RELATED CORRESPONDENCE

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In The Matter Of	)	
	)	
DUKE POWER COMPANY	)	Dkt. No. 70-2623
	)	
(Amendment to Operating License SNM-1773 for Oconee Spent Fuel Transportation and Storage at McGuire Nuclear Station)	)	

DOCKETED USNRC JUN 1979 Office of the Secretary

AFFIDAVIT OF DIMITRI ROTOW

City of Washington)
) ss:
District of Columbia)

Dimitri Rotow, being first duly sworn, hereby deposes and says:

My name is Dimitri Rotow. I have been trained in physics and economics at Franklin and Marshall College and at Harvard University. I am employed as a consultant to the Natural Resources Defense Council, Inc. (NRDC), to assist NRDC in the preparation of a study on various aspects of nuclear waste management, under contract to the Department of Energy (DOE). In the preparation of this study, I have reviewed and studied a very wide range of literature dealing with all aspects of radioactive waste management. As basic research pursuant to the DOE contract, I interviewed numerous managers and utility administrators involved in spent nuclear fuel management at 19 different utilities. The results of this research were published by NRDC on March 26, 1979, under the title of "NRDC Findings on the Alleged Need for Acquisition

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or Construction of an Away From Reactor Spent Fuel Storage Facility" (attached as Exhibit 1).

Following the publication of this document, I met on several occasions with DOE administrators working at the highest levels of United States Government nuclear spent fuel planning, and thus gained a very detailed picture of the interaction between U.S. Government planning and private utility planning in the area of spent nuclear fuel management. I utilized this nowledge, gained from first-hand experience, in preparing another document on spent fuel management, "No Need for AFRs" (attached as Exhibit 2), that was issued by NRDC on May 1, 1979, and that was largely prepared as a factual presentation for use on the Congressional investigation of the need for away from reactor storage.

In 1977, the Carter Administration announced a policy of accepting spent nuclear fuel from private utilities for storage. This policy was intended to close the "back end" of the nuclear fuel utilization process, and to resolve the deep and far-reaching citizen concerns over perceptions that no disposal mechanism had been developed for safely managing radioactive wastes that would be dangerous for millennia. In 1977, it was announced that a keystone of the federal effort to show citizens it could "manage" spent fuel in the near term would be the construction and use of Away From Reactor storage facilities (AFRs), but this announcement was made without the benefit of analyses or studies investigating whether or not cheaper and safer waste management techniques, such as expanding the storage capacity of existing reactor spent fuel pools, would be preferable to the use of AFRs.

In late 1978 and throughout 1979, I have followed the DOE effort to convince Congress that an AFR is needed to manage spent fuel storage problems in the next decade. This effort has been based on reporting to Congress and to the citizens that, based on utilities' latest plans, if no AFR is constructed by the government many utility reactors will face a shutdown for lack of room in which to store their spent nuclear fuel. As can be seen from the DOE fact sheet included as Figure 1 of Exhibit 1, reactors owned and operated by the Duke Power Company at the Oconee and the McGuire sites figure prominently in the federal government's assertions. Although in the attached reports I did not investigate the situation at Duke, I did spend two days at the Duke Power Company headquarters going through all of Duke's documents on spent fuel management, and speaking to several Duke representatives about their plans.

Duke internal memoranda made available to me by Duke Power Company show that the decision to embark on a "cascade" plan was made in the middle of 1976, without analyses comparing the cascade plan to expansion of on-site capacity via the use of neutron absorbing racks. Although I went through all of the Duke documents pertaining to spent fuel management, I have yet to see a complete, comprehensive, and detailed analysis comparing transshipment to modern on-site fuel management techniques (such as the use of neutron absorbing racks, or of physical repacking of fuel rods). Duke's present effort to embark on an off-site spent fuel management plan on a step by step basis where on-site management options exist and where a whole cascade of transshipment is intended has serious and deep implications on a national level.

I now wish to make clear how the actions and announced plans of Duke Power Company significantly influence federal policy and federal actions in nationwide nuclear spent fuel management. First, I will comment on the federal promotion of an AFR.

The actual capacity cited as being required in a federal AFR has been reduced by a factor of three over the two years since the federal push for AFRs began. The capacity now cited as being required in an AFR is the very minimum figure capable of justifying the construction of an AFR facility. To defend this figure (approximately 500 mT spent fuel storage capacity) DOE is basing its figures on what it claims to be the maximum available storage capacity, based on current utility plans. If current utility plans indicate that on-site storage is unfeasible, this is highlighted as justifying AFR construction. If a utility tells DOE that expansion of on-site storage is unfeasible but that transshipment of fuel from one facility to another is a preferred plan, then the DOE contends that AFR need is doubly justified, both because on-site expansion is impossible, and because a utility claims it prefers to truck dangerous wastes about the country to managing it on-site.

This latter case prevails in the matter at hand.

I have read through the Duke Power documents on the proposed "cascade" plan. Two observations come immediately to mind: (1) The cascade plan, of transshipment from Oconee to McGuire, from McGuire to Catawba, and so on, reminds me of a perpetual motion machine whose motion depends on the circular logic of borrowing from Peter to pay Paul. This shuffling of fuel from one reactor to the next, and then to the next, and then to the next again obviously cannot continue forever. It merely postpones a final reckoning to a later day, at a time when the problem can only be worse.

Implicitly, the cascade depends on either a perpetual chain of newly-built reactors, on the construction and availability of a Government AFR, on the construction and operation of an internal Duke AFR, built at one of the reactors or on expansion of spent fuel capacity at each reactor by building a new pool or expanding an existing pool to hold the lifetime output of spent fuel from that reactor.

Duke has deliberately sought to conceal the fact that it has a cascade plan from DOE and Congress and to use the relatively limited relief provided by the proposed Oconee/McGuire transshipment to justify the need for an AFR. DOE has accepted these representations and uses the Oconee/McGuire situation as further evidence of the need for an AFR. By pointing to Duke's plan, the federal government establishes a precedent for ignoring safe and economical at-reactor storage, and establishes a precedent for "managing" radioactive waste by shipping it through the countryside in some inter-reactor nuclear waste shell game.

(2) The actual time when the decision to transship was made has been subject to some attention at Duke. Internal Duke memoranda reveal that the decision to transship was made in June 1976. In my reading of the documents, and in my discussions with Duke personnel, I get the unmistakable impression that the company is hewing to an inappropriate plan of transshipment simply because of inertial bureaucratic resistance to other options. This inertia gains respectability from the Department of Energy's focus on current utility plans in spent fuel management as evidence for AFR need, as opposed to a focus on what represents the best solution from consideration of the general public good. If a utility has calcified an inappropriate waste management technique as a result of organizational inertia, the Department of Energy is not influenced by the existence of better options: rather, they will seize and in fact have seized upon any examples of a transshipment or other off-site strategy being preferred to on-site expansion by a utility.

In short, the present scope of review of the Duke proposal to ship spent fuel from Oconee to McGuire, which focusses only on the first step in the cascade program, is being used by DOE to justify an AFR because under this scheme Duke will need some additional spent fuel storage before 1985. Were this proceeding to address the entire Duke cascade program and similarly broad alternatives to it, it would necessarily resolve much of the future course of Duke's spent fuel management scheme and thus limit the ability of DOE to capitalize on the piecemeal approach to spent fuel management.

All the above statements and true and correct to the best of my personal knowledge.

Dimitri Rotow

Signed and sworn to before me
this 21st day of May 1979.

Notary Public

My Commission Expires September 30, 1982

RELATED ... Exhibit 1

Natural Resources Defense Council, Inc.

917 15TH STREET, N.W.
WASHINGTON, D.C. 20005

202 737-5000

New York Office
122 EAST 42ND STREET
NEW YORK, N.Y. 10017
212 949-0049

Western Office
2345 YALE STREET
PALO ALTO, CALIF. 94306
415 327-1080

NRDC FINDINGS ON THE ALLEGED NEED FOR ACQUISITION OR CONSTRUCTION OF AN AWAY FROM REACTOR SPENT FUEL STORAGE FACILITY

by

Dimitri Rotow

Basic Research Funded Pursuant to
U.S. Department of Energy Contract
ER-78-C-01-6396

March 26, 1979

I. Introduction

On February 26, 1979, the Department of Energy sent to Congress proposed legislation to authorize DOE to construct or acquire facilities for the temporary storage of spent nuclear fuel (nuclear waste) away from the reactors which generated the spent fuel. This proposal will initially require a substantial commitment of federal financial resources, will require spent fuel to be shipped and handled twice as much as would occur if it were retained at the reactor site until a permanent repository was available, and will create the false impression that the ultimate disposal of nuclear wastes is a problem which is unrelated to the operation of nuclear reactors. Apparently aware of these shortcomings, DOE bases its proposal for authorization to build or acquire an away-from-reactor storage facility on the alleged immediate need for the facility. DOE attempts to support this allegation in an analysis recently issued as reference information. The analysis cites 22 nuclear reactors as being in danger of losing Full Core Reserve (FCR) capacity at the reactor spent fuel pool, and thus in immediate need of an away-from-reactor storage facility. The DOE-issued fact sheet summarizing DOE's analysis is attached as Figure 1 to this report.

The purpose of this report is to disclose the results of a survey conducted by the Natural Resources Defense Council[1] to test the accuracy of the DOE statements about the need for an away-from-reactor storage facility. Since NRDC is currently intervening in a case involving the Duke Power Company, the three listings for Duke reactors included in the 22 on the DOE fact sheet were not reviewed, to prevent any possible conflict of interest. Responsible fuel management personnel at all of the remaining 19 reactors were contacted by phone and interviewed in depth as to the precise spent fuel management situation at this reactors.

II. Summary of Findings

The NRDC survey discloses that, contrary to the DOE assertions, there is no need for a new away-from-reactor storage facility. In discussions with the spent fuel managers of the utilities, we learned that planned efficient utilization of the existing spent fuel storage capability at the reactors and existing off-site storage arrangements would enable them to continue to operate without restrictions due to a lack of spent fuel storage space until at least 1985. In addition, that date could be extended by 3 or 4 years for each facility if the utilities would discontinue their voluntary program of holding open space within the spent fuel pool to accommodate the immediate discharge of the full

1/ The basic cost of conducting the survey was paid pursuant to a contract between NRDC and DOE.

reactor core.[2] In short, there is no immediate need for any action to build or acquire an away-from-reactor storage facility.[3]

The main differences between the DOE assertions and the NRDC survey stem from DOE's failure to include current utility plans to expand at-reactor storage capability by implementing re-rack programs or other modern spent fuel management techniques at existing spent fuel pools. In some cases, DOE totally ignored applications on file with NRC for spent fuel storage expansion. In other cases, it ignored the potential for such expansion, potential which our conversations with utility spent fuel managers confirmed would be used if needed. This failure to account for utilities' latest plans, however,

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2/ Although DOE indicates retention of FCR capability is essential, in fact the Nuclear Regulatory Commission (NRC) has never required retention of such a reserve. Nuclear fuel is inherently safer in the shut down reactor with all its safety systems than it is in the spent fuel pool. If there are important safety reasons for retaining the ability to move the entire core out of the reactor, then there should be an NRC safety requirement. Until that happens, there is no basis for DOE to artificially create a premature shortage of spent fuel storage space by assuming FCR inviolability.

3/ In a separate analysis prepared by NRDC in 1978, we demonstrated that operating reactor sites have sufficient acreage to accommodate a facility for interim storage of the lifetime supply of spent fuel from the reactor. Less than one-fourth of an acre is required for such a facility. Analysis of Space Available for Storage of Spent Fuel at Existing Operating Reactor Sites, NRDC, July 1978. Thus reactors now under construction or planned should be including large spent fuel pools in their design and operating reactors should be seeking permission to build additional on-site storage capability. This would enable the utilities on a self-help on-site basis to provide spent fuel storage capacity to accommodate all possible permanent waste disposal contingencies, including the failure to establish a permanent disposal facility prior to the end of the useful life of the reactor.

does not account for all of the disparities between DOE and NRDC findings: in some cases (most notably the Susquehanna 1 reactor), NRDC's review of existing licensed capacity, even without consideration of expansion potential, leads to FCR retention well beyond the DOE-cited dates.

The net result of the NRDC survey is that:

1. All but one of the DOE listings were inaccurate.
2. Of the 19 reactors checked, five reactors are not operational. One of these (Humboldt Bay) has not operated since July 1976 and will not operate before 1980 at the earliest. The other four reactors are scheduled for start-up, if no delays occur, at times ranging from May 1979 (Diablo Canyon 1) to "sometime in 1981" (Susquehanna 1). The NRDC data presented assume these reactors will start up on schedule.
3. None of the reactors will be in need of an AFR before 1985, and most will be able to run into the 1990s without experiencing any such need.

Figure 1.

The Department of Energy Fact Sheet

ESTIMATES OF REACTORS REQUIRING AFR STORAGE

REACTOR	YEAR OF FIRST SHIPMENT TO AFR TO MAINTAIN FCR
OCONEE 1, 2, 3	1978 *
SAN ONOFRE	1978 **
HUMBOLDT BAY	1979
BRUNSWICK 1	1980
BRUNSWICK 2	1981
MCGUIRE 1	1981
NINE MILE POINT 1	1981
DIABLO CANYON 1	1982
ROBINSON 2	1982
DIABLO CANYON 2	1982
YANKEE-ROWE	1982
HATCH 1, 2	1983
PRAIRE ISLAND 1, 2	1983
ZIMMER 1	1984
OYSTER CREEK	1984
FORT CALHOUN	1984
RANCHO SECO	1984
ARKANSAS 1, 2	1985
MCGUIRE 2	1985
MAINE YANKEE	1985
SUSQUEHANNA 1	1985
TROJAN	1985

*CURRENTLY OPERATING WITHOUT FCR, PLAN TRANSSHIPMENT TO REGAIN FCR.

**SAN ONOFRE IS PRESENTLY SHIPPING TO GE AT MORRIS, ILL.

NOTE: THIS IS BASED ON MINIMUM PRUDENT PLANNING CASE. IT ASSUMES:
(a) UTILITIES CAN CARRY OUT LATEST EXPANSION PLANS;
(b) INTRA UTILITY SHIPMENTS ARE POSSIBLE; AND
(c) 70% CAPACITY FACTORS.

Explanatory Footnotes to Table 1.

1/ Assumes no shipment between reactors owned by the same utility.

2/ Lost FCR in 1974-75 but has approval to ship spent fuel to another reactor owned by the same utility.

3/ Reactor is not operational.

4/ DOE assumes shipments from one reactor site to another reactor site for several plants. If this assumption is added to the data from the NRDC Survey, the time period prior to loss of FCR is extended as shown.

Exhibit 2

Natural Resources Defense Council, Inc.

917 15TH STREET, N.W.
WASHINGTON, D.C. 20005
202 737-5000

Western Office
2345 YALE STREET
PALO ALTO, CALIF. 94306
415 327-1080

New York Office
122 EAST 42ND STREET
NEW YORK, N.Y. 10017
212 949-0049

May 1, 1979

NO NEED FOR AFR's

by

Dimitri Rotow

* * * * *

Recent DoE Studies Together With The Results of an NRDC Survey Indicate No Need for Away From Reactor Storage Facilities for Domestic Spent Fuel

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INTRODUCTION

In late February and early March of 1979, NRDC conducted a survey of responsible utility personnel to check the factual basis of a list of reactors which the Department of Energy (DoE) was circulating as evidence of near term Away From Reactor Storage Facility (AFR) need. The list contained 22 entries; however, since NRDC was engaged in a legal proceeding with Duke Power Company, the three entries in the DoE list referring to Duke reactor sites were not investigated. This narrowed the list of reactors to 19 entries.

Although NRDC has different policy viewpoints than DoE on issues such as Full Core Reserve (FCR) maintenance or the desirability of allowing transhipments of spent fuel between reactors owned by the same utility, we did not alter these assumptions in performing the study. Rather, we concentrated on checking the factual accuracy of the DoE assertions that a given reactor would lose FCR by the given date if the utility were able to carry out its latest expansion plans, and if transhipments between reactors owned by the same utility were allowed. This was purported to be the factual basis of the DoE listing, and it was this supposed factual basis which we investigated. This report discusses the results of that investigation and draws conclusions on the need for AFR's.

We conducted the investigation by calling responsible, knowledgable personnel at each of the 19 listings. These people were interviewed over the telephone using a standard questionnaire format (Exhibit A). In some cases, more than one person at each utility was interviewed, in order to make sure that our picture of what the utility had in mind _viz a viz_ spent fuel management was clear and accurate. NRDC did no manipulations of the data received from the utilities. The report issued on this subject was published not as an analysis but as simple reporting on what the utilities said were the facts at their reactors.

An early draft report summarizing the results of the survey (titled "NRDC Survey of Utility Spent Fuel Managers") was issued on March 12, 1979. This draft was circulated for comment to several parties (including the Department of Energy). A final version of the report was issued March 26, 1979 under the title "NRDC Findings on the Alleged Need for Acquisition or Construction of an Away From Reactor Spent Fuel Storage Facility."

NRDC had no knowledge of the general DoE report on this matter (DOE/ET-0075) at the time of the survey; thus, the survey and published NRDC findings did not comment on the numerical manipulations included in the DoE report. The survey simply checked the factual accuracy of the DoE fact sheet listing 22 reactors[1/] in immediate need of an AFR. NRDC survey findings strongly contradicted DoE assertions on the need for AFR's. These findings were criticized in a March 30, 1979 DoE report titled "Analysis of Near Term Reactor Storage Problems." We now turn to a point-by-point discussion of the actual need, if

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1/ The actual DoE-circulated lists of reactors needing an AFR have varied somewhat. The most prevalent list has 22 entries, some of which cover more than one reactor per entry (e.g.: "Arkansas 1,2"). In other cases, two reactors at the same geographic site are listed in two separate entries (e.g.: Brunswick 1 and Brunswick 2 each have their own entry). Thus, the 22 listings refer to 27 reactors located at 19 different geographic sites.

any, for AFR's which develops when the NRDC findings and the DoE commentaries are considered together.

