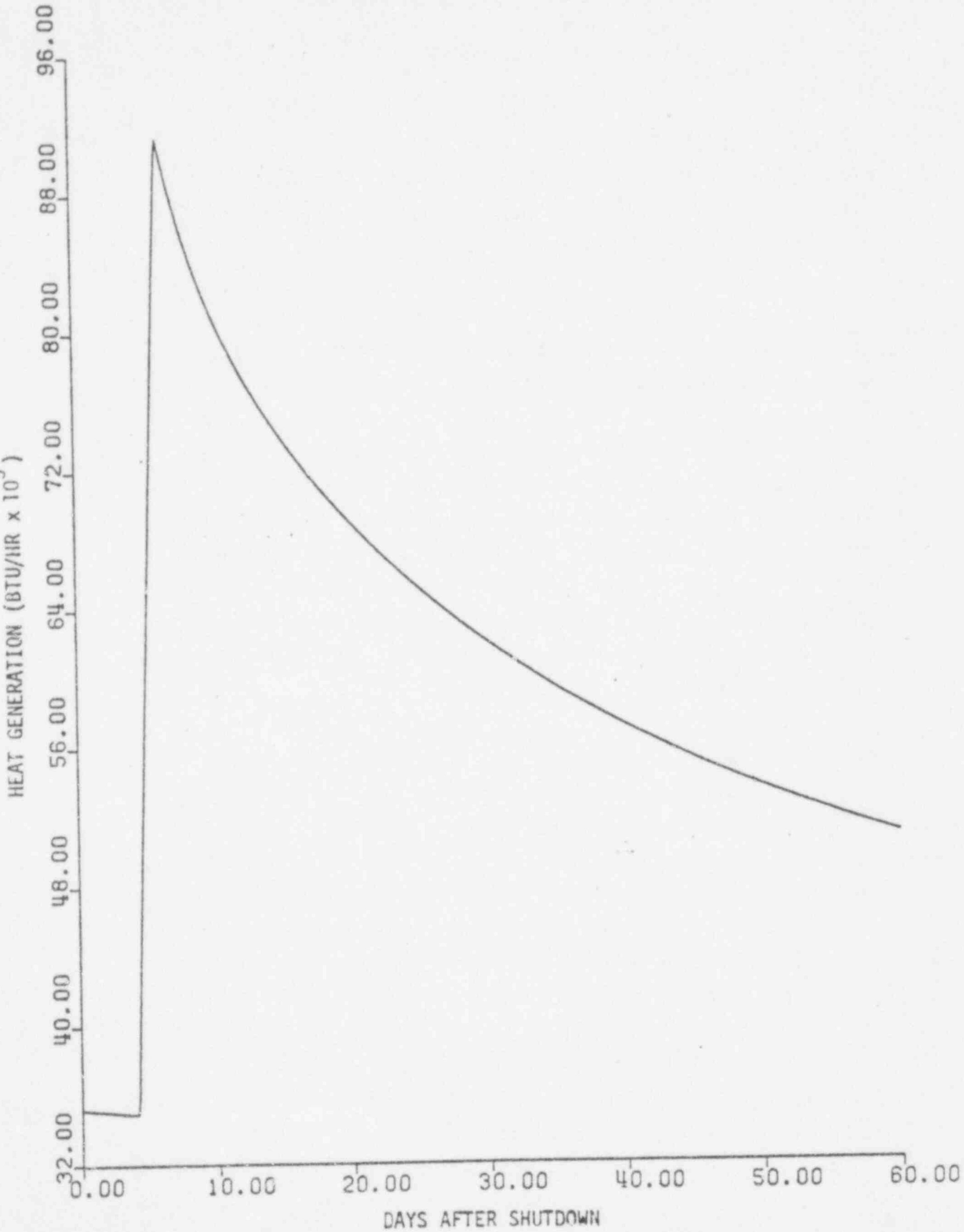

FIGURE 3.5-2: PRAIRIE ISLAND
NORMAL CONDITION
HEAT GENERATION

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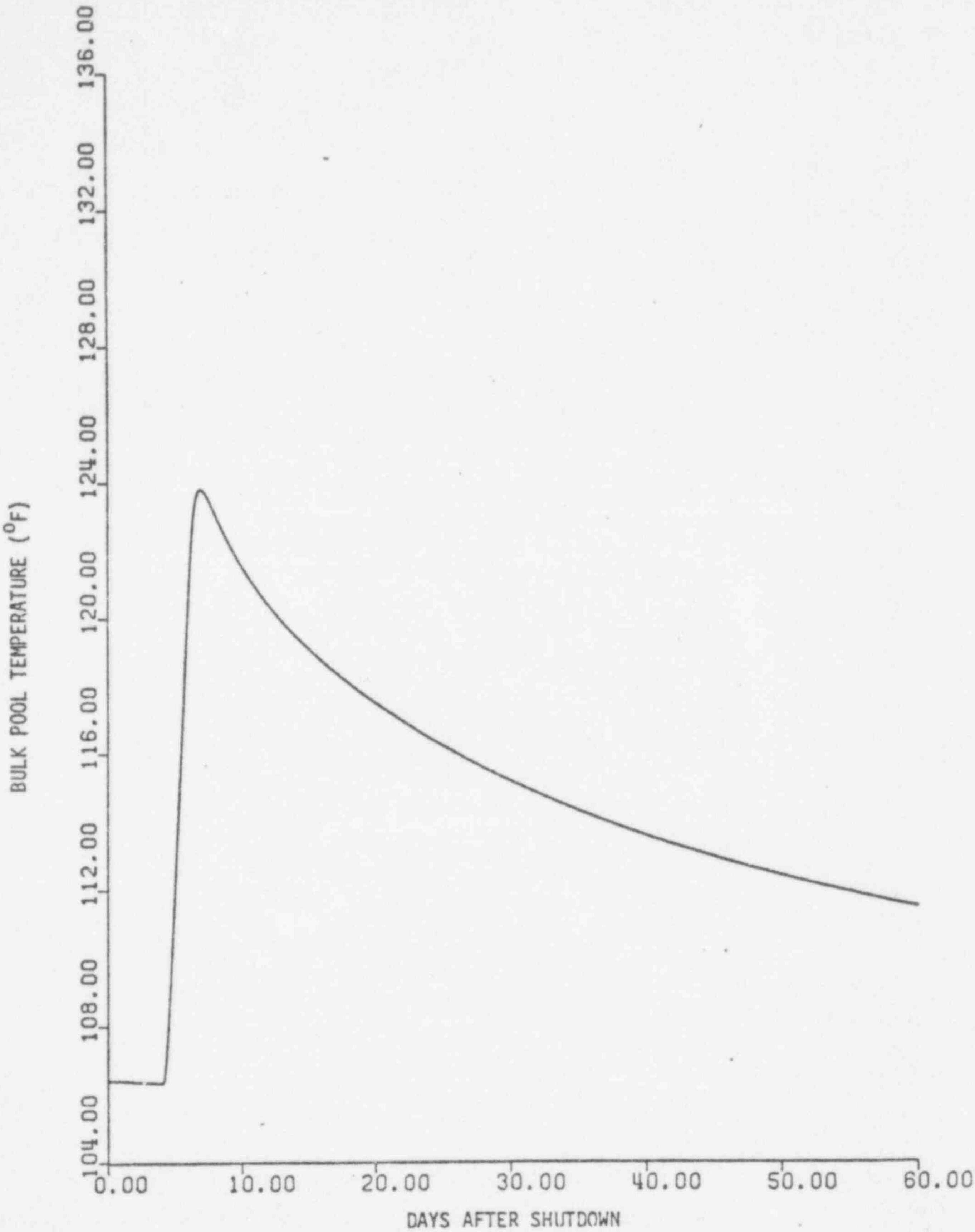