The Basis for Differences Between NRDC and DoE Findings

The DoE March 30 analysis offers "four basic reasons for the differences" between the DoE data and the results of the NRDC survey.

The first reason offered by the analysis is that in some cases utility plans have changed since December 31, 1978, (the currency date of the original report); it gives the Oyster Creek plant as an example. Later on in the analysis, two other plants (Susquehanna I and Nine Mile Point I) are cited as listings where the utilities now have firm plans to expand on site which they did not have by December 31, 1978. In fact, this reason does not explain basic differences. Documents on file at NRC's Public Document Room show that both of these utilities had plans to expand well before December of 1978, with the Nine Mile Point I utility going to the point of bombarding NRC throughout the Fall of 1978 with copious reports on its desire to install "Boraflex" racks at Nine Mile Point I. We have attached copies of these documents.

The second reason for difference offered by DoE is similarly in error: "In some cases (e.g., Robinson, San Onofre), reactors lose FCR for one or two years but then regain it when another storage basin at a new reactor becomes available. The DoE analysis included these near term problems. That of NRDC apparently does not."

In recognition of precisely this effect, both the draft and the final reports on the NRDC survey set out the loss-of-FCR dates supplied by utility managers with a breakdown into dates prevailing if both transhipment and/or on-site expansion are possible, and if only on-site expansion is allowed. In the case of San Onofre, there is no dependence on availability of another storage basin, because on-site expansion easily manages fuel storage needs through 1991. In the case of Robinson, which had lost FCP before 1975, the NRDC survey reported what we were told by responsible utility personnel (including the chief engineer at Brusnwick, which jointly manages spent fuel with Robinson, and the Manager of the utility's Nuclear Fuel Department): if no transhipments occur, Robinson will remain out of FCR while the Brunswick plants can maintain FCR through 1987-88. On the other hand, if transhipments occur, they can run all three plants with no more than one running at any one time without FCR (which is the situation prevailing since 1975) until the Harris facility comes on line in 1983, at which point all three reactors can run with FCR through 1992.

We have already addressed DoE's third reason for why their report differs from the results of our survey; this reason says that their conclusions differ from our survey because of "differing assumptions". Supposedly, the DoE report was based on utility plans and that of NRDC was based on what NRDC considered to be physically possible and reasonable. As was made clear earlier, NRDC made no assumptions not made by DoE, but simply called the utilities to see what they realistically think they can accomplish; since DoE's fact sheet was issued under the premise that utilities could carry out their latest plans, in the single case where the utility felt that expansion was technically possible but not likely to occur due to political constraints (Rancho Seco), we listed the utility under the FCR date they told us was technically feasible. The case of Rancho Seco seems open to honest misinterpretation; other listings supplied by Doe are not.

DoE's final reason for differences in FCR-loss dates is the presence of "internal inconsistencies" in NRDC's report and in DoE's data. There are no internal inconsistencies in NRDC's report. It is true that we criticized DoE for using five reactors which are not operational to make a case for AFR's; somehow, using non-operational reactors to make a case for immediate needs without mentioning that the reactors are not operational seemed to us to be poor analytic form. Nonetheless it was the premise used by DoE, so when we checked the DoE data against utility comments, the same premise was used. Similarly, since the DoE data was unconcerned about the effects of various possible delays, when we checked the DoE data we asked the utility personnel interviewed for comments on that basis.

In summary, we believe that DoE's reationale for the difference between NRDC's and DoE's findings misses the mark. Clearly, the most important difference is that DoE's findings are based on "current" planning". If expansion of pools is possible and reasonable, but the utility simply has made no commitment to do so, this capacity would not show up in DoE's analysis. NRDC, on the other hand, asked the utilities what they realistically thought they could accomplish, including what they would do if the Government AFR were not available. Approached in this way, one gets quite different results. There are obvious differences between utility plans based on the assumption that a Government AFR is forthcoming, and what a utility would do if no Government AFR were available.

Case-by-Case Analysis

We will now examine the DoE assertions and admissions on a reactor-by-reactor basis. Doing so makes clear the errors which occurred in the original documents (and in the subsequent responses, for example, the March 30 analysis) apparent; however, the point of the following discussion is to show exactly why there is no case for an AFR, and not simply to question the appropriateness of the original research and the follow-up DoE responses. We think the issue of alleged AFR need takes precedence over DoE fumbling in trying to make their case. What is clear from a point-by-point review is that in the process of attempting to explain earlier lapses, DoE ends up presenting evidence that bolsters NRDC's findings.

Each of the following sections opens with a quotation from the March 30 DoE analysis. Some sections also contain the original DoE citation from DOE/ET-0075. This is followed by a dis-

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cussion of NRDC findings and overall implications on alleged AFR need.

Yankee Rowe (DOE-1982) (NRDC-1995)

> "Yankee Rowe has a maximum expansion capability which involves double-tiered storage of fuel. This could significantly increase basin capacity. However, the double-tiered storage is a new concept and is yet to be licensed, and this was not used as the base case planned basin capacity for Yankee Rowe. In fact, it should have been and therefore the date used by NRDC would be correct."

On this reactor, DoE admits making a mistake; however to understand how the mistake came to be made it is instructive to see what the citation for Yankee Rowe was before the NRDC investigation: "Yankee Rowe has a capacity of 391 fuel assemblies. This capacity is exhausted when maintaining full core reserve and shipments to AFR storage begin in 1982. Basin is near maximum capacity and the marginal cost of an additional small increment of storage does not appear to be justified."

This citation obviously refers to some sort of economic study done on Yankee Rowe. It acknowledges expansion as being possible, but cites "the marginal cost of an additional small increment of storage" as making the expansion not worthwhile. How could have any economic analysis detailed enough to make marginal cost breakdowns miss the true situation at Yankee Rowe, when a single phone call from NRDC revealed the true situation? If such a study was not done, what justifies the reference to "marginal cost"?

Although in its latest reference to Yankee Rowe DoE admits to making a mistake, the admission casts light on DoE's methodology: the error admitted to is an error of not using the correct "base case planned basin capacity," (whatever that obfuscation means) and not the error of using unreal assumptions about what utilities can do to justify AFR need instead of paying closer attention to utility plans which invalidate that need.

Oyster Creek (DOE-1984) (NRDC-2000)

> "Oyster Creek's current rerack plans are identical to those used by DOE. A further expansion to higher density racks may be possible, but is not currently planned. DOE assumed that, unless a utility specifically described plans to send fuel of one type to a reactor of a different type, shipment would not occur. Since in December 1978 the utility did not have such plans, no shipments between Oyster Creek (BWR) and Forked River (PWR) were assumed. Recent discussions with the utility, however, indicate that Oyster Creek now has firm plans to ship to Forked River."

The original DoE citation (quoted as being correct in the first sentence of the above) is: "This utility has plans to expand this unit to 1796 fuel assemblies in steps between now and 1984. They are basically expanding to maintain full core reserve and according to this analysis first shipments to AFR storage will be required in 1984."

Of course, the word "required" means there is no alternative. The original citation missed two alternatives considered by the utility: compaction (or "pin-packing"), and transhipment to Forked River. As is made clear by the citation for this plant issued by DoE after NRDC's survey, DoE built the case for AFR need for this reactor by "assuming" away alternatives. Why did DoE "assume" no shipments to Forked River could occur, instead of taking the time to call the utility to see if it really had a need for an AFR? Further, the DoE analysis does not make clear that the NRDC dates are correct: onsite expansion of spent fuel capacity through densification would allow Oyster Creek to maintain FCR through 1987-88, and shipments to Forked River would maintain FCR through the year 2000; in any event, Oyster Creek has no immediate need for an AFR.

Nine Mile Point 1 (DOE-1981) (NRDC-1991)

> "The DOE data shows that Nine Mile POint would lose FCR in 1981, if no further action were taken. However, the utility now has definite plans for such reracking which it did not have when the DOE data were collected in December"

The original citation, ostensibly current as of December 31, reads:

"Nine Mile Point-1 has a capacity of 1520 fuel assemblies according to Niagara Mohawk's current plans. According to this analysis a shipment to an AFR would be required in 1981. Subsequent to that Nine Mile 2 becomes available and until 1990 shipments take place to this unit. There are possible seismic problems with further pool expansion."

As the attached documents show, Nine Mile Point 1 was actively involved in increasing its spent fuel storage capacity since March 22, 1978. Thus, the contention that the utility changed its plans after December of 1978 is untrue. Further, the original

DoE citation also appears deceptive, since nowhere in the Nuclear Regulatory Commission's ample documentation on the Nine Mile Point I proposed expansion are there expressions of "possible seismic problems." This reactor, as of the Spring of 1978, posed no evidence at all of AFR need.

Susquehanna (DOE-1985) (NRDC-1994)

> "Susquehanna, which is a new reactor that will come on line in 1981, has recently made firm plans to purchase high density racks, which would provide adequate storage until 1994."

Originally:

"Susquehanna 1, 2 have mutually accessible fuel pools of 1015 fuel assemblies capacity each. First shipments are required in 1985 to AFR storage for this case. The utility has plans to expand by approximately 50%, but has no presently defined firm schedule."

Of course, the original list of 22 reactors was circulated on the basis that the dates cited for AFR needs by DoE represented the utilities' most recent plans. As can be seen by the attached letter from Harold Denton, the Director of NRC's Office of Nuclear Reactor Regulation to Mrs. Stanley Shortz, dated December 13, 1978, that by this time Susquehanna plans for spent fuel pool expansion had progressed to the point that a senior NRC official was aware of them and used the phrase "when a request is received for amendment of the construction permits" (as opposed to "<u>if</u> a request . . ."). It appears from DoE's comment that the utility had plans to expand by 50% that DoE was also aware of the utility's plans to expand, and thus it was quite inaccurate for DoE to use

Susquehanna 1 and 2 as evidence of AFR need. These reactors have no need at all for an AFR before 1994.

Maine Yankee (DOE-1985) (NRDC-1992)

> "Maine Yankee current plans are the same as DOE presented in its data. FCR would be lost by 1985. There are no firm plans to expand beyond this level."

The plans presented by DoE in the earlier data were: "Maine Yankee has a fuel pool capacity of 943 assemblies. There are no plans to expand it further. First shipments to AFR storage are required in 1985. Main [sic] Yankee can get only an additional 185 spaces by going to more closely spaced racks. Their marginal cost for this modification would cause them to consider other alternatives, such as a separate at reactor storage pool, before re-racking for such a small gain in storage capacility."

Attached to this report are copies of documents in NRC's Public Document Room showing that by November 22, 1978, Maine Yankee had an enthusiastic proposal in at NRC to expand its spent fuel storage capacity via compaction. The report forwarded on 22nd indicates a lot of thought and research spent by the utility on this issue, so "planning" must have occurred well before November.

It is true that a simple rerack would buy Maine Yankee only 185 spaces; however, the enthusiastic report sent in to NRC by Maine Yankee proposes not a simple rerack, but a management technique known as compaction, or "pin packing". The same Maine

Yankee report contrasts the simple rerack with the great and economic gains from compaction. There is no need whatsoever for an AFR for Maine Yankee, although one was falsely presented.

Trojan "(DOE-1985) (NRDC-1980)"

This date citation quotes NRDC's original draft loss-of-FCR date for Trojan. The Final NRDC report on this matter credits DoE with citing the date on the Trojan reactor correctly; however, a single reactor that loses FCR in 1985 and can continue to run without FCR for an additional three years (to 1988) hardly makes the case for a near-term AFR.

Robinson 2 (DOE-1982) (NRDC-1992)

> "Carolina Power and Light has indicated that the costs associated with making the necessary structural modifications to Robinson 2 to enable it to expand further are extreme. It would require shutting the plant down and purchasing replacement power while equipment was relocated from under the fuel pool floor to enable supports to be placed to carry the load of a more densely packed fuel pool. Robinson is currently shipping to the Brunswick fuel pools and this will no longer be possible after 1981. Robinson would need to ship to an AFR in 1982 or 1983. Once Harris comes on line in 1984, Robinson 2 would ship there. This capability would be exhausted in about 1991, which comparable [sic] to NRDC date of 1992."

The original DoE citation reads: "Robinson - Carolina Power and Light. Carolina Power and Light has indicated that the costs associated with making the necessary structural modifications to to Robinson 1 to enable it to expand further are prohibitive. It would require shutting the plant down and purchasing replacement power while equipment was relocated from under the fuel pool floor to enable supports to be placed to carry the load of a more densely packed fuel pool. Robinson is currently shipping to the Brunswick fuel pools as previously discussed."

In the discussion of Brunswick mentioned above, there is no indication that Brunswick can be expanded, that Robinson lost FCR and has been operating without it since 1975, or that after 1983 shipments to Harris could ease the fuel problems at Robinson. This is finally acknowledged in the March 30 analysis, but that analysis does not make it clear that in no way could an AFR solve the fuel problems which have been bedeviling Robinson for years. If transhipment is allowed, as was assumed by DoE, Robinson, currently in bad shape, will remain in bad shape until 1983, at which time it will regain FCR until 1992 by shipping to Harris.

<u>Fort Calhoun</u> (DOE-1984) (NRDC- Beyond 1985)

> "The Fort Calhoun has recently changed from an annual reload cycle to a slightly longer cycle. Hence, the fuel discharge which results in loss of FCR slips from 1984 to 1985. Further basin expansion may be theoretically possible, but plans to do so are very preliminary. Hence, reactor basin storage capacities used by DOE remain correct, but the FCR date should be changed to 1985."

Originally,

"Fort Calhoun has a fuel pool of 483 assemblies. The pool is expandable, but there would be a problem working around fuel in residence."

Of course, the list of reactors cited by DoE in testimony before Congress as posing evidence of AFR need was claimed to be based on the assumption that utilities would be able to carry out their "latest plans." In trying to make this alleged AFR need materialize, DoE now resorts to subdividing "plans" into "very preliminary plans", "firm plans", "definite plans", and "specifically described plans." All of these subvarietal references to utility plans occur in the explanatory notes in the March 30 DoE analysis. Considering the basic problems with justifying AFR construction, it is easy to understand the need for so many equivocations. When NRDC questioned utility personnel, we simply asked them what they are doing right now, and what they will do if no government AFR is forthcoming. If a utility executive told us simply and directly that, barring a government AFR, they would expand their on-site storage, we viewed that as planning. This was the case at Fort Calhoun, where the Reactor Engineer told our interviewer that the utility feels that something from the government will not be forthcoming in the required time frame, and so they are making plans independent of the government. In reality, Fort Calhoun does not require an AFR.

<u>Arkansas 1, 2</u> (DOE-1985) (NRDC-1986/7)

> "The basin capacities used By DOE are consistent with the utility's current plans. There may be expansion possible to delay the loss of FCR a few years. However, there are no firm plans to do this at this time."

Again, DoE equivocates in discussing the reality of AFR need. Our intent is not to review DoE error, and it makes little difference whether or not the right number for basin capacity was used in the numerical manipulations in DoE's February report. What is important, is that the Arkansas 1 and 2 reactors offer no support for DoE's suggested AFR legislation. Our 1986/87 figure was based on the minimum expansion of existing capacity allowing FCR to be maintained that was known to be possible and counted on by the utility. The case for an AFR looks even weaker, though, if the utility's contingency plans are included: they will run without FCR if they lose it (delaying any conceivable AFR need into the 1990's), and they have already secured bids for new onsite spent fuel storage, which would extend FCR maintainability for the life of the Arkansas 1, 2 reactors. Clearly, the bottom line is that the Arkansas 1, 2 reactors do not require an AFR.

Zimmer (DOE-1984) (NRDC-1992)

> "The current utility plans are identical with DOE figures. The utility says that further expansion is theoretically possible comparable to the DOE maximum expansion case. However, there are no plans to do that at this time."

In DoE's February report, this citation read: "The current capacity of the Zimmer-1 fuel pool is 1120 assemblies. The utility could approximately double this capacity, but no schedule is presently defined for doing this."

NRDC's interviewer spoke with Dr. Chitkara, the head of the utility's fuel management group, on March 3, 1979. Dr. Chitkara confidently stated that if the government did not provide an AFR, the utility would rerack at Zimmer 1 to run with FCR until 1992. In fact, Dr. Chitkara was so confident about reracking at Zimmer 1 that when asked the standard question of whether the utility would run the reactor without FCR, he did not understand why he was being asked the question. He said he didn't feel the question was applicable to Zimmer 1, since the company feels "very confident" that they could rerack and thus not have to worry about FCR well into the 1990's. Despite DoE's equivocations about "theoretically possible", "further expansions" and "presently defined" schedules, the simple reality is that the Zimmer 1 reactor is in no need of an AFR.

<u>Prairie Island 1, 2</u> (DOE-1983] (NRDC-1992]

> "DOE information on basin capacity is correct. It involves the utility making some modifications which it is planning to do. There are no further expansion plans and the utility felt the political problems associated further reracking [sic] on top of the two previous reracking activities may make this option impossible. Also the Tyrone reactor startup date of 1986 is very uncertain, and may slip. In addition, the utility said it will not ship Prairie Island fuel to Tyrone for storage. The NRDC date of 1995 [sic] for lost of FCR [sic] is totally inconsistent with the data from the utility."

The NRDC data had nothing to do with possible shipments to a reactor at Tyrone. We were told by Mr. Watzl, the plant superintendent at Prairie Island, that the utility is going out for bids on another rerack, using neutron absorbing racks. They expect to use Boral sandwich material, and plan the modification to be done in 1982. This new rerack will allow them to maintain FCR through 1992. When asked if the utility would run without FCR, Mr. Watzl said they "don't really think about it" because they do not perceive themselves as having spent fuel troubles. Further, Mr. Watzl said that if the utility does not have the services of a government AFR in 1992, they will build another spent fuel pool, since they believe that would be cheaper than to base further expansions on the existing fuel pool. Although Mr. Watzl exuded confidence about reracks and never mentioned "political problems", as will be discussed in detail in the next entry, even if there were local political problems, an AFR should not be used to bypass citizen dissent.