FIGURE 3.5-3: PRAIRIE ISLAND NORMAL CONDITION POOL TEMPERATURE

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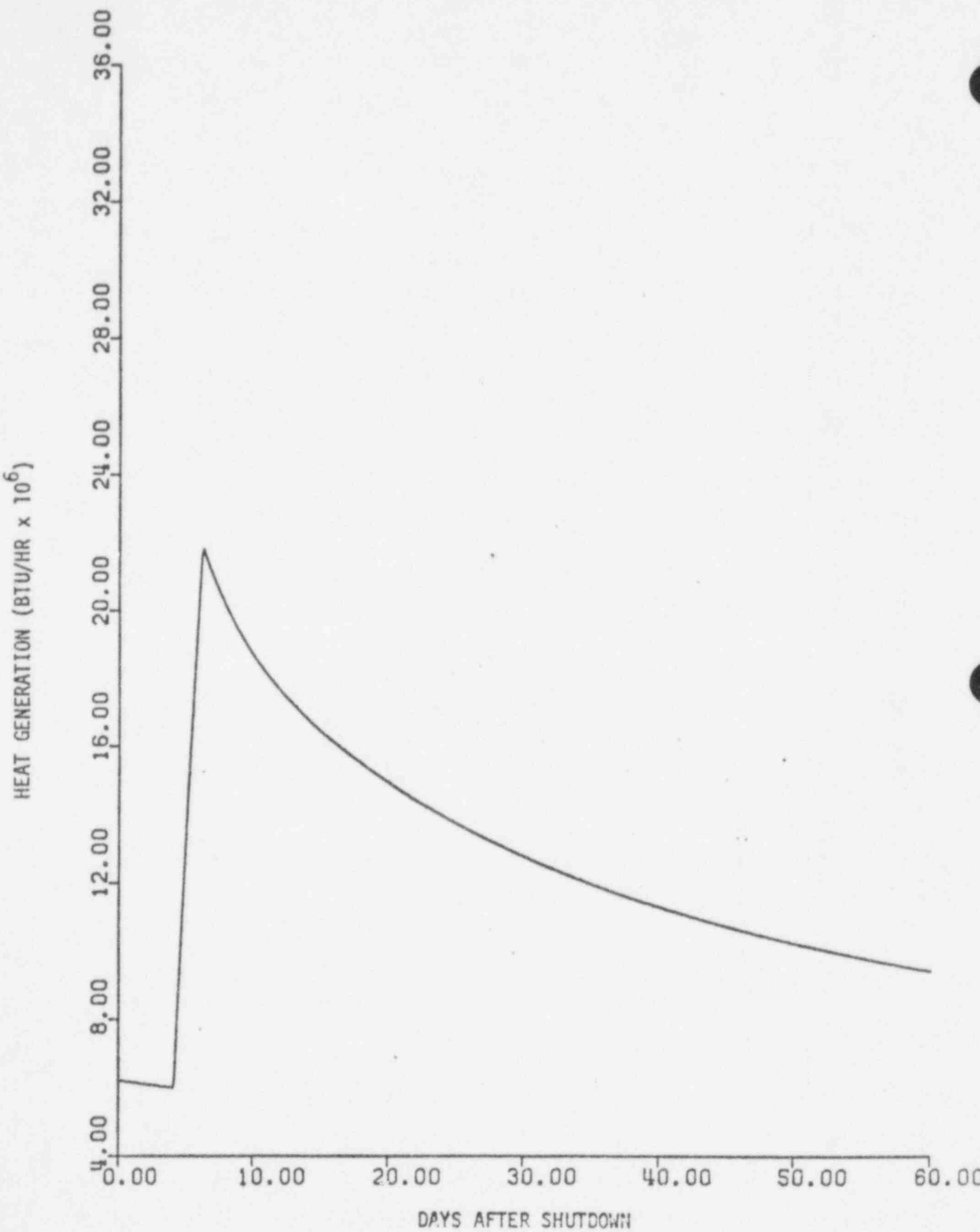

FIGURE 3.5-4: PRAIRIE ISLAND
ABNORMAL CONDITION
HEAT GENERATION

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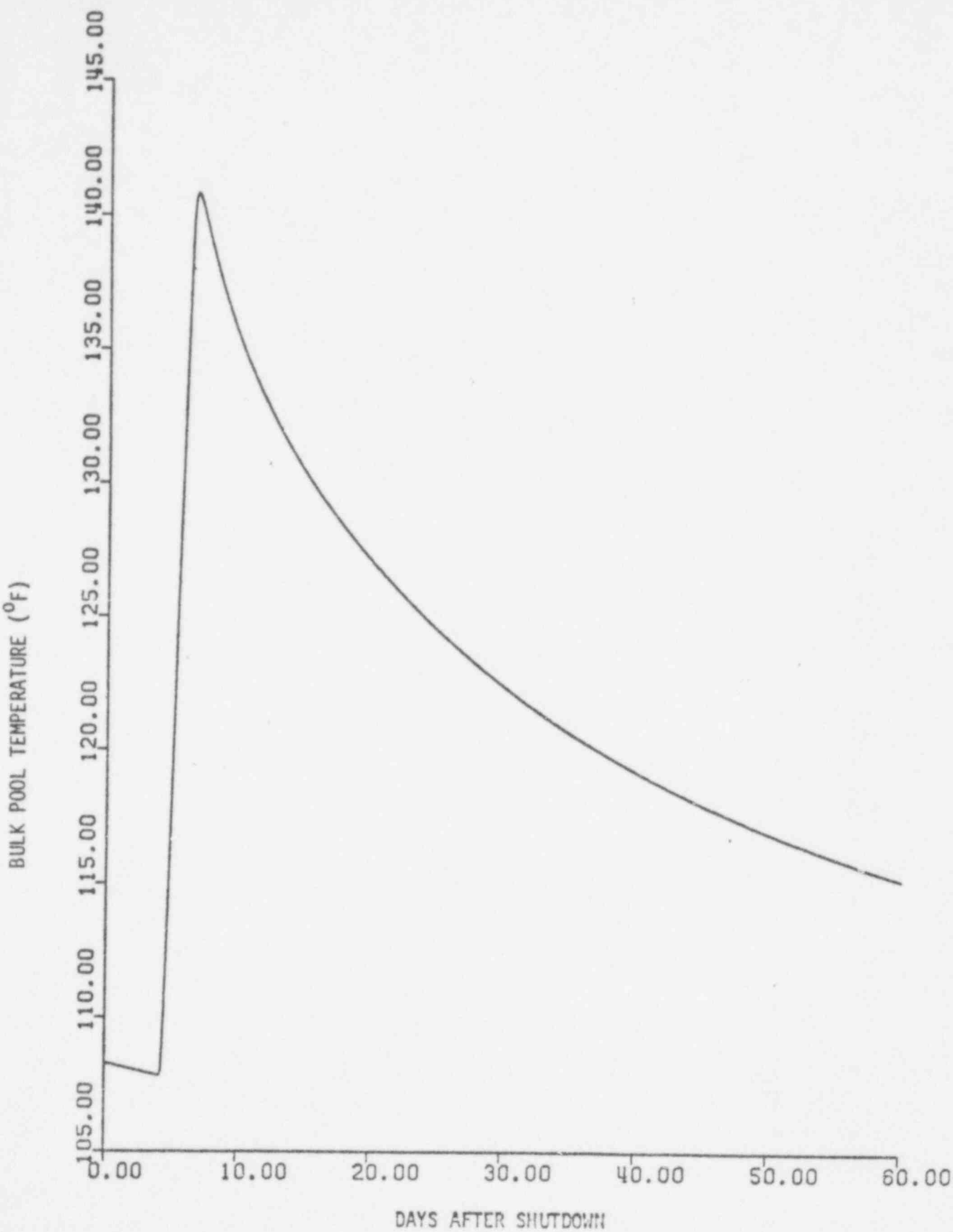

FIGURE 3.5-5: PRAIRIE ISLAND
ABNORMAL CONDITION
POOL TEMPERATURE

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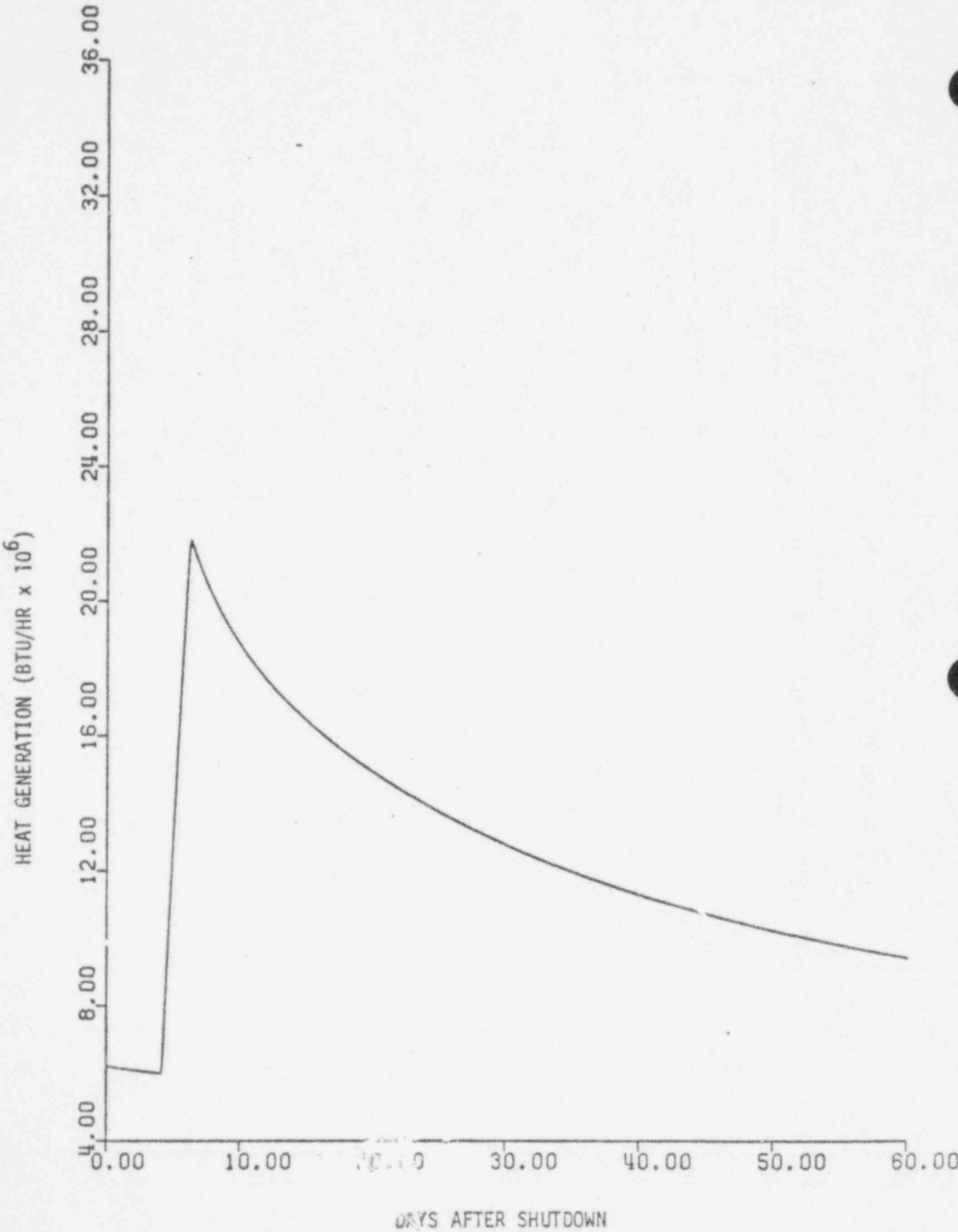