<u>Rancho Seco</u> (DOE-1984) (NRDC-1992)

> "Rancho Seco's current plans are as stated by DOE. While there may be further expansion possible, the utility was particularly emphatic that there were no plans to do this. The utility felt that the NRDC date of 1992 was unfounded."

Of course, this citation leaves open the question of whether NRDC's date of 1992 is correct in indicating no AFR need. It should be clear that what a utility does, or even plans to do, depends heavily on what it thinks the government will do. If a utility believes that the federal government will provide a cheap AFR facility to remove all "back end" fuel cycle concerns from the utility, then of course the utility will not embark on any commitments to handling those concerns itself. (Although it will cost the utility to use an AFR if one is provided, since the government will soak up the entire capital cost of the project, having the option of using an AFR costs the utility nothing.) A utility may well have contingency plans on what it will do if no AFR is forthcoming, but so long as the option of using an AFR is available for free, the utility will try to have that option maintained. Simple common sense demands that the utility supports legislation giving it this free, yet potentially valuable, option.

The main concern at the present time is whether or not an AFR is really needed, and if it is, precisely which reactors require an AFR. DoE has confirmed to NRDC that in the follow-response to NRDC's report which resulted in the March 30 DoE analysis, at no point did the DoE interviewer ask the utility what they would do if no government AFR were available. He asked them many questions regarding assumptions made by DoE in the numerical manipulations in the February report, but he stayed clear of asking questions that confronted the issue of AFR need head on. Further, recent evidence obtained by NRDC indicates that the main purpose of DoE's follow-up study was to warn utilities not to cooperate with outside efforts to obtain accurate information. It is in this context that we review the Rancho Seco entry.

Our interviewer spoke with Mr. Dan Whitney, the Senior Nuclear Engineer at Rancho Seco's controlling utility, the Sacramento Municipal Utility District, on March 6, 1979. Mr. Whitney said that they could "at least double" onsite capacity by reracking using neutron absorbing racks, which would allow them to maintain FCR into the 1990's. He did express strong concern that they would not get a license for such a rerack by local intervenor groups. He went on to say that they were "counting on" government action by 1985 to solve their troubles. In reporting this entry, NRDC used the 1992 date because we wanted to display the data in a format directly comparable to that used by DoE. Since DoE assumed that the utility could realize its latest plans and in no way attempted to incorpc .:e the effects of the many interventions now ongoing, neither did we. Mr. Whitney s_emed absolutely clear that if they could get a rerack by local intervenor groups they would in fact rerack. He seemed equally clear that government action by 1985 would be a simple solution to a political problem and not to a technical problem.

In this light, although views of what is the best interpretation of "latest plans" in this case are subject to honest differences, the simple reality is that Rancho Seco offers no evidence of AFR need. As we understand the matter, the Administration has suggested an AFR not as a politically-expedient means of bypassing citizen dissent over nuclear power, but as a device to manage technical problems. There are no technical spent fuel problems at Rancho Seco, but there apparently is considerable citizen concern. If DoE wants to use citizens' concerns in the Rancho Seco case to justify AFR storage, it should be consistent and take heed of citizen concerns throughout the country which may well result in the shutdown of most of the reactors on DoE's "near

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term AFR need" list, thus eliminating them all as contenders for near-term AFR need. It is technically easy and economically cheap to store Rancho Seco's spent fuel onsite; thus, this reactor does not require an AFR for storing its spent fuel.

Hatch 1, 2 (DOE-1983) (NRDC-1995)

> "The basic capacities used by DOE were correct. The utility has indicated there are no plans to ship to Bogtle when it comes on line. Hence, the DOE assumption of no transhipment between Hatch (BWR) and Vogtle (PWR) remains valid."

This entry is a confusing surprise to NRDC: maintainance of FCR through 1995 at Hatch has nothing to do with transhipment to Vogtle: NRDC was told that Georgia Power Company will maintain FCR through this date without a government AFR by reracking onsite at Hatch. Whether or not the DoE assumption about transhipment is valid has nothing whatsoever to do with the correct loss-of-FCR date at Hatch. If DoE thinks Hatch offers evidence of AFR need, why did they not provide it? The NRDC date is correct; there is no need at Hatch for an AFR.

Diablo Canyon 1, 2 (DOE-1982) (NRDC-1988/89)

> "Again the basic capacities used by DOE were correct and the loss of FCR dates used by DOE were appropriate. The utility felt the NRDC dates were unfounded."

The issue is, of course, not whether a date is "appropriate", because "appropriate" connotes suitability for a purpose: the issue is whether or not the original dates supplied by DoE were correct, under the representations made about them, and thus, whether an AFR is required. On March 5, 1979, Mr. Terry Rapp, the Senior Power Production Engineer at Diablo Canyon, told NRDC's interviewer that Pacific Gas and Electric Company would rerack at Diablo Canyon 1&2 to deal with its spent fuel problems. As the interviewer noted at the time of the survey:

> "They [Pacific Gas & Electric] don't know how far they can extend capacity, but Rapp confidently stated it would be "substantial", with at least five discharges after the first one; this would extend FCR maintainability at least into 1988 for DC1, and into 1989 for DC2."

Rapp also told the interviewer that the utility's target was to have the greater capacity licensed and installed at the time of the first discharge (the Diablo Canyon reactors are not yet operational). Thus, the Diablo Canyon plants do not support DoE's legislative push for an AFR.

<u>Brunswick 1, 2</u> (DOE-1980/81) (NRDC- 1992)

> "The Brunswick spent fuel pools have been expanded to their maximum. The utility confirmed that Brunswick would lose FCR in about 1981. Until the Harris reactor comes on line (planned for 1984), Brunswick would have a storage shortfall. Clearly any delays in the startup of Harris exacerbate the situation."

What the DoE does not say concerning this entry, is that a shortfall occurs at Brunswick only if one assumes that the Robinson 2 reactor (owned by the same utility) regains FCR (which it lost in 1975) by massive dumping of its spent fuel at the Brunswick reactors' pools. If transhipment does not occur, the Brunswick reactors can maintain FCR with their current capacity through the dates given by NRDC. If transhipment does occur, spent fuel can be managed so that at any one time in the three reactor system, (Brunswick 1 & 2, and Robinson 2), there is only one reactor operating without FCR, as is presently the case. There is no disguising that if retention of an FCR is important the Robinson 2 reactor does not have it and has not had it since 1974-75. It __is__ possible to manage the spent fuel situation among these three reactors so as to deliberately cause a mess at all three, but this is not what the utility would do if no AFR were handed to it. In any event, no 1983 government AFR could possibly solve the root of the problem: the congestion at Robinson 2. As the utility personnel told NRDC, the problem can be managed in the current state until the Harris plant comes on line in 1984, at which time there will be ample storage space for all three reactors. At no point in their "analysis" did DoE contest this fundamental point, which denies any Brunswick need for an AFR.

__San Onofre__ (DOE-1978) (NRDC-1991)

> "The DOE data indicated that San Onofre would lose FCR now if some action were not taken. San Onofre has begun shipments to GE Morris. This was footnoted on the DOE table of reactors with near-term problems. It is recognized that the continued transfer of fuel to GE Morris will result in postponing the loss of FCR date for some years."

Again, although the DoE recognizes that San Onofre will have no immediate FCR problems because of their contracts with Morris, the original citation of 1978 has not been amended other than to say that the loss-of-FCR date will be postponed "for some years." DoE makes no comment on the NRDC survey findings, which showed that San Onofre 1 has many options open in dealing with spent fuel. The 1991 date occurs with simple on-site reracks at San

Onofre 1, but the utility personnel interviewed also made a point of citing their ability to utilize the new San Onofre 2 and 3 facilities, or to continue the contract with Morris. Any of these options disallows a need for a government AFR.

Humboldt Bay "(DOE-1979) (NRDC-1986"

First it should be noted that this reactor has been shut down since July 1976 and one can seriously question whether this reactor will ever start operating again. DoE assumes the reactor will start up in 1979 and will require a shipment to an AFR in 1979 to maintain FCR. As the reactor has one year's worth of running time available with FCR on the current pool capacity, and it expected not to start operating before 1980, it is unclear where DoE gets its figures, and why they offer them. Based on our own utility survey, we have assumed the reactor starts up in 1980. It can run for a year without loss of FCR. During this period it can expand capacity (rerack) to meet its needs (with FCR) until 1986.

As an aside, DoE says, "the utility confirmed the basin capacity DOE is using as correct. In addition, there are no plans to ship to Diablo Canyon, as NRDC seemed to assume, which is a PWR." Of course, whether or not the current basin capacity was correctly plugged into DoE's model is not the issue: the essence of the need for AFR's is the utility's plans and ability to expand or better that current basin capacity. Further, well before March 30 (when the analysis was written), DoE possessed a complete copy of NRDC's "raw" survey data, and was fully informed that the increase in FCR maintainability depended, as we were told by the utility, on expanding on-site storage at Humboldt Bay, not on shipping to Diablo Canyon. DoE's views on what NRDC "seemed" to assume are completely misleading, and the analysis supplied fails to contradict NRDC's finding of no AFR need.

Summary

In summary, therefore, out of 19 listings, the Department of Energy admitted that NRDC was right in refuting alleged AFR need on four listings, and that the original DoE report is wrong. In a fifth listing (Maine Yankee) formal NRC licensing documents prove no need for an AFR. In five more listings, DoE admi.s expansion is possible, but utilitizes a self-serving definition of "planning" which excludes from consideration any plans that do not depend on the government providing an AFR. As we have seen, by DoE's own admission, these five reactors, regardless of what definition of "planning" is used, do not offer evidence for a near term AFR. In seven more listings, DoE offers no evidence for AFR need and refuses to reply to NRDC survey findings. Finally, in one reactor, Robinson, the DoE documents quibble with minor points but admit that NRDC's findings are correct. In the last case, Trojan, NRDC and DoE are in agreement: this reactor will lose FCR in 1985, and there may be no realistic measures which can be taken to run this reactor with onsite storage alone beyond 1988 (as a theoretical point, even this reactor can re reracked t xtend FCR, but there are technical problems that appear to make this very difficult). Thus, ou of 19 listings, DoE cited the facts at only one correctly. In several cases, the DoE follow-up study

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belatedly supported NRDC's findings of no need for AFR's. In all cases, no evidence at all has been found in support of AFR's.

As a final commentary, it should be noted that the economic and technical uncertainties which now surround nuclear waste management are legitimate uncertainties which flow directly from the anomaly created by beginning a nuclear power program without having any reasonable notion of how to solve the most serious and irreversible problem created by nuclear reactors -- nuclear waste. It is fair and essential that decisions on whether to build more nuclear plants and whether to continue to build and operate those to which commitments have already been made should be influenced by the real uncertainties created by this anomalous situation. For the government to step forward, as DOE proposes, and offer to take title to and store spent fuel in government-owned interim storage facilities in order to artificially establish for a utility a solution to the mounting volume of nuclear wastes for which in fact no disposal solution exists, and to set a one-time fixed fee for waste disposal in order to artificially establish a fixed cost for waste disposal when no fixed cost can in fact be established, is the worst kind of government subsidy. Nuclear power is rightly burdened by the waste problem and the absence of any solution to it. If its benefits are not sufficient to offset that burden, then it deserves to be halted. The time has come to face up to that choice, not to avoid it as DOE proposes.

RELATED CORRESPONDENCE

Attachment for

Nine Mile Point 1

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

October 27, 1978

Docket No. 50-220

Mr. Donald P. Dise
Vice President - Engineering
Niagara Mohawk Power Corporation
300 Erie Boulevard West
Syracuse, New York 13202

Dear Mr. Dise:

The Commission has requested the Office of the Federal Register to publish the enclosed "Notice of Proposed Issuance of Amendment to Facility Operating License" for the Nine Mile Point Nuclear Station, Unit No. 1, in response to your request of March 22, 1978.

The amendment would revise the Technical Specifications to increase the spent fuel storage pool capacity from 1984 fuel assemblies to 3009 fuel assemblies.

Sincerely,

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosure:
Notice

cc w/enclosure:
See next page

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cc: Eugene B. Thomas, Jr., Esquire
LeBoeuf, Lamb, Leiby & MacRae
1757 N Street, N. W.
Washington, D. C. 20036

Anthony Z. Roisman
Natural Resources Defense Council
917 15th Street, N. W.
Washington, D. C. 20005

T. K. DeBoer, Director
Technological Development Programs
State of New York
Energy Office
Swan Street Building
CORE 1 - Second Floor
Empire State Plaza
Albany, New York 12223

Mr. Robert P. Jones, Supervisor
Town of Scriba
R. D. #4
Oswego, New York 13126

Niagara Mohawk Power Corporation
ATTN: Mr. Thomas Perkins
Plant Superintendent
Nine Mile Point Plant
300 Erie Boulevard West
Syracuse, New York 13202

Chief, Energy Systems Analysis Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 645, East Tower
401 M Street, N. W.
Washington, D. C. 20460

U. S. Environmental Protection Agency
Region II Office
ATTN: EIS COORDINATOR
26 Federal Plaza
New York, New York 10007

Oswego County Office Building
46 E. Bridge Street
Oswego, New York 13126

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-220

NIAGARA MOHAWK POWER CORPORATION

NOTICE OF PROPOSED ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The United States Nuclear Regulatory Commission (the Commission) is considering issuance of an amendment to Facility Operating License No. DPR-63, issued to Niagara Mohawk Power Corporation (the licensee), for operation of the Nine Mile Point Nuclear Station, Unit No. 1, located in Oswego County, New York.

The amendment would revise the provisions in the Technical Specifications to increase the spent fuel storage capacity from 1984 fuel assemblies to 3009 fuel assemblies in accordance with the licensee's application for amendment dated March 22, 1978.

Prior to issuance of the proposed license amendment, the Commission will have made findings required by the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations.

By December 7, 1978 , the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written petition for leave to intervene. Requests for a hearing and petitions for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR Part 2. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of hearing or an appropriate order.

As required by 10 CFR §2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend his petition, but such an amended petition must satisfy the specificity requirements described above.

Not later than fifteen (15) days prior to the first prehearing conference scheduled in the proceeding, the petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter, and the bases for each contention set forth with reasonable specificity. A petitioner who fails to file such ı supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

A request for a hearing or a petition for leave to intervene shall be filed with the Secretary of the Commission, United Sttaes Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Docketing and Service Section, or may be delivered to the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. by the above date. Where petitions are filed during the last ten (10) days of the notice period, it is requested that the petitioner or representative for the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at (800) 325-6000 (in Missouri (800 342-6700). The Western Union operator should be given Datagram Identification Number 3737 and the following message addressed to Thomas A. Ippolito: (petitioner's name and telephone number); (date petition was mailed); (plant name); and (publication date and page number of this FEDERAL REGISTER notice). A copy of the petition should also be sent to the Executive Legal Director, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, and to Eugene B. Thomas, Jr., Esquire, LeBoeuf, Lamb, Leiby and Mac Rae, 1752 N Street, N. W. Washington, D. C. 20036, attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board designated to rule on the petition and/or request, that the petitioner has made a substantial showing of good cause for the granting of a late petition and/or request. That determination will be based upon a balancing of the factors specified in 10 CFR §2.714(a)(i)-(v) and §2.714(d).

For further details with respect to this action, see the application for amendment dated March 22, 1978, which is available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Oswego County Office Building, 46 E. Bridge Street, Oswego, New York 13126.

Dated at Bethesda, Maryland this 27th day of October 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

T. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

NIAGARA MOHAWK

NIAGARA MOHAWK POWER CORPORATION / 300 ERIE BOULEVARD WEST, SYRACUSE, N.Y. 13202 / TELEPHONE (315) 474-1511

December 21, 1978

Division of Operating Reactors
Attn: Mr. Thomas A. Ippolito
Operating Reactors Branch No. 3
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Re: Nine Mile Point Unit 1
Docket No. 50-220
DPR-63

Dear Mr. Ippolito:

Your letter of October 18, 1978 requested additional information regarding the spent fuel pool modification for Nine Mile Point Unit 1. The attached information is in response to your request.

Very truly yours,

NIAGARA MOHAWK POWER CORPORATION

D.P. Dise
Vice President, Engineering

SWW:bd

Attach.

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Responses to October 18, 1978

Nuclear Regulatory Commission Questions

Nine Mile Point Unit 1
Docket No. 50-220
DPR-63

1. Question

Provide a description of the modifications that have been made to the Nine Mile Point Unit 1 (NMP-1) spent fuel pool (SFP) after the pool was licensed to contain up to 1984 fuel assemblies. Discuss the effects of these modifications on operating the SFP. This should include the measured man-rem exposure to do the work, the change in radiation levels in the vicinity of the pool, the change in the amount of crud in the pool, and changes to the operation of the SFP purification system.

Response

Figure 1 outlines the current configuration of the spent fuel storage pool. Seventeen standard General Electric spent fuel storage racks and six channel storage racks were removed from the spent fuel pool. One control blade storage rack was relocated from the north end of the pool to the south end. Seven control blade storage racks were removed from the spent fuel pool and will eventually be relocated. Installed in the north end of the spent fuel pool were 1050 high density stainless steel spent fuel storage locations. Of the racks removed, 15 standard spent fuel storage racks and 2 channel storage racks were decontaminated, packaged and shipped off site.

Table 1 contains the measured man-rem exposure which resulted from performing the aforementioned modification.