FIGURE 3.5-6: PRAIRIE ISLAND
FAULTED CONDITION
HEAT GENERATION

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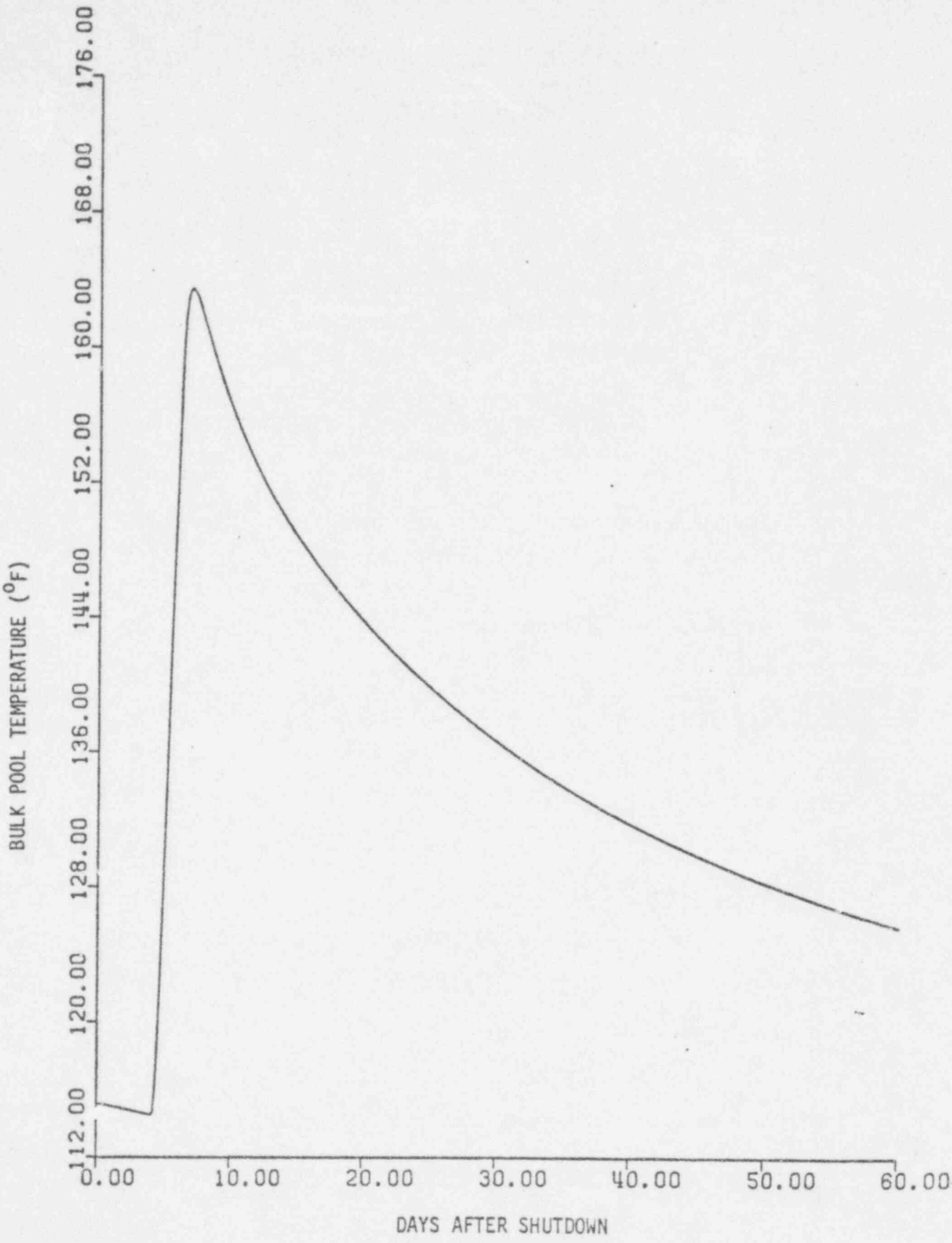

FIGURE 3.5-7: PRAIRIE ISLAND
FAULTED CONDITION
POOL TEMPERATURE

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Makeup to the spent fuel pit cooling system can be provided from the following systems:

a) Chemical Volume and Control System - provides borated water (at concentration selected by the operator) to the discharge side of the SFP heat exchanger

b) Demineralized Water - provides demineralized water to the discharge side of the DFP demineralizer

c) Reactor Makeup Water - provides demineralized water to the discharge side of the SFP heat exchanger

d) Refueling Water Storage Tank

e) CVCS Holdup Tanks via the CVCS Holdup Tank Recirculation Pump

f) Fire Protection System 2

3.6 Radiochemical Analysis

The purpose of this analysis is to determine the dose rate at the pool surface as a result of the concentration of fission products in the fuel pool water.

3.6.1 Analysis Method

Based on the refueling schedule and fuel leak rate, the radioactivity content of the fuel pool water in curies per unit volume is calculated. From this value the surface dose rate is determined.

To solve this problem, Nuclear Services Corporation has developed the POOLRAD computer program. The model on which the code is based

treats the fission product concentration in the pool as a dynamic equilibrium between the pool cleanup system and the diffusion of fission products from the fuel matrix. A closed form solution for the leakage rate from the fuel was developed based on time dependent diffusion from a sphere.

Initial concentrations of 30 fission product isotopes are computed for the time of fuel discharge from the reactor in each of several fuel batches. The equilibrium concentration is then computed for each isotope on a batch-by-batch basis for each of several input time points. Total curies per cubic centimeter and pool surface dose rate are then computed at each time. The results are obtained for 6, 10, 20, 40 and 100 days after shutdown.

The isotopic data were obtained from the ORNL-4628 "ORIGEN- The ORNL Isotope Generation and Depletion Code". The cleanup efficiency for the different isotopes were obtained from NUREG-0016.

It is assumed in this analysis that one percent of the fuel leaks. The diffusion coefficient for Cesium is taken as 10^{-19} sq. cm/sec. This is the same as the value for Krypton-85 below 800°C and is considered to be a conservative value. Data published for reactor primary systems at power would lead to the use of a lower diffusion coefficient for Cesium and, therefore, lower calculated dose rates.

3.6.2 Analysis Results

The results of this analysis indicate that Cs will be the main fission product contaminant in the spent fuel storage pool. Table 3.6-1 presents the calculated water radioactivity content and surface dose rate for 1% fuel leakage. The results are well within acceptable limits.

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TABLE 3.6-1

FUEL POOL WATER RADIOACTIVITY CONTENT
AND SURFACE DOSE RATE FOR
1% FUEL LEAKAGE

Days From Shutdown	Radioactivity Content (curies/cc)	Surface Dose Rate (mr/hr)
6	4.05×10^{-10}	0.44
10	2.69×10^{-10}	0.29
20	1.51×10^{-10}	0.16
40	1.00×10^{-10}	0.11
100	8.07×10^{-11}	0.09

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3.7 Fuel Rack Installation

The spent fuel racks will be installed in the two spent fuel pools as shown in Figure 3.1-1. Th installation will be done according to the following sequence:

1. The spent fuel (442 assemblies) will be stored in pool 2.
2. The portable 15 ton crane will be installed on the fuel handling bridge rails.
3. Using the portable crane and the main crane the present racks will be removed from pool 1.
4. Some of the existing anchor bolts will be cut off flush with the pool liner in order to provide clearance for the rack feet. This work will be done in accordance with approved procedures and proven methods in order to avoid damage to the pool liner.
5. The new racks will be installed in pool 1.
6. All spent fuel will be transferred from pool 2 to pool 1.
7. The protective cover will be installed on pool 1.
8. The present racks will be removed from pool 2.
9. The new racks will be installed in pool 2.
10. Spent fuel will be returned to pool 2.

This procedure will not require the movement of heavy equipment above stored fuel with the exception of movement of equipment above pool 1 when the protective cover is in place. The protective cover has been designed to withstand the drop of a spent fuel rack.

4.0 CONCLUSIONS

The proposed modification to the Prairie Island Nuclear Generating Plant spent fuel storage as described herein has been evaluated in the following areas:

1) Fuel Rack Mechanical Design
2) Nuclear effects
3) Structural and seismic effects, including fuel handling accidents
4) Thermohydraulic effects
5) Radiochemical effects
6) Fuel rack installation

The results of the above evaluations have shown that there are no unresolved safety problems associated with the proposed modification.