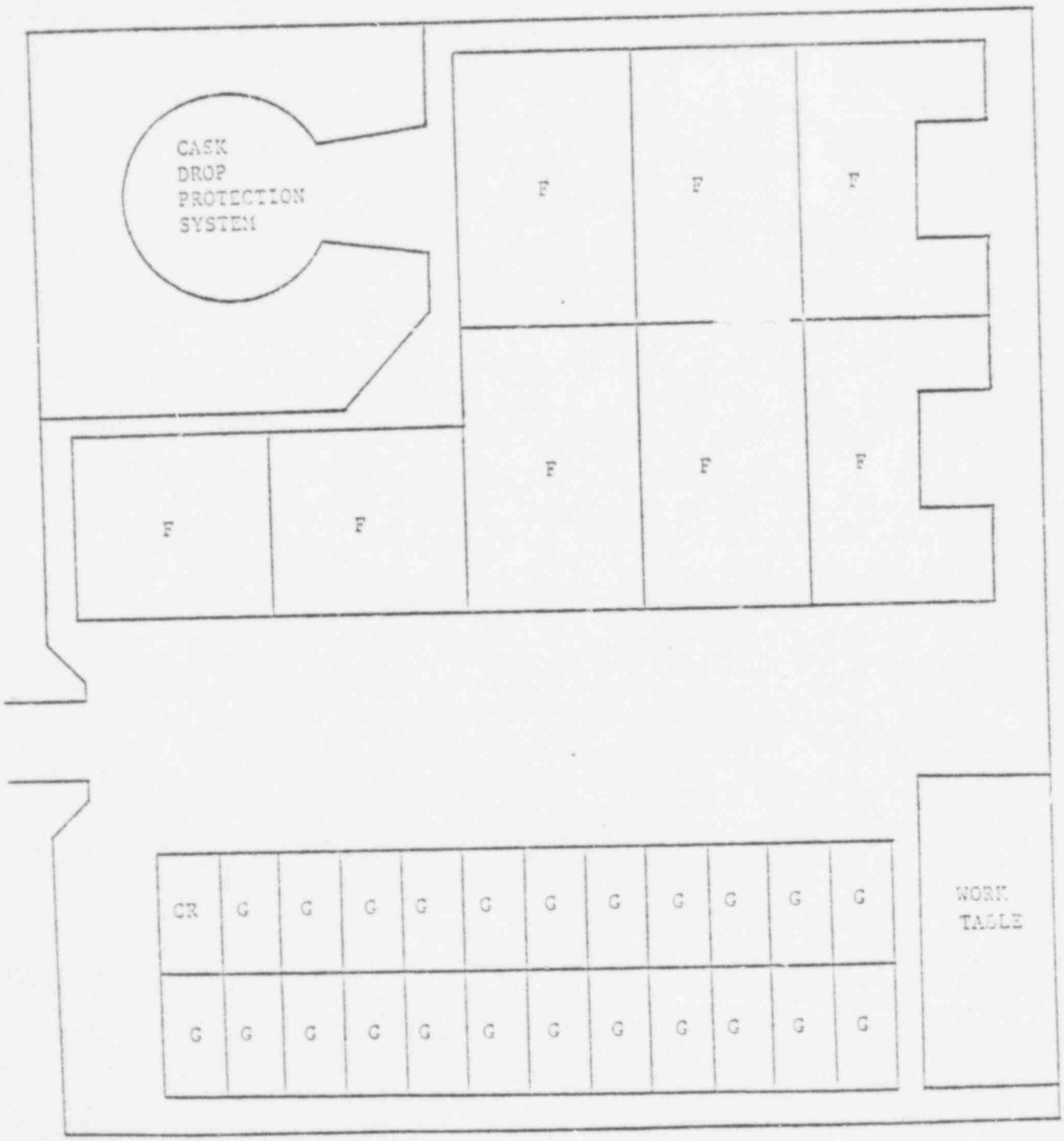
Table 2 contains a summary of the change in radiation levels in the vicinity of the spent fuel pool and changes to the operation of the spent fuel pool purification system. During normal operation the filter is changed out when approximately 1000 disintegrations/minute/milliliter is approached on the filter discharge which normally occurs every 2-3 months. Crud was stirred up from the pool floor during the rack removal and pool clean up prior to installation of the new racks. This necessitated a filter change on 7/11/78, after 22 days of filter residence time. Prior to installation of the new racks crud was removed by vacuum cleaner from the floor of the pool. Therefore, during the modification the total volume of crud in the pool was reduced.

FIGURE 1

NINE MILE POINT UNIT 1

SPENT FUEL POOL CONFIGURATION

FALL 1978



F - FLUXTRAP DESIGN HIGH DENSITY STAINLESS STEEL RACK
G - GENERAL ELECTRIC STANDARD SPENT FUEL STORAGE RACK
CR - CONTROL BLADE STORAGE RACK

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Table 1

1978 SPENT FUEL POOL MODIFICATION
MAN-REM EXPOSURE

Activity	Exposure Man-Rem
General Preparation	1.1
Fuel Movements	0.2
Divers Exposure During Modification	2.6
Install Seismic Restraints and New Fuel Racks	2.8
Decontaminate Lifting Rig	0.5
Remove Old Racks, Decontaminate and Package	1.2
TOTAL	8.4

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Table 2

1978 SPENT FUEL POOL MODIFICATION
FUEL POOL FILTER CHANGEOUT

		Gross Gamma - dpm/ml* BEFORE FILTER	AFTER FILTER	DOSE RATES ABOVE POOL-MR/hr
JUNE	5	803	654	5-20
JUNE	12	976	804	5-10
		FILTER CHANGEOUT		
JUNE	19	326	143	5-10
JUNE	26	398	229	5-15
JULY	2	605	576	5-20
JULY	11	1001	959	5-20
		FILTER CHANGEOUT		
JULY	18	408	396	5-20
JULY	24	396	237	5-20
JULY	31	555	461	5-20

*disintegrations/minute/milliliter

2. Question

You stated in your letter to NRC dated September 29, 1977, in your response to Question 5. that the dose rates around the spent fuel pool are kept between 5 and 10 mrem/hour by controlling the frequency of changing the pool filter. Justify why the radiation fields will be as low as reasonably achievable (ALARA) during each phase of the proposed pool modification. Relevant experience at nuclear power plants show typical values, in the vicinity of spent fuel pools, of 1 to 2 mrem/hour. Your response should consider increased purification system operation or increased filter change out frequency.

Response

Niagara Mohawk stated in its letter to the Nuclear Regulatory Commission dated September 29, 1977, in response to Question 5, that the dose rates above the pool generally vary between 5 and 10 millirem/hour. However, as stated in response to Question 6 of the aforementioned letter, effective dose rates for work carried on in the vicinity of the pool amount to less than 5 millirem/hour because rates in many areas of the floor are only 1 to 2 millirem/hour. Therefore, although normally the dose rates at the pool surface vary between 5 and 10 millirem/hour, the average dose rates in the vicinity of the pool, where work is carried on, is consistent with experience at other nuclear power plants.

During removal of the old spent fuel pool storage racks and during the general pool cleanup where crud was vacuumed from the floor of the spent fuel pool, radiation levels varied between 5-20 millirem/hour at the surface of the spent fuel pool. As shown on Table 2, it has been proven that increased filter change out does not appreciably reduce the dose rates at the surface of the spent fuel pool.

During each phase of installation, general pool cleanup will be performed to remove a substantial portion of crud from the pool floor which in the long run is expected to result in decreased spent fuel pool purification system filter changeout frequency. In any event the modifications will not increase the total volume of crud in the spent fuel pool.

3. Question

You stated in your March 22, 1978 submittal that the additional spent fuel storage capacity you are requesting will be installed in steps as your storage needs dictate. Compare the man-rem exposure for the proposed stepwise pool modification with the exposure for completing the modification in a single step. Show that your proposed course of action is consistent with the ALARA philosophy of 10 CFR Part 20.1(c).

Response

The man-rem exposure for the proposed stepwise pool modification is expected to be approximately the same as the exposure for completing the modification in a single step. A majority of the work activities associated with the spent fuel pool modification such as new rack preparation, fuel movements, old rack removal, decontamination and disposal, and new rack and seismic restraint installation are in direct proportion to the number of new storage locations installed. A slight reduction in exposure due to less spent fuel inventory may result if the modification were completed in a single initial step, but it is expected this decrease would be minimal.

The timing of eventual offsite shipment of spent fuel is not known at this time. Installing the complete modification in a single step may result in non-utilization of locations should offsite shipment of fuel take place prior to the pool being filled. This would result in a higher man-rem exposure/required storage location. If the total requested capacity is not utilized in the plants lifetime, stepwise modification would result in less man-rem exposure, consistent with ALARA guidelines.

4. Question

Discuss the occupational exposure expected during each separate matrix listed in Table 1 of your March 22, 1978, submittal of this proposed SFP modification. Address the expected dose rates (from spent fuel pool water, spent fuel and the equipment to be disposed of), numbers of workers (including divers, if necessary) and occupancy times for each phase of the operation. Include removal and disposal of the present spent fuel racks and installation of the new higher density racks. Provide the estimated man-rem exposure. Compare the measured and estimated man-rem exposure for your 1978 SFP modification.

Response

Table 3 contains a summary of the possible storage matrices outlined in Table 1 of the March 22, 1978 submittal. The selection of future storage matrices will be made based on storage requirements and status of the ultimate disposition of the backend of the fuelcycle.

The spent fuel pool is currently licensed to contain 1984 stainless steel high density spent fuel locations (storage matrix no. 1 of Table 3). The current status of the spent fuel pool is also shown on Table 3. Storage matricics 1-3 outline possible future stepwise expansion of the spent fuel pool utilizing high density stainless steel racks. Storage matricies 4-6 outline the same stepwise expansion utilizing posion design spent fuel storage racks. Storage matrix 7 assumes the high density stainless steel racks installed during the 1978 modification are replaced with posion design spent fuel storage racks.

Table 4 contains the projected occupational exposure for each separate matrix. These projected exposures were extrapolated from data obtained during the 1978 modification. The expected dose rates around the spent fuel pool are expected to be the same as those summarized in Response 3 above.

Niagara Mohawk stated in its letter to the Nuclear Regulatory Commission dated September 29, 1977 in response to Question 6, that the conservative estimate of exposure during removal of the old racks and installation of the new ones would be approximately 16 man-rems. The measured man-rem exposure for the 1978 modification was 8.4 man-rems.

TABLE 3

NINE MILE POINT UNIT 1

SPENT FUEL POOL STORAGE MATRIX

Pool Area	Current Status	Possible Future Storage Matricies						
		1	2	3	4	5	6	7
I	1060/F	1060/F	1060/F	1060/F	1060/F	1060/F	1060/F	1292/P
II	460/G	924/F	924/F	924/F	1080/P	1080/P	1080/P	1080/P
III	C	C	C	440/F	C	C	517/P	517/P
IV	T	T	96/F	96/F	T	120/P	120/P	120/P
Projected Capacity	1520	1984	2080	2520	2140	2260	2777	3009

F - Currently Licensed stainless steel high density flux trap spent fuel storage rack.
G - Standard General Electric spent fuel storage rack.
P - High density poison design spent fuel storage rack.
C - Control blade storage rack.
T - Work table.

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TABLE 4

NINE MILE POINT UNIT 1

PROJECTED REQUIREMENTS FOR SPENT FUEL POOL MODIFICATIONS

	Possible Future Storage Matricies*						
	1*	2**	3**	4*	5**	6**	7**
Number of Workers (Approximate)	80	80	80	80	80	80	80
Projected Manhours (Approximate)	3200	3300	4000	3200	3300	4000	5500
Contaminated Material Shipped Offsite (Cubic Feet)	7600	7600	9100	7600	7600	9100	14300
Pool Modification (Man-Rems)	13.5	14.1	17.1	13.5	14.1	17.1	24.2
Removal & Disposal of Old Racks (Man-Rems)	1.8	1.8	2.8	1.8	1.8	2.8	4.8
Total Man Rems	15.3	15.9	19.9	15.3	15.9	19.9	29.0

* Includes manhours, contaminated material shipped offsite, and exposures accumulated during the 1978 modification.

** Includes manhours, contaminated material shipped offsite, and exposures accumulated during the previous steps of the modification.

5. Question

Provide the additional occupational exposure (in man-rem) from normal operation in the spent fuel pool area, including refueling, over the time period for the complete SFP modification proposed in your March 22, 1978, submittal. Include the expected exposure from more frequent changing of SFP filters and demineralizers, from spent fuel pool water and from spent fuel.

Response

It is anticipated the additional occupational exposure from normal operation in the spent fuel pool area over the time period for the complete spent fuel pool modification will be minimal. Most of the dose rate contribution above the pool is due to activity in the water from recently discharged spent fuel bundles. Since the proposed expansion would not increase the number of recently discharged bundles, the change in dose rates above the pool is expected to be minimal.

A slight increase in the spent fuel pool filter change out frequency may be required due to the spent fuel pool modification. However, filter change out is performed remotely and usually results in less than 5 millirem of exposure. Additionally, general pool cleanup performed during the modification may result in decreased filter changeout frequency over the life of the pool.

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6. Question

Describe the method that will be used to dispose of the present racks (i.e., crating intact racks or cutting and packaging). If the racks are to be cut and packaged, show that the exposure received by this disposal method, as compared to crating the intact racks for disposal, will provide as low as is reasonably achievable (ALARA) exposure to personnel. Your response should include a description of the method that was used to dispose of the racks and resultant exposures from the 1978 SFP modification.

Response

The method of disposing of the present racks has not yet been determined. The methods of crating intact racks or cutting and packaging appear at this time to be the most likely options. Prior to disposal of racks, Niagara Mohawk will evaluate both the economic and personnel exposure costs and benefits of all proposed methods of disposing of racks. The results of this evaluation will determine the method which will be used.

The racks which were disposed of from the 1978 spent fuel pool modification were crated intact, shipped in low specific activity boxes and disposed of at a low level burial site. Our evaluation at that time showed that the disposal costs saved by cutting and packaging did not warrant the added exposure which was required for cutting. The resultant exposure from the removal, decontamination and packaging of old racks from the 1978 spent fuel pool modification was 1.2 man-rems.

7. Question

Discuss the leakage of water from the spent fuel pool and the pool leak collection system. Where would the pool leakage be transferred to?

Response

The spent fuel pool at Nine Mile Point Unit 1 does not have drains or hose connections that by maloperation or failure could cause loss of coolant. The spent fuel pool filtering and cooling system (described in Section X.H of the Nine Mile Point Unit 1 FSAR) is a seismic Category 1 system, and is provided with siphon breakers to prevent loss of coolant in the event of a line break. Based upon the preceding information, leakage is not possible from any piping connections to the spent fuel pool.

Niagara Mohawk has performed an evaluation to determine if there would be leakage in the event of a rupture of the spent fuel pool liner. The results of the evaluation indicate that inconsequential leakage through the construction joints at elevation 318 feet 0 inches may be present should the liner be ruptured. This leakage would be collected in the reactor floor drain system and processed as liquid waste.

8. Question

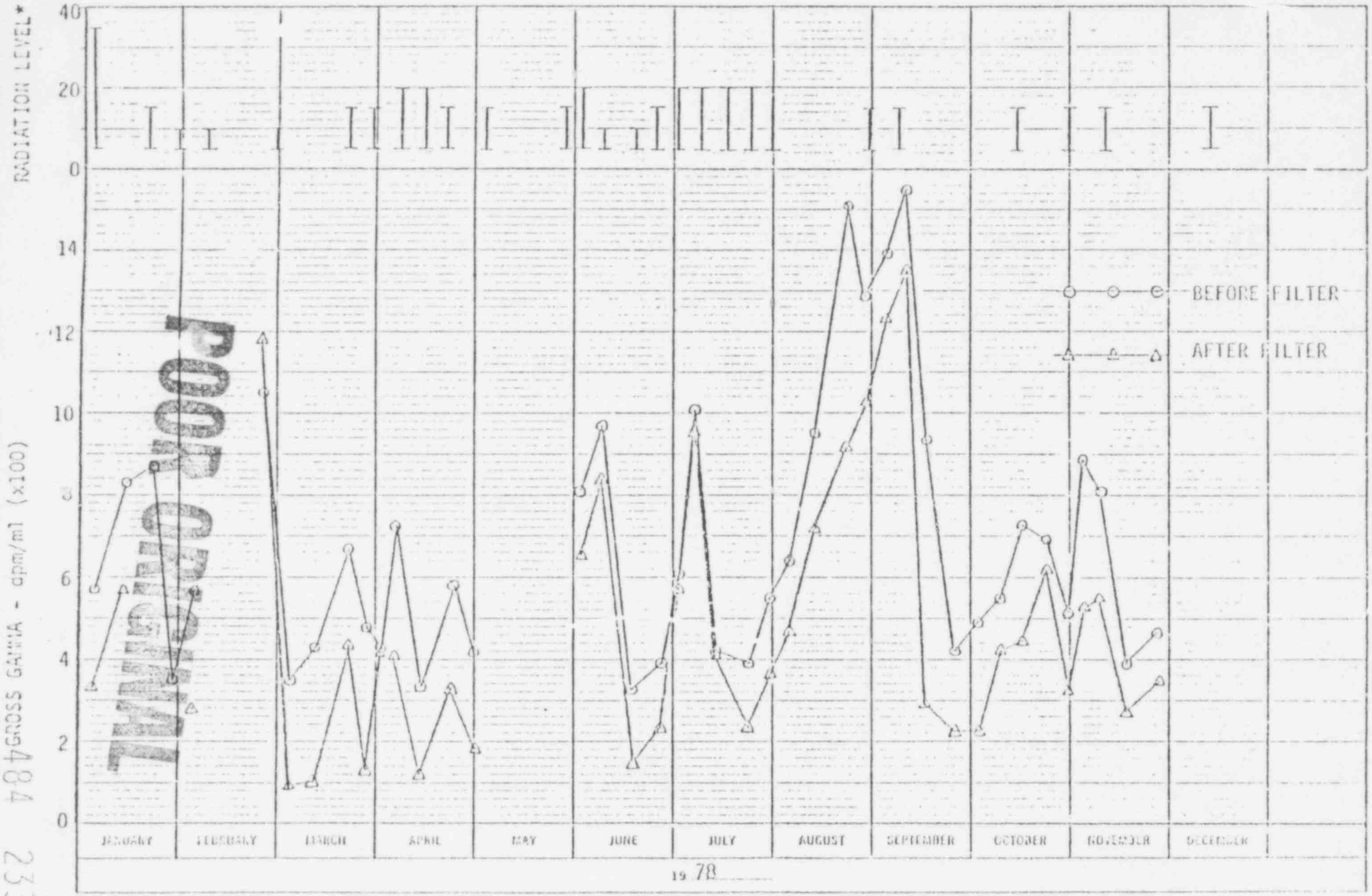
Discuss the effect of the fuel assembly movements during your 1978 SFP modification on the amount of crud in the pool water and the radiation levels in the vicinity of the pool. Your response should include measured radioactivity concentrations of SFP water and general radiation levels in the vicinity of the pool before, during and after the 1978 SFP modification.