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EXHIBIT D

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

License Amendment Request dated January 31, 1980

Docket Nos. 50-282	License Nos. DPR-42
50-306	DPR-60

Exhibit D consists of the Nuclear Services Corporation document:

QUAD-1-79-558

"Fuel Pool Building Structural Evaluation
for Prairie Island Nuclear Generating Plant
Units 1 and 2 Spent Fuel Storage Modification"

QUAD-1-79-558
JOB NO. NOR-0174

FUEL POOL BUILDING STRUCTURAL EVALUATION
FOR
PRAIRIE ISLAND NUCLEAR GENERATING PLANT
UNITS 1 AND 2
SPENT FUEL STORAGE MODIFICATION
NSP PROJECT NUMBER E-78Y075

Prepared for

NORTHERN STATES POWER COMPANY
Minneapolis, Minnesota

By:

NUCLEAR SERVICES CORPORATION
A Division of Quadrex Corporation
1700 Dell Avenue
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REV. NO.	ENGINEER	REVIEWER	PROJECT QA ENGINEER	PROJECT ENGINEER	ISSUING MANAGER	DATE OF APPROVAL

NUCLEAR SERVICES CORPORATION

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CAMPBELL, CALIFORNIA 95008

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1.0 INTRODUCTION

This report presents the structural analyses and evaluations performed by Nuclear Services Corporation to determine the structural adequacy of the spent fuel pool building structures of Prairie Island Nuclear Generating Plant (Units 1 & 2) to support the additional loads that would result from the proposed fuel pool storage modifications. The configurations of the fuel pool structures and the arrangement of the proposed high density racks in the pool are described in Section 2.0. Section 3.0 lists the loads and load combinations for which the pool structures were evaluated. This section also describes the acceptance criteria used to determine the structural adequacy. Methods of analysis and load computations are presented in Section 4.0. The evaluation of section capabilities to withstand the computed loads has been summarized in Section 5.0. Section 6.0 lists the references used.

2.0 DESCRIPTION OF SPENT FUEL POOL STRUCTURES

The fuel storage area at Prairie Island Nuclear Generating Plant is located between the two reactor buildings, and consists of a new fuel pit, two pools for storing spent fuel, and a canal for transfer of fuel elements. Figures 2-1 and 2-2 show plans of the fuel storage area at elevations 715 ft. and 693 ft. A typical cross section through the pool is shown in Figure 2-3.

The two spent fuel storage pools are designated as Pool No. 1 and Pool No. 2. Pool No. 1, the smaller of the two pools, has inside plan dimensions of 18'-11" x 18'-3". Pool No. 2 has inside plan dimensions of 18'-11" x 43'-5". Normal water depth for both pools is about 40 feet. Pool No. 1 is presently designated as cask loading and unloading area. However, in the proposed modification, as shown in Figure 2-4, this pool would store nine racks. Pool No. 2 will hold 21 racks. These racks are free standing and are not anchored to the floor or the walls (Reference 1).

For the purpose of evaluation, the fuel pool structures were divided into three groups: (1) the spent fuel pool floor supporting the racks, (2) the pool walls above the pool floor, and (3) the shear walls and columns M_b - 10_b and M_b - 7_b below the pool floor.

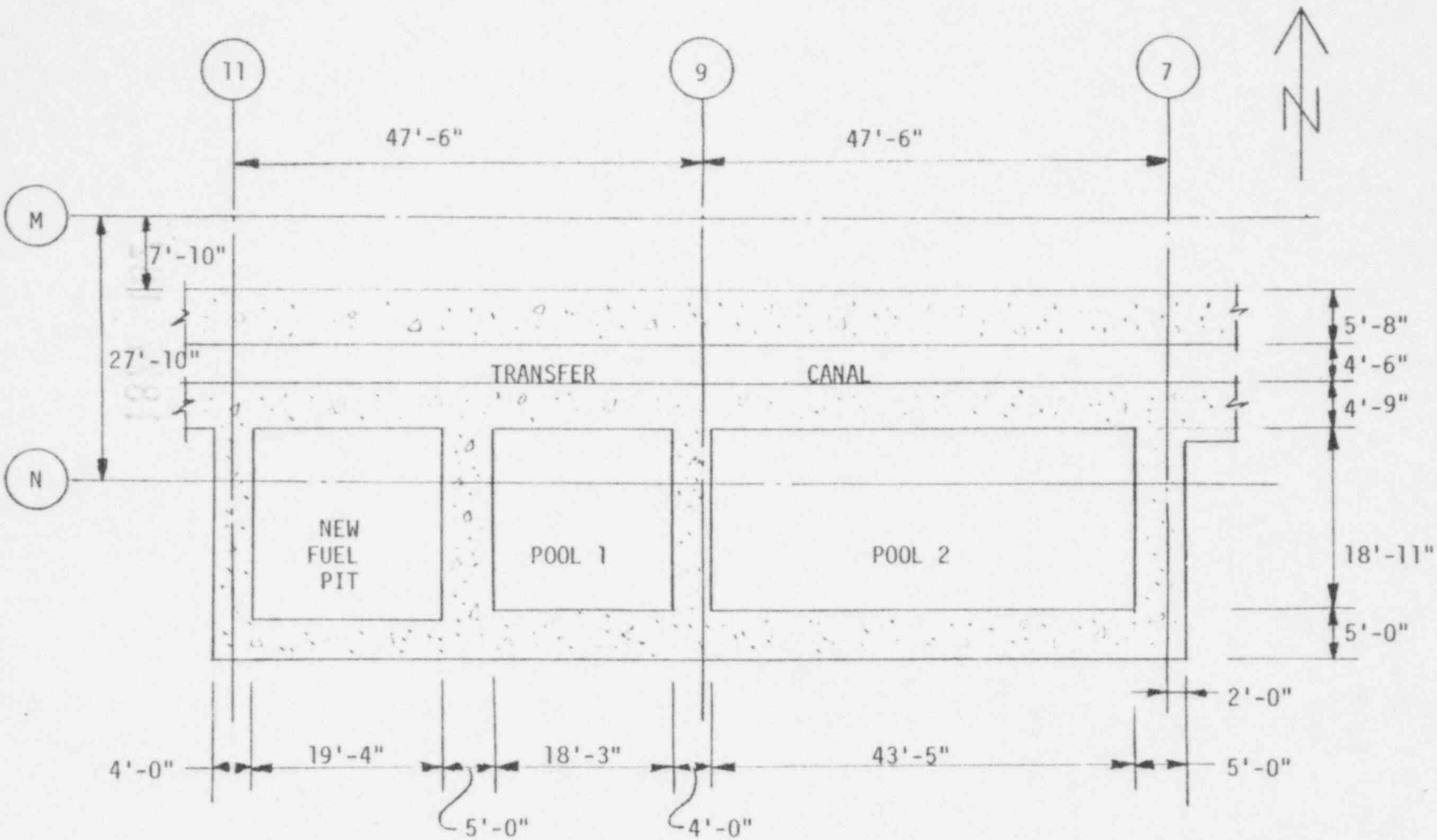

FIGURE 2-1 PLAN OF SPENT FUEL POOL AREA AT 715 FT. (REF. 11)

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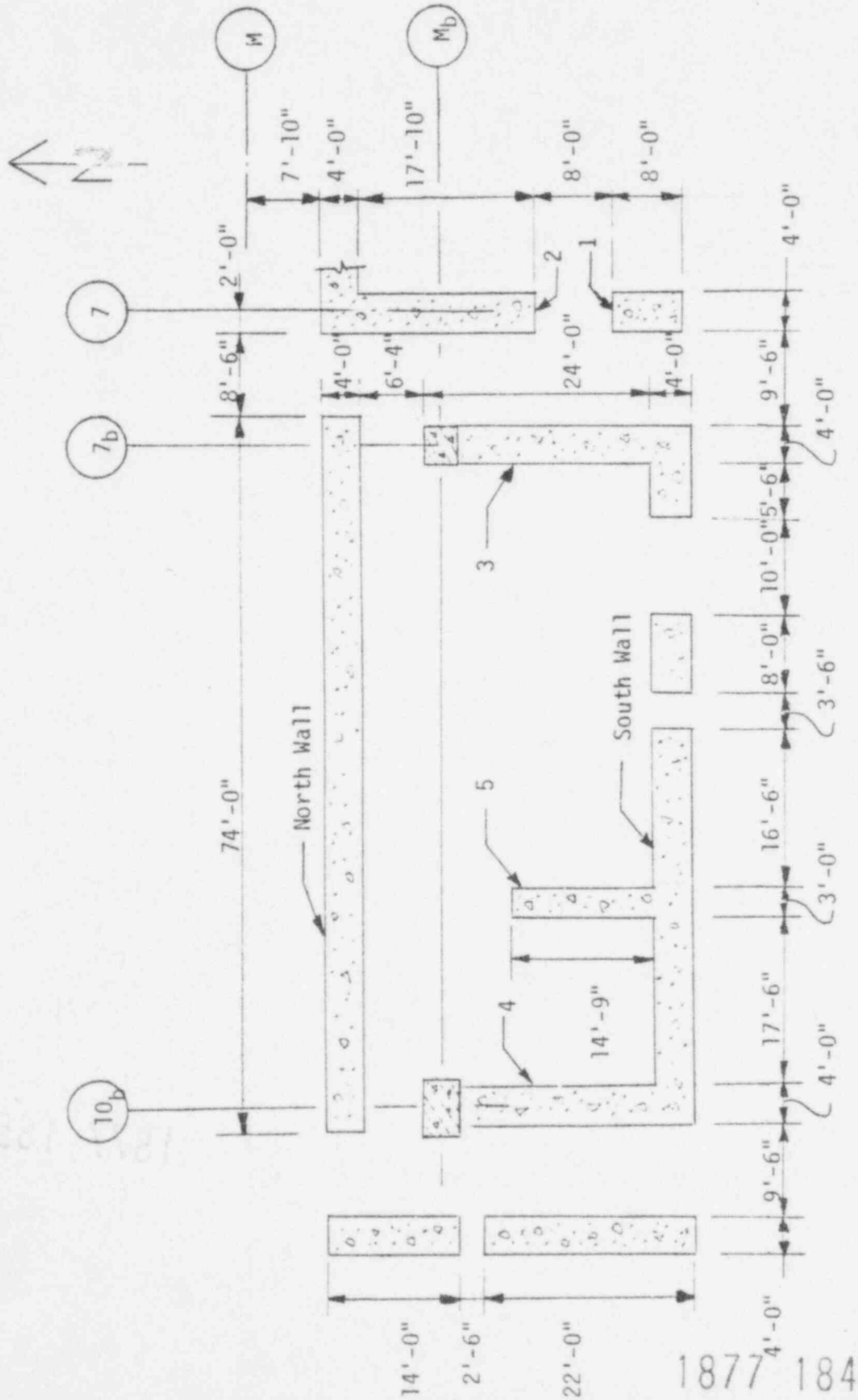

FIGURE 2-2 PLAN OF SPENT FUEL POOL AREA AT ELEVATION 693 FT. (REF. 6,11)

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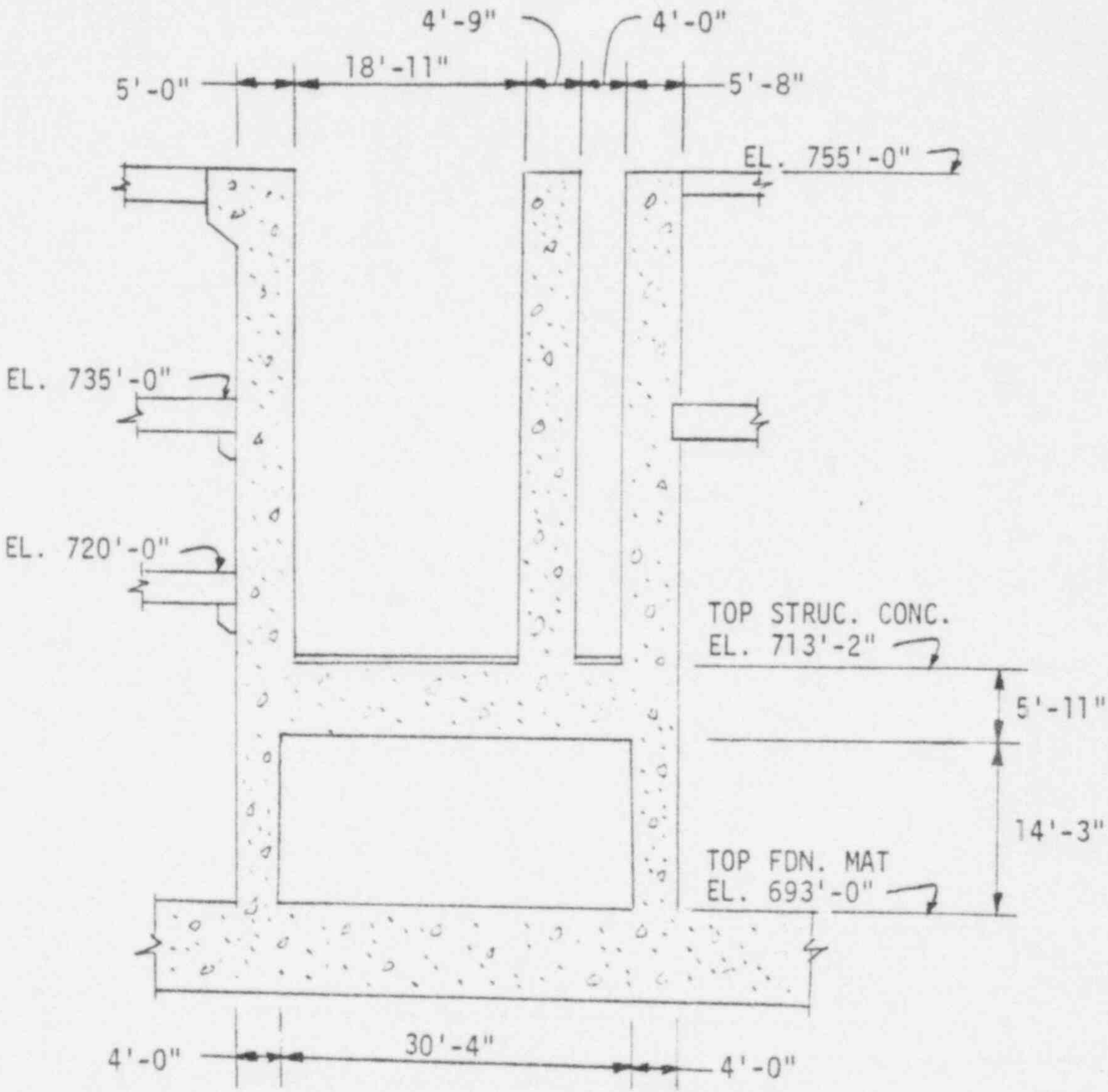

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FIGURE 2-3 TYPICAL CROSS-SECTION THROUGH SPENT FUEL POOL

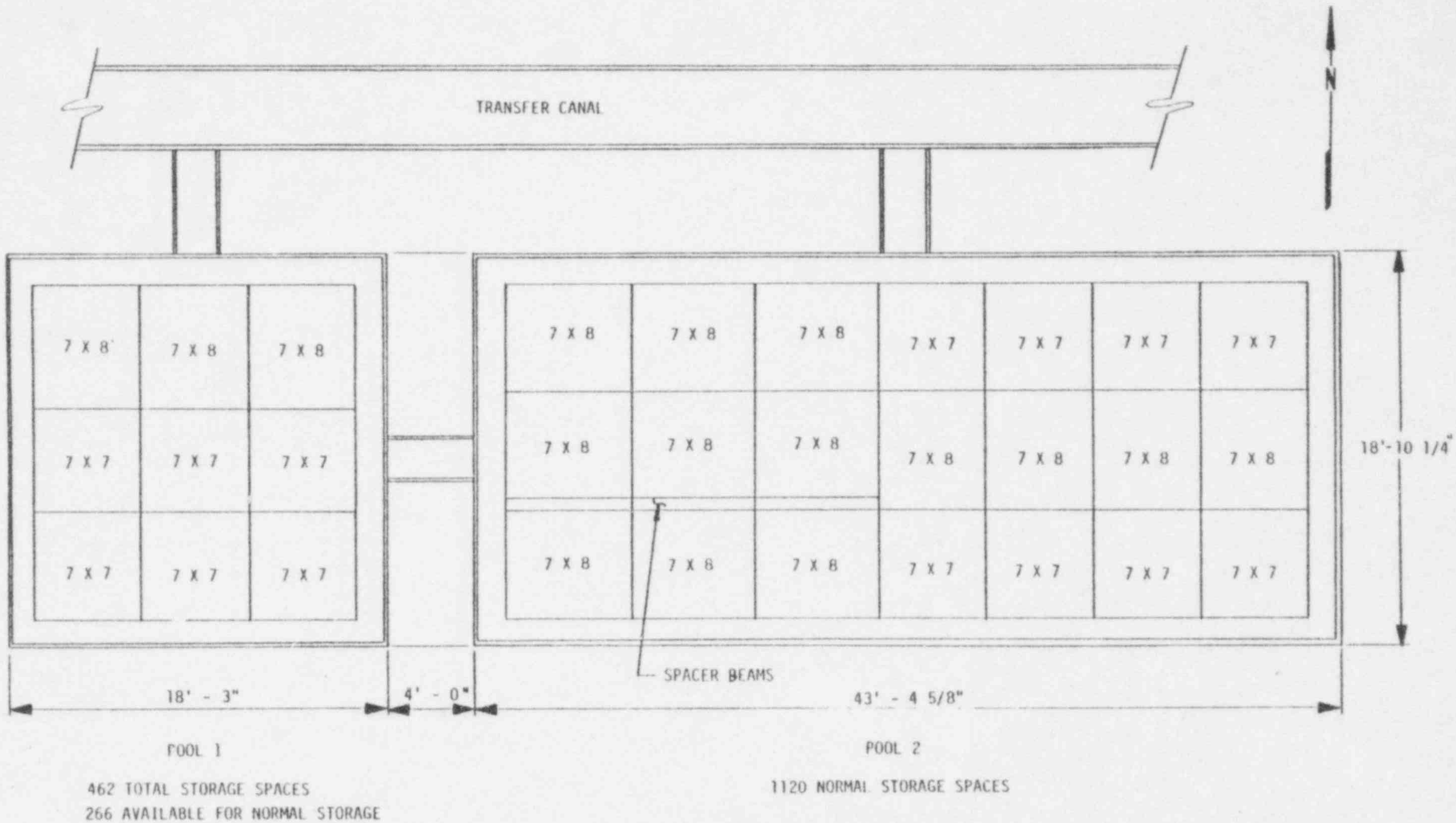

FIGURE 2-4 PROPOSED ARRANGEMENT OF SPENT FUEL RACKS

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3.0 LOADS, LOAD COMBINATIONS & ACCEPTANCE CRITERIA

3.1 Loads

Fuel pool structures were evaluated for the following loads:

a) Deadweight (D) -- Weight of pool structure components including the buoyant weight of the proposed spent fuel racks.

b) Hydrostatic Load (H) -- Loads on the pool floor and walls exerted by the water in the pool.

c) Operating Basis Earthquake (E) -- Loads from horizontal and vertical components of the Operating Basis Earthquake (OBE) using the applicable vibratory motion from Reference 2.

d) Safe Shutdown Earthquake (E') -- These loads were assumed to be twice that of OBE loads.

e) Thermal Load (To, Ta) -- Loads resulting from the increase in fuel pool water temperature during normal operation (To) and during accident condition (Ta).