Response

Figure 2 summarizes the spent fuel pool filter gross gamma count and the general radiation levels above the surface of the pool before, during and after the 1978 spent fuel pool modification. Fuel assembly movements were conducted between June 2-9 and resulted in very little increase in crud disposition. The greatest disturbance of crud was encountered during removal of the old spent fuel storage racks and during the general pool clean-up where the crud was vacuumed from the floor of the spent fuel pool. During the modification, due to the pool cleanup, the total volume of crud in the spent fuel pool was decreased.

FIGURE 2

NINE MILE POINT UNIT 1 SPENT FUEL JOL DATA



*Radiation level above the surface of the spent fuel pool in millirem per hour.

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9. Question

Your March 22, 1978 submittal did not completely address the impact of the proposed SFP modification on the environment. Discuss in some detail the impact of the proposed SFP modification on the following:

a) radioactive liquid effluents from the plant, including leakage of water from the pool and

b) radioactive solid wastes from the plant, including the change in the frequency of replacing the SFP filter demineralizer resin.

Response

The proposed spent fuel pool modification will not result in any change in the radioactive liquid effluents from the plant. All liquid wastes from the spent fuel pool including filter sludge and possible leakage of water from the pool will be processed by the Liquid Radwaste System.

There may be a slight increase in the volume of filter sludge due to the possible slight increase in filter change frequency.

At the present time, the spent fuel pool filter sludge is filtered and concentrated. The filter cake and concentrator waste are disposed of off site as solid radioactive waste. The modification is expected to increase the Liquid Radwaste System usage by less than 1 percent.

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10. Question

Provide the estimated volume of contaminated material (e.g., spent fuel racks, seismic restraints) expected to be removed from the spent fuel pools during each step of the entire modification and shipped from the plant to a licensed burial site.

Response

During the 1978 spent fuel pool modification, approximately 4600 cubic feet of contaminated material including standard General Electric spent fuel storage racks and channel racks were decontaminated, packaged and shipped offsite. Table 4 contains the estimated volume of contaminated material to be removed and shipped from the plant for the possible future storage matrices.

11. Question

Provide a list of typical loads that might be carried near or over the spent fuel pool. Provide the weight and dimensions of each load. Discuss the load transfer path, including whether the load must be carried over the pool, the maximum height at which it could be carried and the expected height during transfer. Provide a description of any written procedures instructing crane operators about loads to be carried near the pool. Provide the number of spent fuel assemblies that could be damaged by dropping and/or tipping each typical load into the pool.

Response

In a letter from D. P. Dise to V. Stello dated July 17, 1978, Niagara Mohawk submitted information relative to heavy load movement near the spent fuel storage pool. This information was requested by Mr. Stello on May 17, 1978 in a letter to all Licensees of Power Reactors except those in the Systematic Evaluation Program.

12. Question

Discuss the instrumentation to indicate the spent fuel pool water level and water temperature. Include the capability of the instrumentation to alarm and the location of the alarms.

Response

This question was previously answered in response to question 20 of our letter to NRC dated December 20, 1978.

13. Question

Propose a technical specification which prohibits carrying loads greater than the weight of a fuel assembly over the spent fuel in the storage pool; or justify why this specification is not needed to limit the potential consequences of accidents involving dropping heavy loads, other than casks, onto spent fuel to those of the design basis fuel handling accident.

Response

As stated in response to Question 11, the Nuclear Regulatory Commission has received information concerning movement of heavy loads over the spent fuel pool. It is assumed that when this review is completed, recommendations will be published relative to heavy load movement over spent fuel pools.

NIAGARA MOHAWK POWER CORPORATION/300 ERIE BOULEVARD WEST SYRACUSE, N.Y. 13202/TELEPHONE (315) 474-1511

December 20, 1978

Division of Operating Reactors
Attention: Mr. Thomas A. Ippolito
Operating Reactors Branch #3
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Re: Nine Mile Point Unit #1
Docket No. 50-220
DPR-63

Dear Mr. Ippolito:

Your letter of August 21, 1978 requested additional information regarding the spent fuel pool modification for Nine Mile Point Unit 1. The attached information is in response to your request.

Very truly yours,

NIAGARA MOHAWK POWER CORPORATION

D. P. Dise
Vice President - Engineering

SWW/kmb
Attachment

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Response to August 21, 1978

NUCLEAR REGULATORY COMMISSION QUESTIONS

Nine Mile Point Unit 1
Docket No. 50-220
DPR-63

1. Question

Provide the details of the uranium-235 loading and distribution in the fuel assembly which were used for the spent fuel pool criticality calculations.

Response

Figures 7 and 8 contained in the March 1978 submittal detail the fuel assembly model used in the criticality calculations. The fuel assembly is comprized of 64 0.64 inch by 0.64 inch cells containing homogenized fuel, water and cladding. The fuel assemblies are assumed to be unirradiated with an assembly average enrichment of 3.0 weight percent which is equivalent to a loading of 15.186 grams per axial centimeter of fuel assembly. A calculation, representing an explicit fuel pin distribution of selected U-235 enrichments typical of a 3.0 weight percent General Electric intra-assembly fuel pin arrangement is performed and included as a perturbation on the single average enrichment model results.

2. Question

Provide the areal density of boron-10 in the Boraflex plates which was used in the criticality calculations and provide the "quality assured" minimum value of this areal density (i.e., atoms of boron-10 per square centimeter of Boraflex plate).

Response

The boron-10 loading used for the criticality analysis was based on the minimum B_4C loading of 34.8 weight percent in the Boraflex and assuming 76 weight percent boron in the B_4C powder. This corresponds to a boron-10 loading in the minimum 0.10 inch thick Boraflex sheet of 0.0217 grams per square centimeter of cross section area (0.0217 gm B-10/cm^2-0.1 inch). This represents the minimum areal density required by specification to be incorporated into the as fabricated Boraflex plates. As a perturbation on the criticality calculation it was assumed that the boron content of the B_4C powder was 70 weight percent (0.0200 gm B^{10}/cm^2-0.1 inch), which represents an 8.5 percent reduction in boron-10 content of the Boraflex. Current production techniques indicate that the actual boron-10 loading in the Boraflex will be well above the 0.0217 gm B-10 value.

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3. Question

Provide the calculated change in the storage lattice k∞ with a small change in the boron-10 concentration in the Boraflex plates.

Response

The calculated change in the storage lattice k∞ with a small change in boron-10 concentration in the Boraflex is shown in Figure 1.

4. Question

Will there be any place in the pool under any of the proposed changes where it will be possible to place a fuel assembly very close to the outside of a rack which is filled with fuel assemblies? Such a place might be the space between the outer periphery of the rack modules and the walls of the pool. What is the maximum neutron multiplication factor that can be obtained in this situation?

Response

Yes. However, the reactivity effect of a fresh fuel assembly located outside the fully loaded spent fuel storage rack has been evaluated for all postulated locations, including this area. The maximum perturbation associated with this event has been calculated to be +0.018Δk.

5. Question

Provide the nominal value of the k∞ for the PDQ-7 cell model shown in Figure 7 of your submittal.

Response

The calculated k∞ of the basic cell at 100 degree F is 0.8775.

6. Question

Provide the maximum value of the k∞ that is obtained when the perturbations listed in Table 3 of your submittal are all assumed to be in the direction that gives an increased neutron multiplication factor.

Response

The maximum value of k∞ for the spent fuel rack under worst mechanical and thermal conditions with calculational uncertainties is 0.8971.

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7. Question

Provide the change in the multiplication factor when axial neutron leakage is assumed.

Response

The change in multiplication factor when axial leakage is assumed is -0.0021Δk.

8. Question

NRC procedures require that an on-site test be performed to verify within 95 percent confidence limits, that a sufficient number of neutron absorbing plates (i.e., poison plates) in the installed racks contain the required boron content to maintain the $k_{eff} \leq 0.95$. When the poison plates are made an integral part of the racks and the condition of the poison plates is continually monitored by surveillance tests, the NRC finds that a single, initial neutron attenuation test on the racks is sufficient. However, a single, initial neutron attenuation test will not be sufficient for the proposed racks with the removable poison inserts. Describe how you propose to periodically perform tests to verify, within 95 percent confidence limits, that there will always be a sufficient number of poison plates which contain the required boron content to maintain the $k_{eff} \leq 0.95$ in the proposed storage racks with the removable poison inserts.

Response

The rack design provides one semi-permanent poison assembly per fuel assembly. The poison inserts are not normally removeable and it is anticipated will not be removed from the fuel storage rack for the life of the rack, therefore we will satisfy NRC procedures. The poison assembly is locked in place by a lock bolt that cannot be removed without mechanically deforming the locking element. The lead-in funnel at the top of the fuel cell location is an integral part of the poison assembly and thus the presence of the neutron poison plates can be verified visually. The presence of the neutron poison plates will be verified visually prior to the installation of the storage racks in the pool.

Quality Assurance records, during manufacture, will confirm the installation of the Boraflex poison material into every poison assembly. A neutron attenuation test will be performed at the manufacturer's plant on 10 percent of the storage locations after the poison assemblies are installed and locked in the racks. A site verification neutron attenuation check will be performed on at least 5 storage locations in each rack module.

It should be noted that manufacturing process Quality Assurance procedures and controls provide assurance that the actual B_4C content of the Boraflex poison material is above minimum calculational levels established by Reactor Physics calculation models for each batch of B_4C and silicone material.

9. Question

Will any sources of neutrons other than spent fuel assemblies be stored in the spent fuel pool? If so, at what rate will they emit neutrons?

Response

Tabulated below are sources of neutrons other than spent fuel assemblies which may at times be stored in the spent fuel pool.

NEUTRON SOURCE	DESCRIPTION	NEUTRON RATE - n/sec
Start-up Sources	5 Sb-Be (γ-n) sources will be removed from the core during the next refueling outage, temporarily stored in the pool and eventually shipped off site (60 day half-life)	$\sim 4\times10^7 - 3\times10^9$ (each source)
Am-Be Source	1 Am-Be (α-n) source; occasionally stored in the spent fuel pool	$\sim 12\times10^6$
Depleted LPRM's	Upon removal from the reactor during refuelings, LPRM's are stored in the spent fuel pool until disposed of	Minimum

10. Question

What will the maximum integrated neutron and gamma flux be in the Boraflex material over the lifetime of the racks? What spent fuel assembly power density and burn-up, and what rack life were assumed in calculating these maximum integrated fluxes? What is the assumed energy spectrum for the gamma flux?

Reponse

The maximum integrated neutron and gamma dose expected for the poison material will be 1.2×10^{11} Rads. The neutron dose is expected to be no more than 5×10^8 Rads. For the gamma dose calculation, it is assumed that the poison is subjected to irradiation from the hottest spent fuel assembly (gamma dose design basis peaking factor of 1.25, burnup c 33,000 MWD/MT) from 10 days out of the reactor to 18 months when the next refueling occurs. It is assumed that the fuel assembly is then replaced by the hottest spent fuel assembly discharged at that time. There are 27 such cycles that correspond to a rack life of 40.5 years. Since over 70 percent of the gamma energy is in the 0 to 0.5 MEV group, 1 MEV gammas were assumed, making the resulting dose estimate conservatively high.

11. <u>Question</u>

What will the maximum temperature be in the center of the Boraflex material, assuming the highest neutron and gamma flux and the worst accident conditions?

<u>Response</u>

The maximum temperature in the center of the Boraflex material assuming worst accident conditions is 247 F.

12. <u>Question</u>

State the quantity and composition of the gas which will come out of the Boraflex material when it is being irradiated in the spent fuel pool.

<u>Response</u>

Currently a sample of Boraflex material is being exposed in the Phoenix Reactor at 2×10^5 R/hr (Gamma). The gas release from a 0.5 inch x 0.1 inch x 6 inch sample is 15 ml/hr. The composition of the gas being released has been measured as 10% O_2, 15% H_2, 45% N_2, 14% methane, and 3% ethane.

13. <u>Question</u>

On page 9 of your March 22, 1978 submittal, it is indicated that there is a poison slab which is sealed in a stainless steel casing. Is the Boraflex going to be sealed inside of a stainless steel casing?

<u>Response</u>

The Boraflex is canned in a stainless steel casing. The assembly is vented at the top and bottom to prevent gas build up in the casing.

14. <u>Question</u>

What will the chemical composition of the Boraflex be after receiving the design dose of irradiation?

<u>Response</u>

There will be little or no change in the chemical composition of the Boraflex after receiving the design dose of irradiation.

15. <u>Question</u>

What is the melting temperature of the Boraflex in the unirradiated condition?

<u>Response</u>

The Boraflex material does not melt below 1000 F. At 1300 F the Boraflex material begins to char.

16. Question

Is the Boraflex going to be bonded to stainless steel? If so, what will happen to this bond under the design dose of irradiation in conjunction with the design number of thermal cycles?

Response

The Boraflex is not bonded to the stainless steel casing.

17. Question

What will the physical properties such as the density, the modulus of rupture, and the compressive strength of the Boraflex be after it receives the design dose of irradiation in the spent fuel pool?

Response

Tests performed to date at greater than the design dose indicate that the only change in properties of the Boraflex after irradiation is an increase in stiffness of the material. This change does not consititute a problem with the material in regard to its function as a neutron absorber.

18. Question

Provide a detailed description of and the documented results of a prototypical experiment, which includes all significant aspects of the spent fuel pool situation and environment, that shows that these Boraflex plates will not become so brittle from irradiation in the spent fuel pool that they could be broken up by the insertion and removal of fuel assemblies or by a safe shutdown earthquake at some time in the design life of the spent fuel racks.

Response

A continuous program of irradiation of Boraflex is in progress at the Phoenix Reactor of the University of Michigan. Various tests on Boraflex to date, summarized in the attached "Boraflex 1 Suitability Report," No. 1047-1, have demonstrated that the material undergoes no significant physical or chemical change when exposed to 2 x 10^{19} rads gamma irradiation. Additional testing of the material is documented in the proprietary BISCO Procedure 748-10, "Irradiation Studies of Neutron Shielding Materials", a copy of which can be supplied on request.

The Boraflex material is fully supported in a double stainless steel can that is vented at the top and bottom to allow any gas generated to escape. The poison assembly is a complete unit that does not provide the bearing surface for the spent fuel. The rack structure provides this . Any distortion of the poison box assembly is contained in the slot within the rack that it occupies. A distortion will not affect the fuel cell.

All the structural analyses performed on the poison design racks assumed the poison material had zero strength and contributed nothing to the strength of the rack.

19. Question

If the Boraflex is to be exposed to the pool water, state the maximum percentage of boron oxide, B_2O_3, in the B_4C. Since B_2O_3 is soluble in water it will either be necessary to assume that this amount of boron is leached from the Boraflex plates or to experimentally show that this will not happen during the life of the racks.

Response

The B_2O_3 content in the B_4C is less than 0.75.

The Boraflex will be exposed to pool water, however, the silicone rubber matrix of the Boraflex essentially coats the B_4C and B_2O_3 particles to prevent the water from contacting it. A 4700 hour test at 240 F in Borated water has shown no leaching to occur. Even if leaching should occur, analysis shows this small decrease in Boron content would have little effect on the rack k_∞ (see figure 1).

20. Question

Describe the instrumentation and alarms on the spent fuel pool water level and temperature or reference the location in the FSAR where this description can be found.

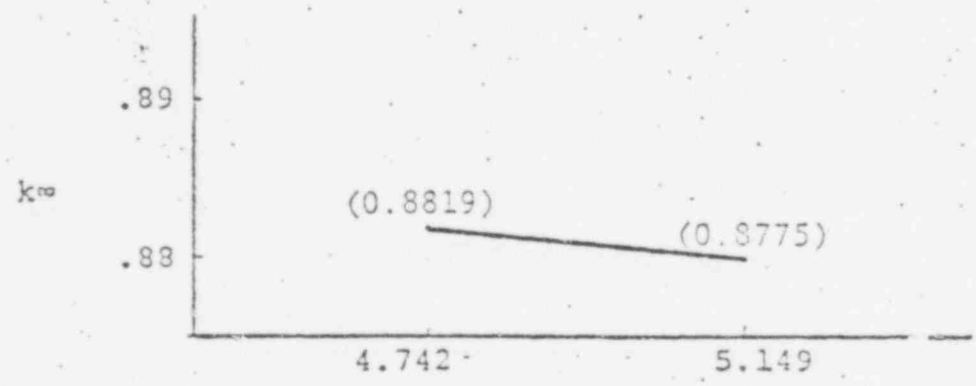
Response

The level alarm system initiates when the water level in the spent fuel pool falls to elevation 338 feet 0 inches. The level alarm annunciator is located in the control room and on the spent fuel pool panel located on elevation 281 feet of the reactor building. The spent fuel pool filtration system has a low pump suction trip alarm and a low flow alarm, both of which would indicate low water level. The pump suction trip alarm and low flow alarm annunciate on the spent fuel pool panel at elevation 281 feet and the computer prints out an alarm in the control room. In addition, the computer prints out an alarm when the water level falls to elevation 338 feet 0 inches.

The temperature alarm system annunciator is located on the fuel pool panel on elevation 281 feet 0 inches, and a high fuel pool temperature alarm is printed on the computer. In addition, there is a fuel pool alarm in the control room which can be initiated by high temperature or anyone of several other parameters. When the fuel pool alarm is annunciated, it can be determined from the computer printout which of the parameters has caused the alarm, and action can be taken. The temperature alarm system is set to initiate at 113 F.