For the modified pool arrangement shown in Figure 2-4, the buoyant deadweight of the loaded racks was computed as 2.0 kips per square foot (ksf). However, for the present evaluation this was conservatively increased to 2.4 ksf. Response spectra values for OBE horizontal and vertical motion corresponding to 1% damping are listed in Tables 3-1 and 3-2 and are developed from Reference 2. To determine the thermal loads, normal operating temperature of the pool water was assumed to be 120°F; water temperature during accident condition was assumed to be 212°F.

3.2 Load Combinations and Acceptance Criteria

The spent fuel pool structures were evaluated as Seismic Category I structures in accordance with USNRC Regulatory Guide 1.29 (Reference 3). Their structural adequacy was verified in accordance with USNRC Standard Review Plan Section 3.8.4 (Reference 4), using section strength per American Concrete Institute (ACI) Building Code 318-77 (Reference 5). Table 3-3 lists the pertinent loading combinations for which the pool structures were evaluated. It also lists the respective permissible loading limits.

TABLE 3-1
RESPONSE SPECTRUM FOR OBE HORIZONTAL MOTION
(1% DAMPING)

Period (sec)	Frequency (cps)	Acceleration (g's)
.025	40.00	.073
.050	20.00	.083
.075	13.33	.094
.100	10.00	.104
.150	6.67	.130
.200	5.00	.190
.275	3.64	.595
.325	3.08	.822
.375	2.67	1.227
.425	2.35	.599
.600	1.67	.308
.800	1.25	.164
1.000	1.00	.140
1.500	.67	.092
2.000	.50	.043

TABLE 3-2
RESPONSE SPECTRUM FOR OBE VERTICAL MOTION
(1% DAMPING)

Period (sec)	Frequency (cps)	Acceleration (g's)
.025	40.00	.061
.050	20.00	.065
.100	10.00	.095
.150	6.67	.138
.200	5.00	.210
.238	4.20	.635
.250	4.00	.635
.400	2.50	.340
.500	2.00	.190
.600	1.67	.120
.800	1.25	.068
1.000	1.00	.060
1.500	.67	.042
2.000	.50	.025

TABLE 3-3

LOAD COMBINATION AND EVALUATION CRITERIA

Loading Condition	Design Method	Pertinent Load Combination Used (1)	Loading Limit (2)
Service	Working Stress	$D + H$	S
		$D + H + E$	S
	Ultimate Strength	$1.4\ D + 1.4\ H$	U
		$1.4\ D + 1.4\ H + 1.9\ E$	U
		$(1.4\ D + 1.4\ H + 1.7\ T_o + 1.9\ E)\ 0.75$	U
Factored	Ultimate Strength	$D + H + T_o + E'$	U
		$D + H + T_a$	U
		$D + H + T_a + 1.25\ E$	U
		$D + H + T_a + E'$	U

(1) For definitions of symbols, see Section 3.1.

(2) S is the required section strength based on the working stress design method and the allowable stresses defined in ACI-318-77.

U is the section strength required to resist design loads based on the strength design methods described in ACI-318-77.

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4.0 METHOD OF ANALYSIS AND COMPUTATION OF DESIGN LOADS

This section describes the analyses and the computations of design loads for the components of the fuel pool structure listed in Section 2.0.

4.1 Spent Fuel Pool Floor

The spent fuel pool floor consists of a 5'-11" thick reinforced concrete slab supported as shown in Figure 2-3. This was analyzed for dead loads and hydrostatic loads using a finite element model shown in Figure 4-1. For evaluating the pool floor, this model is conservative since it assumes one-way slab action. Also, the support provided by the 4'-9" thick canal wall (supported between two north-south walls and acting as a deep beam) was conservatively ignored. Instead, the dead weight of the canal wall was assumed to be supported by the floor. The analysis was performed using the computer program STARDYNE (Reference 7). To determine the worst loading, two load cases were considered, one with the transfer canal empty and the other with the canal full of water.

To determine the seismic loads on the pool floor due to the vertical component of the earthquake motion, the out-of-plane fundamental frequency of the pool floor was computed using the method outlined in Reference 8. An upper bound value of the amplification factor in the vertical direction was obtained by computing the upper bound and lower bound frequencies of the pool floor slab and selecting the largest amplification factor in that frequency range. The OBE response in the vertical direction at 1% damping, thus computed, was 0.063g, but a conservative value of 0.08g was used in the evaluation. For SSE vertical motion, a value of 0.16g was used.

The seismic loads on the pool floor resulting from the horizontal seismic motion of the pool water were computed using methods outlined in Reference 9. These loads were combined with those due to vertical seismic motion by the square-root-of-the-sum-of-the-square (SRSS) method.

Thermal loads in the pool floor slab due to T_o and T_a were computed using methods outlined in Reference 10 and assuming a cracked concrete section.

Dead load, hydrostatic loads, seismic loads and thermal loads were combined in accordance with Table 3-3 to determine the design factored forces, moments, and shears. The resulting design values for the critical load combinations are listed in Table 4-1.

4.2 Spent Fuel Pool Walls Above Pool Floor

The proposed high density spent fuel racks are free standing and would not have any lateral wall supports, and so would not impose any transverse loads on the pool walls above the floor. Hence, with the proposed modification, the design loads for these walls would be smaller than those with the existing racks, since the existing racks are designed to transmit lateral seismic and thermal loads to the pool walls. Thus, no new evaluation of the pool walls above the floor is necessary.

To be conservative, the above justification was not used in the case of the 4'-9" thick wall separating the pools and the transfer canal (see Figures 2-1 and 2-3). Even though, as a conservative approach, the spent fuel pool floor was evaluated assuming that it supports this canal wall, the latter was evaluated assuming that it carries half of the load from the spent fuel pool floor as a deep beam supported between the two north-south cross-walls. The resulting design moments and shears for the critical load combinations are listed in Table 4-2.

4.3 Shear Walls and Columns Supporting the Pool Floor

The shear walls and columns supporting the pool floor are shown in Figure 2-2. Due to the increased deadweight of the racks and fuel bundles on the spent fuel pool floor, dead load stresses and seismic overturning stresses (compressive and shear) on these walls and columns

will be larger than those predicted during the earlier densification study (Reference 6). The difference between the existing rack loads on the pool floor (1.7 ksf) and the proposed rack loads with the high density racks (2.0 ksf) is only 0.3 ksf. This additional load is about five percent of the total floor load resulting from dead weight of the slab, existing racks, and canal wall and the hydrostatic loads. However, to be conservative, the shear walls and columns supporting the pool floor were evaluted for an additional floor load of 0.7 ksf instead of 0.3 ksf. The dead load compressive stresses and seismic bending and shear stresses thus computed were added to those computed earlier in Reference 6. These are listed in Tables 4-3 and 4-4.

Columns M_b - 10_b and M_b - 7_b support the canal wall separating the transfer canal and the pools. As was stated in Section 4.2, to be conservative, this wall was assumed to support the pool floor in deep beam action; hence the loads computed for the supporting columns are also very conservative. The design loads on these columns were determined by adding the reaction loads from the canal wall (acting as a beam) to the other existing loads. The latter loads were obtained from Reference 6. The combined loads are shown in Table 4-5. The computed additional loads, shown in this table, include the seismic effect on the additional rack weight on the pool floor.

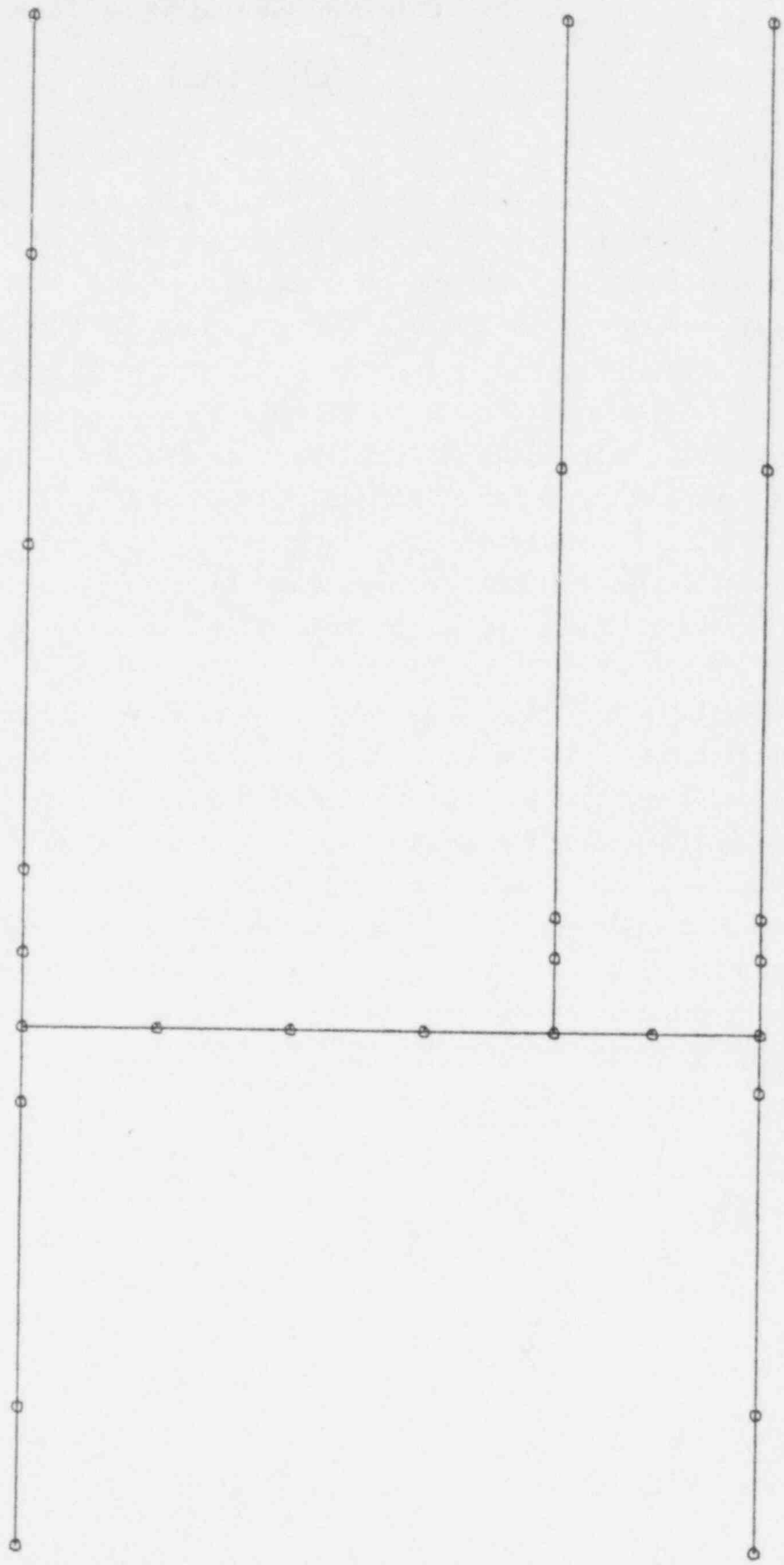

FIGURE 4-1 FINITE ELEMENT MODEL USED FOR POOL FLOOR EVALUATION

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TABLE 4-1

MAXIMUM POOL FLOOR LOADS FOR CRITICAL LOAD COMBINATIONS

Consideration	Canal Condition	Critical Load Combination	Moment (K-ft/ft)	Axial Force (k/ft)	Shear (k/ft)
Bending	Full	1.4 D + 1.4 H	-350	47.3	----
		1.4 D + 1.4 H + 1.9 E	-498	59.5	----
		D + H + T_a	773	33.8	----
		D + H + T_a + E'	855	46.6	----
	Empty	1.4 D + 1.4 H	-448	65.8	----
		1.4 D + 1.4 H + 1.9 E	-600	78.9	----
		D + H + T_a	865	47	----
		D + H + T_a + E'	1022	60.8	----
Shear	Full	1.4 D + 1.4 H	-112	----	75.7
		1.4 D + 1.4 H + 1.9 E	-224	----	86.0
	Empty	1.4 D + 1.4 H	-210	----	68.6
		1.4 D + 1.4 H + 1.9 E	-324	----	78.3

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TABLE 4-2

MAXIMUM BENDING MOMENTS AND SHEARS IN THE CANAL WALL

Critical Load Combination	Consideration	Load	Location	
			At Critical Section	At Support
1.4D + 1.4H + 1.9E	Bending	Moment (k-ft)	17,200(1)	-34,117
	Shear	Moment (k-ft)	-29,012(2)	-34,117
		Shear (kips)	2,170(2)	3,226

NOTES: (1) At the center of span

(2) At the gate

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TABLE 4-3

SHEAR STRESSES IN THE WALLS SUPPORTING THE POOL FLOOR

Shear Wall (1)	Existing Shear Stress (Psi)		Additional Shear Stress (Psi)		Total Shear Stress (Psi)	
	OBE	SSE	OBE	SSE	OBE	SSE
1	29.2	58.4	2.0	3.9	31.2	62.3
2	48.0	95.9	3.6	7.2	51.6	103.1
3	34.4	68.7	3.8	7.6	38.2	76.3
4	34.4	68.7	3.8	7.6	38.2	76.3
5	3.2	6.3	3.5	6.9	6.7	13.2
North Wall	18.9	39.7	2.5	5.0	21.4	44.7
South Wall	24.6	49.1	3.1	6.1	27.7	55.2

(1) See Figure 2-2

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TABLE 4-4

COMPRESSIVE STRESSES IN CRITICAL SHEAR WALLS

SUPPORTING THE POOL FLOOR[1]

Shear Wall[2]	Existing Comp. Stress (Psi)	Additional Comp. Stress (Psi)	Total Comp. Stress (Psi)
1	670	53	723
2	670	53	723
5	184	54	238
South Wall	618	19	637

NOTE: (1) For D + H + E' Loadings

(2) See Figure 2-2

TABLE 4-5

LOADS ON COLUMNS M_b - 10_b AND M_b - 7_b[1]

Column No.	Loads with (2) Existing Racks (kips)	Loads with Proposed Racks (kips)
M_b - 7_b	1773	2195
M_b - 10_b	2972	3549

NOTE: (1) Includes SSE Loads

(2) From Ref. 6

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5.0 EVALUATION RESULTS & CONCLUSIONS

5.1 Evaluation of Pool Floor

The pool floor slab was evaluated from two considerations: (1) combined bending and axial loads, and (2) shear.

Like most reinforced concrete floor slabs in nuclear power plants, the pool slab is under-reinforced, i.e., its bending strength is limited by the amount of tensile reinforcement. The tensile steel cross-sectional area requirements for combined bending and tension loads listed in Table 4-1 were computed using the strength method. These requirements are compared to the available steel area in Table 5-1.

Shear capacities of the pool slab were also calculated for load combinations and loads listed in Table 4-1, and compared to the computed factored shear loads. This comparison is shown in Table 5-2.

5.2 Evaluation of Spent Fuel Pool Canal Wall

The canal wall separating the pools and the transfer canal was analyzed as a beam spanning between Columns M_b - 10_b and M_b - 7_b. For evaluating the bending capacity, the minimum depth of this beam (at the two gates) was used. Shear capacities were computed at the face of the supporting columns as well as at the location where the depth is minimum due to the cut-out for the gates. In both cases, the ultimate strength method was used to compute the capacities. Table 5-3 compares the available moment and shear capacities with the computed moments and shears.

5.3 Evaluation of Shear Walls and Columns Supporting the Pool Floor

5.3.1 Shear Walls

Compressive, shear and bending stresses due to additional rack

weight including their seismic effect on the shear walls supporting the pool floor were listed in Tables 4-3 and 4-4. The combined shear stresses for the critical load combinations are shown and compared to the allowable shear stresses in Table 5-4. The allowable shear stresses were computed using the ultimate strength method. Combined compressive stresses due to bending and axial compression for the critical load combinations are shown in Table 5-5. Allowable compressive stresses, computed using the working stress method of design, are also shown in this table.

5.3.2 Evaluations of Columns M_b - 10_b and M_b - 7_b

Load carrying capacities of Columns M_b - 10_b and M_b - 7_b were obtained from Reference 6. These capacities were computed for service load conditions using the working stress method. These are listed in Table 5-6 and compared to the computed loads shown in Table 4-5. Even though the computed loads include the effect of SSE on the additional rack weight, for simplicity and conservatism, these were compared with the capacities computed for service load conditions using working stress method.

5.4 Evaluation Results and Conclusions

Results of evaluation show that fuel pool structures are structurally adequate to withstand the additional loads that would result from the proposed modification in which the present racks will be replaced by the high density racks described in Reference 1. A comparison between the safe load carrying capacity of the structural components and the computed loads for various critical load combinations as shown in Tables 5-1 through 5-6 shows the calculated margin of safety. The actual margin of safety, in general, is more than depicted in these tables. The reasons are listed below:

a) Due to age-hardening effect, the actual strength of concrete is more than the design rated (28-day) strength. This difference can be as high as 20 percent.

b) Even though the floor load from the proposed high density racks was computed to be 2.0 ksf, the evaluation presented was performed using a rack load of 2.4 ksf.

c) The canal wall separating the pools from the transfer canal was evaluated assuming that it supports the pool floor in a deep beam action. Thus, the design loads for this beam-wall, as well as for Columns M_b - 10_b and M_b - 7_b which support it, are conservative.

d) For evaluating the pool floor, the canal wall was assumed to be resting on it. Hence, the design loads for which the pool floor was evaluated are conservative.

e) For Columns M_b - 10_b and M_b - 7_b, for simplicity, capacities computed for service loads by the working stress design method were compared to loads which included the effects of SSE. Since SSE loads are included only in factored load combinations for which higher allowables are permitted, the above comparison was conservative.