FIGURE 1

NINE MILE POINT NUCLEAR PLANT
SPENT FUEL STORAGE RACK k∞
vs
BORON-10 LOADING IN BORAFLEX®



B^{10} Loading in Boraflex, Atoms B^{10} cm^{-3} x 10^{21}

48? 247

RELATED CORRESPONDENCE

Attachment for

Susquehanna

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DEC 14 1978

Docket Nos. 50-387
and 50-388

Mrs. Stanley Shortz
RFD #1, Box 245
Berwick, Pennsylvania 18603

Dear Mrs. Shortz:

This is in response to your letter of November 9, 1978, to the Secretary of the Nuclear Regulatory Commission concerning storage of spent power reactor fuel at the Susquehanna Steam Electric Station.

Storage facilities for approximately 1900 spent fuel assemblies were authorized by the Commission in the construction permits for Units 1 and 2. These facilities are being constructed in the reactor building. The storage capacity thus provided would accommodate approximately two and one-half reactor cores since each core would contain 764 fuel assemblies.

Irradiated spent fuel must remain in safe storage on site after removal from the reactor until the radioactivity and thermal levels within the fuel have declined to those for which the shipping packages are licensed. Typical storage intervals for this purpose are in the range of 90 to 150 days. NRC does not set a maximum limit on the length of time spent fuel can remain at the plant; however, the available storage capacity does impose a practical limitation. The Pennsylvania Power and Light Company's present schedule is for startup of Unit 1 in May 1980 and startup of Unit 2 in November 1981. Assuming operation at 80 percent capacity factor and replacement of one-quarter of each core every 18 months thereafter, the spent fuel storage facility on site would be full by August 1988 unless some of the spent fuel is shipped elsewhere. [However, we understand that plans for expansion of its capacity are being formulated by the licensee. When a request is received for amendment of the construction permits relative to this matter, the NRC will review the details to assure that the health and safety of the public are protected.]

Additional power reactor spent fuel storage is eventually expected to become available at facilities which will be provided for a fee by the Department of Energy, in accordance with its policy announcement in October 1977. Various options are being considered which could result in shipments to such interim storage facilities in the mid-1980s and to

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disposition facilities commencing during the 1983-1993 time frame. Further details are provided in the Draft Environmental Impact Statement on Storage of U. S. Spent Power Reactor Fuel (DOE/EIS-0015-D) which can be requested from Mr. W. H. Pennington, Mail Station E-201, U. S. Department of Energy, Washington, D. C. 20545.

Sincerely,

/s/

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

[illegible]

RFD #1 Box 148
Berwick, Penna 18603

November 9, 1978

Secretary, U.S. NRC
Washington, D.C.,
Attn: Docketing & Service Branch

Gentlemen:

This letter is in reference to the storage of spent nuclear fuel at the Susquehanna Steam and Electric Station at the Bell Bend site. This is my formal objection to such an arrangement.

On the 20th of September I wrote a letter concerning this subject to Mr. Wm. H. Regan Jr., chief of environmental projects. In response to this letter, I received a telephone call from a Mr. (Singhay) of the office. At that time he assured me that the spent fuel would be stored at the site only until it would be safe to ship it to a storage site. This he said would be a maximum of six months. During this conversation I was informed that a team from the NRC would be in this area to inspect the plant site on the 17th and 18th of October and they would stop by our home and discuss these issues with me.

On the 11th of October a very small article appeared in our local paper concerning the storage of the spent fuel at the Bell Bend site. This article couldn't have been any smaller. In fact it contained three sentences To say the least it went unnoticed by the majority of the area residents.

However, when the team made their visit to our home, I questioned Mr. Singhay (who was accompanied by two other persons) and he informed me that the spent fuel would indeed be stored at the plant site for 10 years. He didn't say it was pending any certification. It was a definite statement that the fuel would indeed be stored there for the 10 year period. It seems that this like all the other things that PP&L has decided to do in this area will indeed be a reality.

Up to this point the government has done nothing to protect the residents of this area. We have received damage to our properties during the construction of the plant, which I am sure the authorities were well aware of at the time, but chose to ignor. Our beautiful area has been desecrated by the scars of power lines in all

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POOR ORIGINAL

[illegible] with the distinct possibility of more to come. All this for someone else's benefit. We didn't need the power that will be generated at this plant for this area. Why wasn't this plant constructed in the area that it would serve?

Now the final destruction of this area by the storage of this spent fuel. No other area in the U.S. wants this spent fuel stored in their area and I am sure the apparent apathy of the people in this area gave assurance there would be no objection to this plan.

This subject was brought to the attention of the area supervisors at their meeting last evening. They were completely ignorant of the proposed plan.

I <u>think</u> it is about time the people in this area are informed as to what is to take place. We live here and we are definitely effected by this decision. At the present time there are over 300 sites in the U.S. where toxic chemicals and other dangerous materials have been stored. These sites have begun to make their presence known to the people who live there. I am not in favor of our area being added to that list.

I would like a definite statement on this issue. Just what is going to be done with the spent fuel? Where will it be stored?

Very truly yours,

Mrs. Stanley Shortz

Mrs. Stanley Shortz

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Attachment for

Maine Yankee

50-309 P

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DECEMBER 4 1978

P

Postponed Date TO Be Determined

Docket No. 50-309

MEMORANDUM FOR: Robert W. Reid, Chief, ORB#4, DOR

FROM: C. Nelson, Project Manager, ORB#4, DOR

SUBJECT: FORTHCOMING MEETING WITH MAINE YANKEE ATOMIC POWER COMPANY

Time and Date: 10:30 a.m., December 13, 1978

Location: P-110, Phillips Building
Bethesda, Maryland

Purpose: M.Y. proposes to store fuel in the spent fuel storage pool in a modified "compact" fuel bundle. The normal fuel assembly is a 15 x 15 pin matrix. The "compact" bundle is a 17 x 17 pin matrix. M.Y. has determined that this is not an unreviewed safety question. We will discuss this determination.

Participants:

Maine Yankee
R. Groce

NRC
C. Nelson, B. Grimes, R. Reid, E. Lantz, J. Donohew

M. B. Fairtile for
Christian Nelson, Project [illegible]er
Operating Reactors Branch [illegible]
Division of Operating Reacto[illegible]s

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MEETING NOTICE DISTRIBUTION

ORB#4

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484 255

MAINE YANKEE ATOMIC POWER COMPANY
ENGINEERING OFFICE
TURNPIKE ROAD (RT. 9)
WESTBORO, MASSACHUSETTS 01581
617-366-9011

B.3.2.1

WMY 78-104

November 22, 1978

United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Office of Nucl r Reactor Regulation

References: a) License No. DPR-36 (Docket No. 50-309)
b) MYAPC letter to USNRC, dated March 27, 1975 (Proposed Change No. 28)
c) MYAPC letter to USNRC dated June 26, 1975
d) MYAPC letter to USNRC dated July 25, 1975
e) USNRC letter to MYAPC dated October 31, 1975 (Amendment No. 11)

Dear Sir:

Subject: Modified Spent Fuel Pin Storage

Maine Yankee anticipates the loss of full core discharge capability in 1984, and the loss of refueling spent fuel discharge capability by 1987 unless the present spent fuel storage capability is enhanced. Therefore, Maine Yankee has developed and evaluated a spent fuel storage concept which will increase the spent fuel storage capability of the existing storage racks at an environmental and economic cost below that of any known alternative.

The new concept involves two distinct phases.

Phase one is termed "compaction" and involves disassembly of spent fuel assemblies, after a suitable cooling period, and reassembly into modified compact fuel bundles designed to provide a more compact fuel pin array. These compact fuel bundles fit into the storage locations of the existing spent fuel racks, yet allow storage of up to 60% more fuel pins in each storage location. This phase does not increase the amount of spent fuel stored in the spent fuel storage facility. Rather, it allows existing spent fuel to be stored in fewer spent fuel rack storage locations than would otherwise be possible.

The second phase involves storage of compacted fuel in those spent fuel rack storage locations vacated during phase one or as yet unfilled. This allows more spent fuel to be stored in the spent fuel pool than the 953 spent fuel assemblies which can be stored without phase one.

Based upon evaluations enclosed with this letter, which are provided for information and not for formal Staff review, Maine Yankee has determined that phase one can commence without prior Commission approval,

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pursuant to 10 CFR 50.59. Our determination is based on the following considerations:

a) Spent fuel compaction does not involve a change in the Technical Specifications of the Maine Yankee Facility Operating License.

b) Spent fuel compaction does not involve an unreviewed safety question, because:

 i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety evaluation report is not increased.

 ii) The possibility of an accident or malfunction of a different type than any evaluated previously in the safety evaluation report is not created.

 iii) The margins of safety as defined in the bases for Maine Yankee's Technical Specifications are not reduced.

Maine Yankee recognizes that the compact fuel bundle concept does involve changes to the facility and procedures as described in the safety analysis report and shall take the necessary measures to develop and review these changes.

These determinations have been reviewed by the Plant Operating Review Committee (PORC) and the Nuclear Safety Audit Review Committee (NSARC).

Phase two, the increase in total spent fuel inventory at the Maine Yankee site, will be addressed in future correspondence with the Commission.

Maine Yankee is providing this evaluation in advance of the annual summary of such changes, as required pursuant to 10 CFR 50.59, to keep the Commission informed of an innovative concept with potential for more widespread application.

We trust this information is satisfactory; however, if you should require additional information, please feel free to contact us.

Very truly yours,

MAINE YANKEE ATOMIC POWER COMPANY

D. E. Vandenburgh

D. E. Vandenburgh
Vice President

Enclosure

INTRODUCTION

Maine Yankee anticipates the loss of full core discharge capability in 1984, and the loss of refueling spent fuel discharge capability by 1987 unless the present spent fuel storage capability is enhanced. Therefore, Maine Yankee has developed and evaluated a spent fuel storage concept which will increase the spent fuel storage capability of the existing storage racks at an environmental and economic cost below that of any known alternative.

The new concept involves two distinct phases.

Phase one is termed "compaction" and involves disassembly of spent fuel assemblies, after a suitable cooling period, and reassembly into modified compact fuel bundles designed to provide a more compact fuel pin array. These compact fuel bundles fit into the storage locations of the existing spent fuel racks, yet allow storage of up to 60% more fuel pins in each storage location. This phase does not increase the amount of spent fuel stored in the spent fuel storage facility. Rather, it allows existing spent fuel to be stored in fewer spent fuel rack storage locations than would otherwise be possible.

The second phase involves storage of compacted fuel in those spent fuel rack storage locations vacated during phase one or as yet unfilled. This allows more spent fuel to be stored in the spent fuel pool than the 953 spent fuel assemblies which can be stored without phase one.

This document presents an evaluation of both phases: (1) compaction and (2) storage of more spent fuel than currently can be accommodated.

The reader is advised that for convenience both phases have been evaluated together. Therefore, when considering either compaction or increased spent fuel storage alone, some judgement must be exercised in separating the effects of one phase from those of the other.

1.0 PIN STORAGE CONCEPT

The reader is advised that for convenience, both phases have been evaluated together. Therefore, when considering either compaction or increased spent fuel storage alone, some judgement must be exercised in separating the effects of one phase from those of the other.

The pin storage concept, termed "fuel compaction", involves the disassembly of irradiated fuel assemblies and the reassembly of the loose fuel pins created into a skeleton cage similar to the standard cage. The compaction is derived from the elimination of CEA guide tubes and of rod spacer springs in the egg-crate grid assemblies and the elimination of poison rods (shims). The skeleton cage is depicted in Figure 1.1. A tighter but still square pitch is achieved with an envelope identical to that of a standard fuel assembly.

This scheme for increasing storage capacity is very simple, and can be implemented by adapting the procedures and equipment previously used for fuel reconstitution with which Maine Yankee and many other utilities are familiar.

The potential for increased storage is significant. The existing racks have spaces for a total of 953 fuel assemblies. Of these, 217 have been reserved for the potential to discharge a full core of fuel; the remaining 736 cells are available for storage. Based on the proposed compact storage scheme each rack cell can accommodate 285 fuel pins. Since there are a maximum of 176 fuel pins per uncompacted assembly (some bundles have up to 16 non-fuel shims), the minimum equivalent number of assemblies which can be accommodated in the 736 cells is:

$$736 \times \frac{285}{176} = 1192 \text{ uncompacted assemblies}$$

or an additional 456 fuel assemblies. The final number will be slightly higher because of the poison shim rods in some assemblies. These non-fuel rods would be disposed of as radwaste. This is enough added space for somewhat more than six additional regular discharges of 72 assemblies, and therefore can assure adequate spent fuel storage for an additional six years or more.

1.1 Mechanical Disassembly

Maine Yankee has working experience in the disassembly of irradiated fuel assemblies. The time required for disassembly is a function of fuel design and tooling constraints. The earlier fuel types (A, B, C and RF) have a lower retention grid and a permanently fastened upper end fitting, and six assemblies of this design have been disassembled at Maine Yankee. While the disassembly is time consuming, experience has shown it to be straight forward. With the pins being handled singly, the per pin removal time averaged ten minutes. This "pin removal time" included the time to position the bundle, remove the upper end fitting and replace the rods.

More recently, Maine Yankee has accomplished reconstitution of the E type fuel. (Fuel types D and beyond are reconstitutable by design.) In the E fuel, 16 burnable poison shim rods per assembly were removed and replaced by 16 water filled shim rods. The operation required about four hours per assembly, three hours was the best time observed. Removal of the upper end fitting consumes about 20 minutes. Based on these experiences, we estimate pin handling time to range from five to ten minutes per pin for the overall extraction and storage operation. Again, this estimate includes not only the time to extract a given pin, but the time also to position the assembly, remove the end fittings, extract the rods, store the rods, and transport the empty cage and end fittings to storage.

1.2 Mechanical Reassembly

Reassembly operations are similar to the reconstitution techniques proven at Maine Yankee. The storage cage or skeleton depicted in Figure 1.1 maintains the assembled fuel pins in a square array on a fixed pitch which has been determined desirable from mechanical, physics and thermal-hydraulic considerations. The fuel compaction cage, with an envelope identical to the standard assembly, is compatible with the present storage racks and standard spent fuel shipping casks.

Rods are withdrawn from the assembly cage and loaded directly into the storage skeleton cage, exactly as in a reconstitution process. The grid-defined rod positioning also lends itself to potentially automated disassembly/loading schemes. For manual rod manipulation, it is a configuration similar to that with which plant personnel have experience.

The intended loading scheme is to fill each storage cage to capacity before placing it into the rack for storage. This limits the number of storage bundles and spent fuel assemblies in a state of partial disassembly.

1.3 Schedule for Fuel Compaction

To accommodate the 72 assembly equilibrium cycle discharge, would require the disassembly and compaction for storage of approximately two hundred assemblies per cycle. This is judged within the capabilities of a well designed, production oriented process for the proven single pin extraction and handling technique. However, full benefit of compact storage would likely require at least limited automation of pin handling due to the already accumulated non-compacted spent fuel assembly backlog.

2.0 THERMAL/HYDRAULIC ANALYSIS

The reader is advised that for convenience, both phases have been evaluated together. Therefore, when considering either compaction or increased spent fuel storage alone, some judgement must be exercised in separating the effects of one phase from those of the other.

2.1 Pool Heat Load

The Spent Fuel Pool (SFP) cooling system design is described in the Final Safety Analysis Report (FSAR), Section 9.8. The pool heat load remains unchanged from the Proposed Change 28 analysis. Therefore, the capacity of the existing cooling system is adequate to maintain pool temperatures below design limits for the compacted fuel storage concept.

2.2 Pin Cooling Analysis

To assure that both compacted storage bundles and fuel assemblies receive adequate local cooling, a detailed thermal/hydraulic analysis was conducted. The criteria for adequate cooling is the preclusion of nucleate boiling in the fuel rack under the maximum pool heat load conditions. This analysis utilized the RETRAN computer program to model a selected row of fuel storage cells as shown in Figure 2.1. Selection of a single row of assemblies, containing either compacted storage bundles or fuel assemblies, provides a conservative thermal evaluation of any possible assembly loading or placement in the spent fuel pool.

In the evaluation of the row of compacted fuel bundles, it was conservatively assumed that the compacted storage bundles were composed of fuel pins taken from fuel assemblies which had been removed from the reactor and cooled for 120 days. (Initially, Maine Yankee does not intend to disassemble any fuel that has been cooled less than three years.) The operating history of these pins while in the reactor was assumed to be infinite. Additionally, a row of freshly discharged fuel assemblies was also addressed. These fuel assemblies were conservatively assumed to cool in the reactor for only four days following shutdown prior to being placed in the pool. Assembly average exposures of 35,000 MWD/MTU, a conservatively high burnup, were assumed.

The maximum pool heat load and bulk pool stabilization temperature occurs when a full core is discharged just after a shutdown from full power. Using Branch Technical Position APCSB 9-2 guidelines (Rev 1, 11/24/75), cooling times were established consistent with the above worst case full core discharge. To bound all future considerations, the full core discharge was assumed to fill the last available spaces in the pool with the assumption that all fuel is compacted once it has cooled for 120 days. Required cooling times for the full core discharge are based on not exceeding a bulk pool temperature of 154oF or a heat load of 22 x 106 BTU/hr, assuming a primary

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component cooling water temperature of 85°F as described in the FSAR. In the event that a full core discharge becomes necessary, pool temperatures will be monitored. The bulk pool temperature will be controlled by simply limiting fuel movement from the reactor to the pool.

The RETRAN model nodalization is shown in Figure 2.2. Each fuel storage cell shown in Figure 2.1 was modeled separately. The upper bulk pool volume, the downcomer region between the edge of the racks and the pool walls, and the region along the pool floor were addressed. Conservative fluid friction and form losses were assumed for all flow paths.

The channel region between each fuel storage cell was assumed to be heated by gamma deposition in the channel walls. Results indicate that all bundles and assemblies receive adequate flow. The outlet conditions of the hottest compacted bundle and the hottest discharged fuel assembly are shown in Table 2.1.

The maximum void fraction in the assemblies and the maximum void fraction in the space between fuel storage cells are zero. Thus, boiling is not of concern as a source of moderator density variations in reactivity calculations.

2.3 Loss of Forced Pool Circulation

If all forced circulation cooling flow to the pool is lost, the large volume of pool water (363,000 gal.) provides a heat sink which allows time for corrective action. The minimum time for the pool water to reach the saturation temperature is 7.8 hours, for the maximum design condition heat load of 22×10^6 BTU/hr. In the event that normal cooling flow is lost, on either shell or tube side of the pool heat exchanger, back-up cooling is available. The plant fire protection system, with two 2,500 gpm pumps, can be connected to the pool heat exchanger shell side via two emergency cooling connections (designated ECC on FSAR Figure 9.8-1). Heat exchanger tube side flow is provided by two (2) 750 gpm pumps. In the event that one (1) pump is lost, the alternate pump is available to provide flow. Additionally, emergency cooling connections are provided at the suction and discharge of the pumps to permit connection of a portable pump.

2.4 Loss of Pool Water

Loss of pool cooling water as a result of failure of the coolant inlet and outlet lines exterior to the pool is addressed in the response to Question 9.15 in Amendment 18 to the FSAR. It was determined that it is not possible to drain the fuel pool by siphoning as a result of a rupture of any line normally connected to the fuel pool.

3.0 CRITICALITY ANALYSIS

The reader is advised that for convenience, both phases have been evaluated together. Therefore, when considering either compaction or increased spent fuel storage alone, some judgement that must be exercised in separating the effects of one phase from those of the other.

3.1 Methodology and Assumptions

Criticality calculations were performed using the methodology* described and utilized in Proposed Change No. 28 and its Supplements. Proposed Change No. 28, supporting the conversion of the spent fuel racks at Maine Yankee to the high density, boron poison rack design received NRC approval in October, 1975.

The compacted fuel storage rack configuration is comparable to that considered in Proposed Change No. 28 with the significant difference being the placement of fuel pins in the compacted cage. The fuel pin pitch of 0.475 inches established by the storage grid structure was utilized in the criticality calculation and represents the nominal case. The criticality calculation assumed the mechanical uncertainties to be at "worst case" conditions. That is, the minimum water inside the flux trap and the minimum BORAL plate thickness are assumed. The nominal case also utilizes the 95/95 confidence level Boron-10 content as described in Brooks and Perkins report No. 540 (0.007235 atoms B-10/bn-cm). Additionally, a number of other conservative assumptions were used in this study, including the following:

i) Unirradiated (Fresh) 3.2 w/o U235 fuel

ii) No soluble poison in the pool water

iii) No axial or radial leakage from the rack

iv) Calculations were done at the most reactive pool temperature anticipated (68°F)

v) A calculational uncertainty of 3% $\Delta\rho$ was added to all results based on Monte Carlo calculations and the analysis of critical experiments.

The PDQ07 program, utilizing four energy groups with LEOPARD cross section input and an X-Y representation of the racks was used to determine

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*YAEC-1090, "Criticality Calculations on Maine Yankee Spent Fuel Racks Containing Boron", November 1975.

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$K_{effective}$. The accuracy of this representation was verified by comparison to KENO-II, Monte Carlo calculations.

3.2 Nominal Conditions

The sensitivity of rack $K_{effective}$ to the storage lattice pitch and the density of the Boron-10 atoms is presented in Figures 3.1 and 3.2. No calculational uncertainty has been added to these values. The $K_{effective}$ of the fully loaded nominal compacted pin rack, including calculational uncertainty, is 0.604, well below the value of the same racks loaded with fuel assemblies. The reason for this large difference (0.604 versus 0.773) is the under moderation of the racks storing compacted fuel. This result is well below the values presented in Proposed Change No. 28 and/or the existing Facility License.

3.3 Off-Nominal Conditions

Off-nominal conditions were reviewed for the compacted pin storage rack concept. One area which needed to be considered was the $K_{effective}$ of partially loaded cages stored in a rack. To bound all possible loading configurations, the most reactive fuel pin pitch was determined. A pin pitch of 0.65 was used in calculation of new fuel cross sections. These cross sections were then assigned to the fuel in the nominal 68°F calculation. This configuration is calculated to be 33.9%$\Delta\rho$ more reactive (or $K_{effective}$ = 0.760) than the nominal design. This limiting case for off-nominal conditions demonstrates the low $K_{effective}$ of the racks storing compacted fuel under any loading configuration.

3.4 Accident Conditions

A review of accident conditions for compacted fuel storage reveals that the information presented in Proposed Change No. 28 and/or the existing Facility License is bounding and need not be addressed in this submittal.

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4.0 MECHANICAL, MATERIAL, AND STRUCTURAL CONSIDERATIONS

The reader is advised that for convenience, both phases have been evaluated together. Therefore, when considering either compaction or increased spent fuel storage alone, some judgement must be exercised in separating the effects of one phase from those of the other.

Maine Yankee proposed to change the spent fuel pool storage design with Proposed Change 28 submitted to the NRC March 27, 1975. After an extensive review, the NRC issued Amendment No. 11 to License No. DPR-36 which allowed Maine Yankee to replace the existing storage racks with high density poison racks.

From a mechanical, material and structural standpoint, Maine Yankee has re-evaluated this design to reflect the compacted fuel storage concept and at the same time incorporate the April 14, 1978 NRC guidelines for fuel storage modifications*.

4.1 Mechanical and Material Considerations

The design concept, materials, manufacturing techniques, along with the quality assurance and surveillance programs were all presented in detail in the original submittal and subsequent supplements. The compacted fuel storage concept will not impact these items and, therefore, does not require their re-evaluation.

4.2 Structural Considerations

Under the compacted fuel storage concept the effective weight of a fuel assembly in a storage cell increases. Therefore the pool and rack structures are re-evaluated as described below to ensure their structural integrity under design loads.

4.2.1 Spent Fuel Pool

The 6 foot thick reinforced concrete floor of the Maine Yankee fuel storage pool rests on bedrock. The increase in weight supported by the floor utilizing the pin storage concept would be distributed in such a manner to assure allowable floor loadings. The increase in weight is well within the static supporting capacity of the pool floor. The storage pool will be reanalyzed incorporating the compacted fuel concept to ensure conformance to the original Seismic I design requirements.

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*"Review and Acceptance of Spent Fuel storage and Handling Applications," NRC Publication dated April 14, 1978.

4.2.2 Cask Drop

All of the conclusions reached concerning a dropped cask accident for the existing Maine Yankee fuel storage pool are equally applicable for the new design.

If the cask should drop while being handled over the pool, the pool floor would not be damaged to the extent that makeup capability could not be assured or that resultant flooding could cause critical systems to become inoperable. Electrical interlocks prevent the crane from passing over spent fuel. The proposed modification will not increase the area assigned to fuel storage. Therefore, if a cask drop should occur, the cask will not contact spent fuel assemblies.

Our review of the cask drop accident leads us to conclude, therefore, that there is no increase in the probability or consequences of a fuel cask drop accident with the compacted fuel storage design.

..3 Storage Racks

The original structural design analysis for the high density storage racks has been re-evaluated incorporating the pin storage concept. The resulting stress levels indicated that the present racks with minor modifications will accommodate the increased weight without exceeding any original design basis.

In addition, Maine Yankee has reviewed the April 14, 1978 NRC guidelines suggested by the NRC concerning the structural analyses for new storage rack submittals. Following the intent of these guidelines, the racks will be analyzed for seismic and impact loads.

Seismic excitation will be imposed simultaneously in the N-S, E-W, and vertical directions. Inertia, rack to rack impact, and fuel bundle impact effects will be considered. Responses will be combined and load combinations made in accordance with the guidelines. All stresses shall remain within the applicable code limits. Maine Yankee will perform all analyses and make any required rack modification necessary to satisfy the intent of each guideline.

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5.0 RADIOLOGICAL EVALUATION

The reader is advised that for convenience, both phases have been evaluated together. Therefore, when considering either compaction or increased spent fuel storage alone, some judgement must be exercised in separating the effects of one phase from those of the other.

Radiological considerations of compaction and increased spent fuel storage have been reviewed under the guidelines of the April 14, 1978 NRC document.

5.1 Solid Waste

The present annual quantity of solid radioactive wastes generated by the SFP purification system is approximately 20 ft^3. It is expected that the increase in solid wastes as a result of the expanded capacity will not be significant i.e., less than 10% increase.

5.2 Gaseous Waste

Data regarding Krypton-85 measured from the fuel building ventilation system are not available for Maine Yankee. Airborne Kr-85 releases from MY vent stack (which would include releases from the fuel building) for 1976 and 1977 are 68.91μcuries and 17.81μcuries respectively.

5.3 Dosage to Personnel

5.3.1 The principal radio-nuclides present in detectable concentrations in the SFP water at MY are ^{58}Co, ^{134}Cs, and ^{137}Cs. The concentrations of each of these detectable nuclides is approximately 3×10^{-3} μCi/ml.

5.3.2 Dose rates as measured in the spent fuel pool area at MY are typically on the order of 1 mrem/hr. Extensive calculations were performed to bound the maximum expected increase in radionuclide concentrations and resulting dose rates as a result of storage capacity increases for Yankee Rowe. The expansion ratio for Yankee Rowe was 2.3 compared to 1.6 for Maine Yankee. The calculated incremental exposure rate due to increased radionuclide concentrations in the YR case was less than 1/10 mrem/hr. based on successive discharges of leaking fuel (1% failed fuel). Therefore, the incremental exposure rate in the SFP area due to the increased storage at MY is not expected to exceed 10% of the present level of approximately 1 mrem/hr.

5.3.3 Routine air sampling in the spent fuel pool area indicates no detectable activity i.e., 10^{-11} μCi/cc.

5.3.4 As stated above, routine airborne activity samples indicate no detectable activity in the SFP area. It is expected that any increase in airborne activity in the SFP area as a result of the expension would not be significant i.e., less than 10% of present levels. Extensive calculations were performed on the increase in activity in spent fuel pools in conjunction with the expansion of fuel storage capacity at Yankee Rowe and Vermont Yankee. Using an extremely unlikely set of occurences, i.e., the successive unloading of full cores of leaking fuel (1%) into the SFP until it was filled to its enlarged capacity, the increment in radionuclide concentrations in the pool water due to the expanded capacity were calculated to be less than 10% of the normal measured concentrations. The associated incremental exposure rate in the SFP area and at the site boundary would be extremely minimal.

5.3.5 The changing of the SFP demineralizer resin at MY is accomplished remotely. It is expected that any increase in man-rem associated with the operation as a result of the expansion will be minimal. The filter associated with the SFP demineralizer has required changing approximately four times since MY began power operation in 1972. The annual man-rem exposure from the change out of the SFP filter is estimated to be less than 0.050 man-rem. The increase in annual man-rem burden from more frequent changing of this filter is estimated to be less than 0.005 man-rem.

5.3.6 The buildup of crud products (e.g., ^{58}Co, ^{60}Co) along the sides of the spent fuel pool has not created any measurable difference in exposure rates along the sides of the pool. However, if crud buildup were to become an exposure problem, underwater vacuum cleaning could be used to remove the buildup. Such methods have been used successfully to clean both the sides and bottom of the spent fuel pool at Yankee Rowe. The cleaning operations in the Yankee Rowe SFP were conducted to prepare for in pool diving necessary to accomplish certain modifications and not because of an exposure problem at the pool edge.

5.3.7 The expected total annual man-rem received by personnel occupying the fuel pool area during a year which included extensive fuel sipping is estimated to be considerably less than 10 man-rem. Levels of radiation in the SFP area at MY are typically on the order of 1 mrem/hr. while airborne levels are not detectable. As a result, the spent fuel pool area at MY does not present serious radiation protection problems. Air and water sampling together with direct radiation surveys are routinely conducted on a weekly basis in the SFP area. During refueling operations and whenever extensive work is done in the SFP area, continuous air sampling is performed in conjunction with more frequent Health Physics surveys.

5.3.8 Even with the effective expansion of the spent fuel pool permitted by fuel compaction, it is not expected that any changes in the radiation protection program at MY will be necessary to maintain exposures as low as reasonably achievable.

6.0 COST/BENEFIT ASSESSMENT

The reader is advised that for convenience, both phases have been evaluated together. Therefore, when considering either compaction or increased spent fuel storage alone, some judgement must be exercised in separating the effects of one phase from those of the other.

6.1 Need for Increased Storage Capacity

The Maine Yankee Nuclear Power plant began operation in late 1972. The core of the reactor contains 217 fuel assemblies, and the plant spent fuel storage pool was originally designed to contain 318 spent fuel assemblies. It was expected that spent fuel would be shipped off site for reprocessing after a short cooling period, so a large storage capability was not needed.

In 1974 it was becoming apparent that availability of commercial fuel reprocessing would be significantly delayed and that reprocessing could offer no near-term relief for a continuing lack of storage capacity. In March of 1975, Maine Yankee forwarded to the NRC a finding under 10CFR 50.59 to allow expansion of storage capacity from 318 to 953 assemblies by installation of new poisoned storage racks. This modification was deemed by the NRC to require a license amendment which was granted in October of 1975. Fabrication and installation of the new racks, which have a center-to-center spacing of 12 inches versus 20 inches in the original racks, is taking place in three phases which are scheduled for completion in 1979.

There have been two subsequent spent fuel discharges since 1975 - one in 1977 and one in July of 1978. These discharges bring the total number of spent fuel assemblies residing in the Maine Yankee pool to 433. Thus, there are currently 520 locations for spent fuel assemblies remaining in the pool.

The projected discharge/refueling schedule for Maine Yankee is shown in Table 6.1. This table shows that full core reserve will be lost in 1984 and full reload discharge capability will be lost in 1987 with the current storage pool configuration. In order to assure continued economic and reliable service to its electricity customers, Maine Yankee Atomic Power Company believes that maintenance of full core reserve is advisable. Loss of discharge capability, of course, would force a shutdown of the reactor.

The need to provide more storage space for spent fuel at Maine Yankee has been precipitated by changes in Federal government policy. The current Federal policy of indefinitely deferring the reprocessing of LWR fuel effectively precludes the shipment of spent fuel from Maine Yankee to an off-site reprocessing plant as had been originally planned. At this time, it is not possible to predict when reprocessing will eventually be allowed. However, even if the policy were changed now, the need to complete the GESMO

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proceeding, complete the construction and licensing of reprocessing plants, and develop a transportation system to move spent fuel would likely postpone the onset of significant reprocessing activity until the mid-1980's. At that time, the demand for reprocessing will far exceed the limited facilities available. Maine Yankee has no contractual arrangements for reprocessing.

It has been proposed by the current Administration that spent fuel be disposed of directly in a waste repository rather than being reprocessed. Maine Yankee believes that such a policy is undesirable in view of the large potential energy resource available in spent fuel as uranium and plutonium. Nevertheless, the possibility of disposing of spent fuel has been considered. In February of this year, the Department of Energy announced that operation of the first Federal waste repository for spent fuel could not be expected until the 1988-1993 timeframe. Therefore, a repository will not be available in time to meet the storage requirements of Maine Yankee.

The Department of Energy (DOE) has also proposed a new spent fuel policy under which the government would offer to take title to spent fuel upon payment of a one-time charge and delivery to a government-approved storage site. Unfortunately, no such storage facilities currently exist, and there is yet no legislative authority or funding to provide them. The DOE has suggested that they will try to provide appropriate facilities by 1983-85, but at this time, such a possibility can best be characterized as speculation. Under these circumstances, Maine Yankee believes that it would not be prudent to rely on the Federal government to provide spent fuel storage space needed for continuing operation.

We conclude that neither reprocessing nor waste disposal at a repository nor storage at a government facility can be relied upon to correct a lack of storage space during the 1980's. Thus, there is a clear need for Maine Yankee to pursue other alternatives to provide the necessary storage space.

Compaction of spent fuel assemblies will satisfy the need for more storage at Maine Yankee, resulting in space for an additional 436 assemblies or somewhat more than six regular discharges of 72 assemblies. This space is adequate to assure that loss of full core reserve will not occur until 1992 as can be seen by examining Table 6.1. Additional regular refuelings could continue until the year 1996 (without the capability for discharge of a full core) before loss of refueling capability due to a full pool.

While fuel compaction will result in the capability to store more spent fuel on site, it will not necessarily extend the time period that fuel is kept on site. Spent fuel could be sent off-site for reprocessing or off-site storage after the government makes a firm decision on the reprocessing option and a facility is available to accept the fuel.

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6.2 Costs of Spent Fuel Compaction

The only significant capital expense associated with compaction is for increased seismic restraints on the existing racks; it is estimated that this will cost less than $100,000 in total. Only minor investments in fuel handling equipment are expected.

The actual costs of assembly disassembly are estimated as follows. A new compacted cage consisting of several grids, four tie rods, and upper and lower tie plates should cost significantly less than the cages for the standard fuel assemblies. Without the requirement to sustain in-core loadings under normal and accident conditions, the mechanical requirements and, therefore, costs are much reduced. Specifically, the grids become simple egg crates without springs, the guide tubes are replaced by stainless steel rods, and the complex end fitting castings become flat plates. Standard replacement cages are estimated to cost $6000 each. The cost of the compacted cage is estimated to be $4,000; this is equivalent to $2470 per uncompacted assembly.

Experience at Maine Yankee with fuel bundle reconstitution suggests that a complete assembly disassembly and reassembly into a compact array can be accomplished within 30 hours by two men. Therefore, the labor costs involved will approximate 30 hours x 2 men x $15/man-hour = $900 per uncompacted assembly. The old fuel cages must be cut up and shipped off-site for disposal; this process is expected to cost about $1200 per uncompacted assembly (10 scrap assemblies per load and cost of $12,000 per load, consisting of $6000 for cut-up and shipping and $6000 for burial). The total of these three costs: new cage, labor, and shipping/burial is therefore $4570 per uncompacted assembly. The cost of compacting 1192 assemblies is then about $5,450,000.

The total project cost is estimated at $5,550,000. If this cost is allocated only to those additional 456 assemblies which can be stored following compaction, then the cost becomes about $12,200 per additional stored assembly.