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TABLE 5-1

EVALUATION OF BENDING CAPACITY OF POOL FLOOR

Canal Condition	Critical Load Combination	Available Steel Area(in^2)	Required Steel Area(in^2)
Full	$1.4D + 1.4H$	7.62	2.46
	$1.4D + 1.4H + 1.9E$	7.62	3.38
	$D + H + T_a + E'$	7.62	5.1
Empty	$D + H + T_a + E'$	7.62	6.2

TABLE 5-2

EVALUATION OF SHEAR CAPACITY OF POOL FLOOR

Canal Condition	Critical Load Combination	Safe Shear Capacity (kip/ft)	Computed Shear Load (kip/ft.)
Full	1.4D + 1.4H	300.5	75.7
	1.4D + 1.4H + 1.9E	300.5	86.0
Empty	1.4D + 1.4H	273.1	68.6
	1.4D + 1.4H + 1.9E	195.3	78.3

TABLE 5-3

EVALUATION OF SPENT FUEL POOL CANAL WALL

Critical Load Combination	Load Type	Safe Capacity		Computed Load	
		At Critical Section	At Support	At Critical Section	At Support
1.4D + 1.4H + 1.9E	Moment (k-ft)	31700 (1)	-52,700	17200 (1)	-34117
	Shear (kip)	2653 (2)	6,471	2170 (2)	3226

NOTE: (1) At the center of span

(2) At the gate

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TABLE 5-4

EVALUATION OF SHEAR CAPACITIES FOR WALLS SUPPORTING THE POOL FLOOR

Shear Wall (1)	Computed Shear Stress (psi)		Allowable Shear Stress (psi) (2)
	1.4D + 1.4H + 1.9E	D + H + E'	
1	59.3	62.3	126.5
2	94.2	103.1	126.5
3	72.6	76.3	126.5
4	72.6	76.3	126.5
5	12.7	13.2	126.5
North Wall	40.7	44.7	126.5
South Wall	52.6	55.2	126.5

NOTES: (1) See Figure 2-2

(2) Contribution from steel reinforcement not included

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TABLE 5-5

EVALUATION OF COMPRESSIVE STRESSES

FOR WALLS SUPPORTING THE POOL FLOOR

Critical Load Combination	Shear Wall (1)	Computed Comp. Stress (Psi)	Allowable Comp. Stress (Psi) (2)
D + H + E'	1	723	1800
	2	723	1800
	5	238	1800
	South Wall	637	1800

NOTES: (1) See Figure 2-2

(2) Allowable Stress computed by Working Stress Method

TABLE 5-6

EVALUATION OF COLUMNS M_b - 10_b AND M_b - 7_b

Column No.	Computed Loads (Kip)[1]	Allowable Loads (Kips)[2]
M_b - 76	2195	2424
M_b - 10_b	3549	3636

NOTES: (1) Includes SSE Loads

(2) Allowable Loads are from Ref. 6 and are computed by Working Stress Method

6.0 REFERENCES

1. "Licensing Report for Prairie Island Nuclear Generating Plant, Units 1 and 2, Spent Fuel Storage Modification," Report No. QUAD-1-79- 509.

2. "Prairie Island Nuclear Generating Plant Earthquake Analysis," Revised February 16, 1971, John A. Blume & Associates.

3. "Seismic Design Classification," USNRC Regulatory Guide 1.29, Revision 2, February 1976.

4. "Other Category I Structures," USNRC Standard Review Plan, Section 3.8.4.

5. "Building Code Requirements for Reinforced Concrete (ACI-318-77), American Concrete Institute, 1977.

6. "Safety Evaluation Report for Auxiliary Building Spent Fuel Storage Pools and Their Supporting Structures Affected by Alterations in Fuel Storage Capability, Revision 2," by Flour Pioneer, Inc., Prairie Island Nuclear Generating Plant, Units 1 & 2, Project No. 21-7450 C.0001 (E75GA01), Activity No. 000-232, June 21, 1977.

7. "STARDYNE - A Static and Dynamic Structural Analysis Program," Mechanics Research Institute, Los Angeles, California.

8. "Introduction to Structural Dynamics," by J. M. Biggs, McGraw & Hill Book Company, 1964.

9. "Nuclear Reactors and Earthquakes," TID-7024, U.S. Atomic Energy Commission, August 1963.

10. American Concrete Institute Standard 307-69, 1969.

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11. Pioneer Service & Engineering Company, Project No. 216197, Drawing Nos: NF-39213E, 38303-8L, 38303-15K, 38303-1Q, and 38303-4K.

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EXHIBIT E

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

License Amendment Request dated January 31, 1980

Docket Nos. 50-282	License Nos. DPR-42
50-306	DPR-60

Exhibit E consists of the following:

Letter, L O Mayer (NSP) to Don K Davis (NRC), dated April 14, 1977.

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NSP

NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

April 14, 1977

Mr Don K Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors
U S Nuclear Regulatory Commission
Washington, DC 20555

Telecopied to the NRC
4/14/77

Dear Mr Davis:

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket No. 50-282 License No. DPR-42
50-306 DPR-60

Modification of Spent Fuel Storage Pool

On April 11, 1977 we were requested by your staff to provide a structural evaluation of the protective cover which is to be used during the installation of the modified spent fuel storage racks in the Prairie Island Spent Fuel Pool. The requested information is contained in the Attachment to this letter.

Full-sized drawings of the cover were provided to Mr M Grotenhuis earlier and, due to the size of the drawings, are not included with this submittal. If additional copies of these drawings are necessary, please contact us.

Yours very truly,

/s/ L O Mayer
L O Mayer, PE
Manager of Nuclear Support Services

LOM/LLT/ak

cc J G Keppler
G Charnoff
MPCA
Attn: J W Ferman

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STRUCTURAL EVALUATION
FOR
PROTECTIVE COVER OF AUXILIARY BUILDING
SPENT FUEL STORAGE POOL #1

I System Description:

In order to protect the small spent fuel pool (spent fuel pool #1) against the accidental drop of a spent fuel rack onto the stored spent fuel due to the proposed new spent fuel storage arrangement which provides the additional storage capacity, a protective cover is provided.

The cover is designed in three separate pieces for easy handling. Each piece weighs approximately 1.85 tons. The capacity of the gantry crane presently being used is 3 tons.

The cover is made of 3/16" stainless steel ASTM A167 Type 304 plate welded to a frame of built up wide-flange beams and structural tees using structural steel ASTM A588 Grade A (COR Ten B). The compressible material pads (silicone rubber SE-551 - AMS Specification 3335A) are used underneath the end supports to absorb a part of the energy generated from an accidentally dropped rack.

The cover is designed to clear the crane, which has a clearance of 5¼" (as measured in field) above the top of the pool walls.

The clearance between the cover and the crane is designed to be ½ inch.

The cover clears the temperature and water level indicator. However, the interference with light fixtures, fuel handling test racks and electrical receptacles requires relocation of these items.

II Loadings:

The weight and size of the spent fuel rack used in the design is as specified in our License Amendment Request dated November 24, 1976. No other external loads are assumed to act during the time the racks are being installed.

When transferring the rack over the small pool the height of the bottom of the rack in the non-tilted position is limited to a maximum of 6" above the top of the protective cover.

The spent fuel racks shall be put down on the protective cover at a vertical speed not exceeding 2 feet per minute. The racks can be unloaded any place on the cover, maintaining a minimum clearance of 5" between the nearest edge of the rack's leg and outside edge of the cover.

The following loading cases are investigated:

CASE 1 - Sudden application of a load due to manipulation of the rack in the non-tilted position (a normal operation case).

CASE 2 - Sudden application of load due to crane failure, dropping the rack from a height of 6" in the non-tilted position.

CASE 3 - Sudden application of load due to crane failure, dropping the rack in a titled position from a height of 3".

III Allowable stresses and design procedure:

Concrete: ACI 318-71 Building Code requirements for reinforced concrete.
Steel:

For Case 1 - Elastic design. Allowable stresses as per AISC, 1969, 7th Edition.

For Cases 2 & 3 Inelastic design. The procedure of Williamson and Alvy is used to determine the equivalent static loads caused by the impact of dropped racks. The allowable stresses are as per Part 2 of AISC, 1969.

Only the wide-flange sections are designed to carry the load due to an accidental drop of a rack; not the stainless steel plate. In the event the plate is punctured, following an accidental drop, the ductile nature of stainless will prevent the generation of secondary missiles.

IV Results of the design:

Protective Cover

The designed protective cover is shown on Prairie Island structural drawing NF-38303-29 (three copies of this drawing were provided separately). Two wide-flange sections W14x26 are used for each piece of cover supporting the inverted MT4x3.25 members (or alternate plates 5/16'x5") space at 6". The 3/16" stainless steel plate covers are approximately 6'-6" in width. The stresses and deflections under the investigated loadings are as follows:

For Loading Case #1:

Maximum bending stress in the wide-flange section is 23.3 KSI against an allowable of 33 KSI.

Maximum bending stress in the MT sections is 27.3 KSI (30 KSI for plates) against an allowable of 30 KSI.

Maximum shear stress in the wide-flange section is 3.5 KSI against an allowable of 20.0 KSI.

Maximum bending stress in the 3/16" cover plate is 14.3 KSI against an allowable of 24 KSI.

The maximum deflection is 3/4".

For Loading Case #2:

The ductility ratio 3.3 is obtained against an allowable of 10.

For Loading Case #3:

When the load is applied at the middle of the span:

a. Maximum deflection is 11.2".

b. Ductility ratio is 8.28 against an allowable of 10.

When the load is applied near the support, the whole energy is absorbed by the 1" thick compressible material pads.

a. The equivalent static load is 243 KIPS and deflection of compressible material pad is 13/16".

b. Bending stress is 48.72 KSI $< F_y = 50$ KSI

c. Shear Stress is 9.5 KSI $< F_{v_{all}} = 27.5$ KSI

d. Bearing on concrete $F_p = 3.4$ KSI $< F_{p_{all}} = 3.57$ KSI

e. The capacity of the weld is 400K against the shear force of 318.8 KIPS.

Lifting Ring

Each lifting ring is designed to carry twice the entire weight of the protective cover (2 x 3.8 = 7.6 KIPS) allowed for 100% impact. The shear stress in the ring so obtained is 6.8 KSI against 20 KSI allowable.

Existing Pool Structure

The existing spent fuel pool concrete structure was reviewed for the new load condition by reviewing Structural Drawings NF-38303-1 through 41, Auxiliary Building - Concrete Fuel Pool (latest revisions) and project files of the original calculations files. The accidental drop of a rack is assumed not to coincide with the other loading conditions which are controlling the supporting structure design, i.e., earthquake, accident temperature, tornado, etc. Hence, the supporting structures are adequate to carry the additional loads imposed by the system under consideration.