No significant plant operation cost increases are expected as a result of spent fuel compaction.

6.3 Resources Committed

Spent fuel compaction will not result in any irreversible and irretrievable commitments of water, land, and air resources. No additional allocation of land would be made; the land area now used for the spent fuel pool will be used more efficiently by increasing the density of fuel storage.

The only irreversible commitment of materials will be material used for seismic restraints and for new compact fuel rod cages. This material, mainly stainless steel, should not exceed 100,000 pounds. This is an insignificant quantity when compared to the U.S. annual consumption of stainless steel which is about 10^{11} pounds.

6.4 Consideration of Alternatives

Alternatives that could alleviate the current need for additional spent fuel storage capacity have been evaluated. Consideration was given to availability, environmental impact, cost and benefits. Discussion of these alternatives follows:

a) Shipment to a Reprocessing Plant

As described previously in Section (6.1), the need for additional spent fuel storage at Maine Yankee is due to a Presidential decision to indefinitely defer reprocessing and to a lack of readily available reprocessing capability. However, there is potential storage space available at existing U.S. reprocessing plants. Unfortunately, none of this space is available for storage of Maine Yankee fuel. The Nuclear Fuel Services facility in West Valley, New York is not receiving any additional spent fuel. The General Electric facility at Morris, Illinois is accepting spent fuel only from its previous reprocessing customers who have contracts; Maine Yankee has no contractual arrangements with General Electric. The Allied General Nuclear Services (AGNS) facility at Barnwell, South Carolina is not licensed to accept spent fuel. A license is not expected in the near future and even if one were obtained, it is likely that AGNS would restrict storage to those customers who previously had reprocessing contracts; Maine Yankee has no contracts with AGNS. It is extremely unlikely that any new reprocessing plants would be constructed in time to receive Maine Yankee fuel as needed in the 1980's.

b) Shipment to an Independent Spent Fuel Storage Facility (ISFSF)

At the present time, there are no existing ISFSF's. As mentioned in Section (6.1), the U.S. DOE has suggested that it may build such facilities for the mid 1980's, but no firm commitment has yet been made. The Tennessee Valley Authority (TVA) has suggested that it may build an ISFSF for its own spent fuel and make available some space for other utilities fuel; this project is also very speculative at this time. No other private ISFSF's are expected to be built in the near future.

c) Shipment to Another Reactor Site

Storage of spent fuel at another reactor facility would be physically possible but was found not to be a realistic alternative. Maine Yankee Atomic Power Company does not own any reactor except Maine Yankee. All the other operating reactors in New England have storage problems that are similar to that of Maine Yankee, although the need varies by several years at the various plants. Because storage of Maine Yanke spent fuel at a different "host" reactor would serve to exacerbate the "host's" storage problem, other reactors are unwilling to accept Maine Yankee fuel. A similar situation exists outside of New England. It is not deemed prudent to plan on shipping fuel to a reactor under construction or planned because of the potential for significant construction delays or cancellation which would effectively eliminate the planned storage space.

d) Increase the Size of the Existing Maine Yankee Spent Fuel Pool

The possibility of physically expanding the size of the Maine Yankee spent fuel pool was briefly examined and found to be impossible due to the inability to move existing structures and hardware.

e) Replace the Existing Spent Fuel Racks with Even More Tightly Spaced Racks

At the time that Maine Yankee first reracked its spent fuel rack pool, a 12-inch center-to-center spacing was chosen. Such spacing is conservative based on improved neutronics calculation methods. It would theoretically be possible to rerack a second time with about 10-3/4 inch center-to-center spacing and still meet NRC design criteria for spent fuel pools. This reduced spacing could add only about 232 assembly spaces; thus loss of full core reserve would only be postponed about 3 years to 1987 as compared to 1993 with compacted fuel storage. The cost of this second rerack would exceed $3.5 million. Thus, the cost per additional stored assembly would be more than $15,000; this far exceeds the cost of the proposed modification. The resources of stainless steel, boron carbide and aluminum consumed by another rerack would also exceed those from the proposed modification. Reracking would, therefore, only receive consideration in the event insurmountable, non-technical obstacles are projected in developing the compacted fuel storage concept.

f) Construction of an Additional On-Site Storage Pool

Construction of a new spent fuel storage pool on the Maine Yankee site was considered as another alternative. A new pool of about 275 MTU capacity could provide storage through 1996, comparable to the proposed modifications. Such a facility is estimated to cost at least $27 million. Since the facility would contain about 710 assemblies, the cost per added assembly is around $38,000. This alternative is seen to have a capital cost that is extremely large compared to other alternatives. Furthermore, annual operating costs approaching $1 million would also be involved, and movement of fuel into the new pool would require on-site transport by cask. Clearly, a new pool would have much larger environmental impacts and would require a much larger commitment of resources than the compaction procedure.

g) <u>Shutdown of the Reactor</u>

In the event that compaction (or some other increase in storage) are not carried out, the plant would have to be shut down by 1987, at least until some spent fuel could be shipped out of the pool. Replacement energy would have to be generated from other existing power plants in New England, most of which are nuclear or oil-fired. The average cost of replacement power is currently about 23 mills/kwh which is equivalent to about $137 million per year for the 850 MWe Maine Yankee plant at 80% capacity factor. Of course, by 1987 the cost of replacement power can be expected to be even higher. Since a large fraction of the replacement power must be provided by imported oil, the environmental effects (air pollution, oil spills, transportation accidents, etc.) are significantly worse than continued nuclear generation. Furthermore, it is contrary to national policy to rely on imported energy resources. Burning of oil results in an irretrievable commitment of resources which cannot be used by future generations. The alternative of shutting down the plant is obviously unacceptable.

In summary, spent fuel compaction has the lowest cost per additional assembly space, has no major front end capital costs, represent the smallest commitment of irretrievable resources, and has the smallest environmental impact. Implementation of compact storage will not foreclose future alternatives since actual disassembly/reassembly operations can be postponed until needed. If other alternatives, such as off-site shipment become available in the near term, they could still be exercised without incurring major cost penalties.

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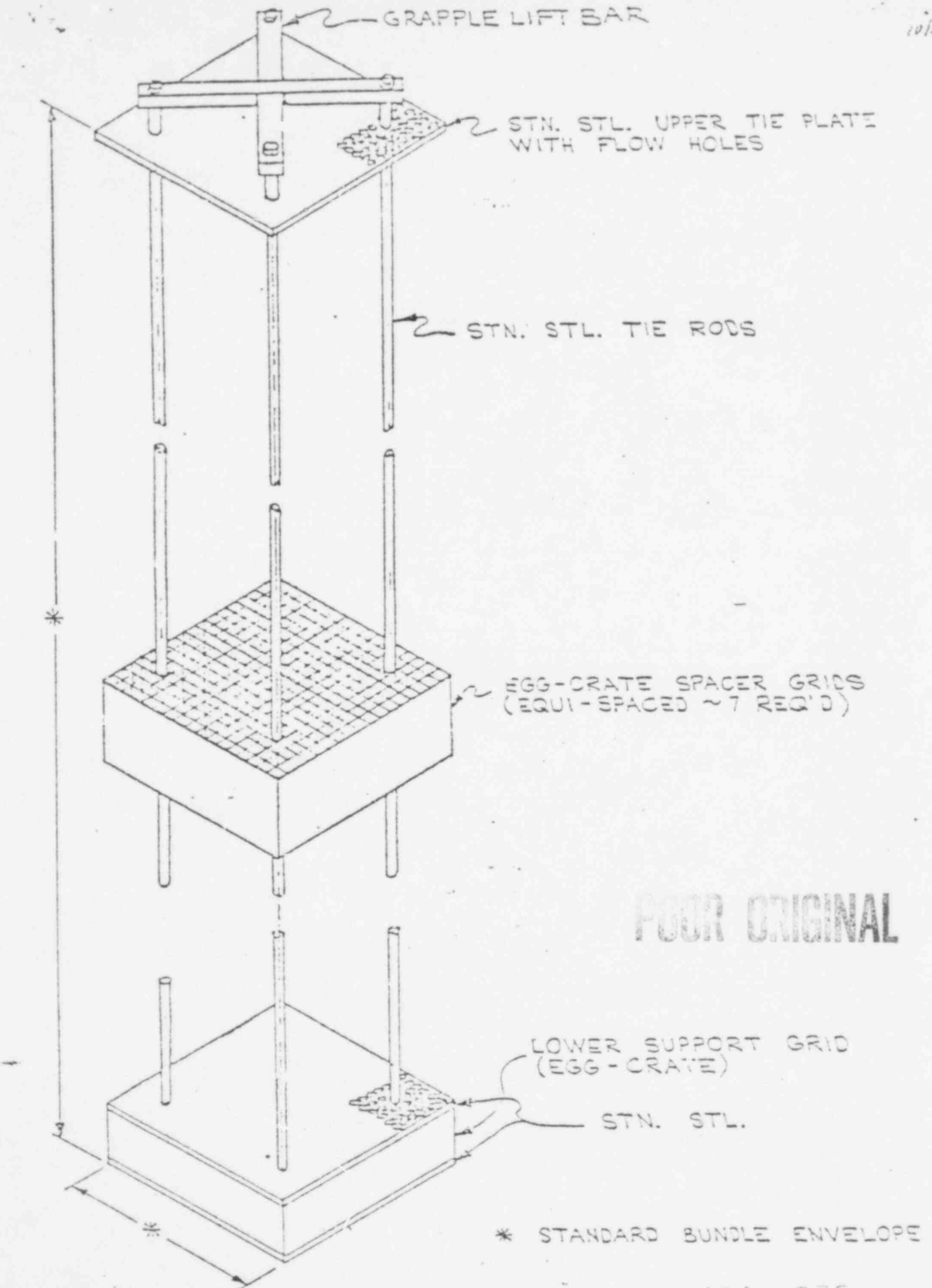

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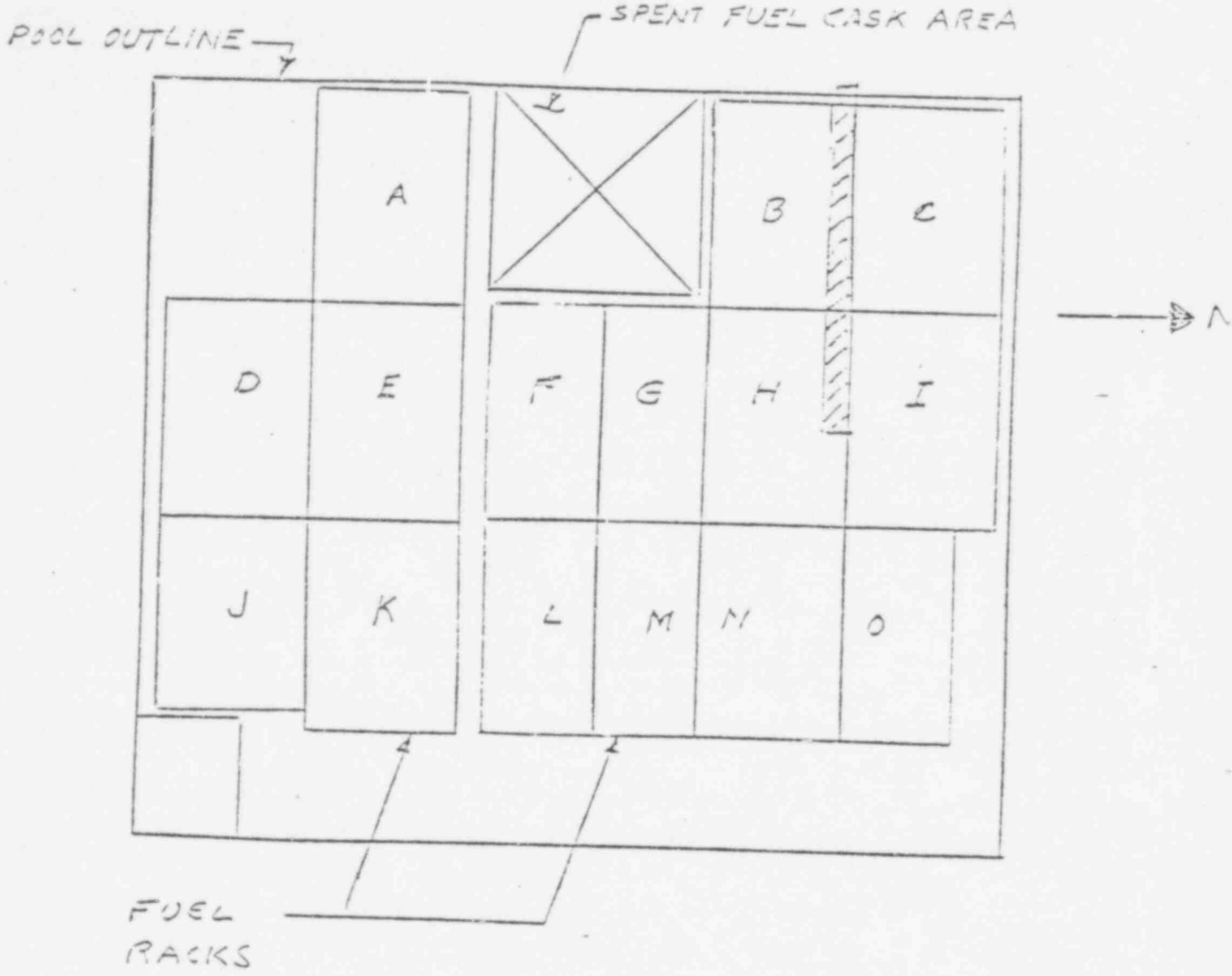
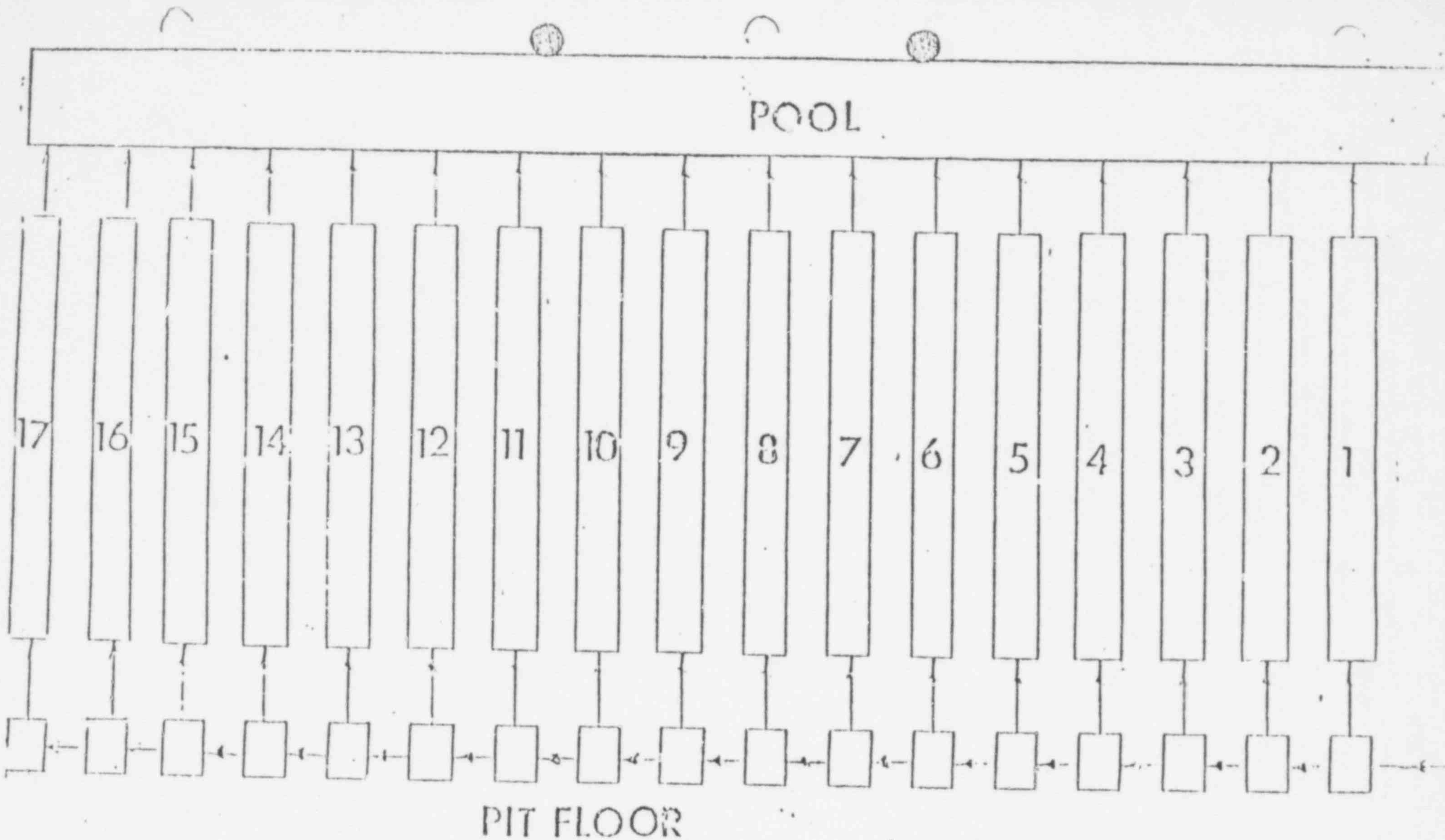
FUEL PIN STORAGE SKELETON
CONCEPTUAL DESIGN

TABLE 2.1

	Maximum Outlet Temp. (°F)	Maximum Enthalpy (BTU/lbm)	Void Fraction
Hottest Pin Bundle	207	175	0.
Hottest Fuel Assembly	220	188	0.
Saturation Condition at Outlet Height	234	202.4	-



CROSS-HATCHING INDICATES ROW MODELLED



MAINE YANKEE

MODEL FOR THERMAL HYDR.
ANALYSIS OF MY SPENT FUEL

FUEL BUNDLES

DOWN FLOW ARE/

FIGURE 2.2

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FIGURE 3.1

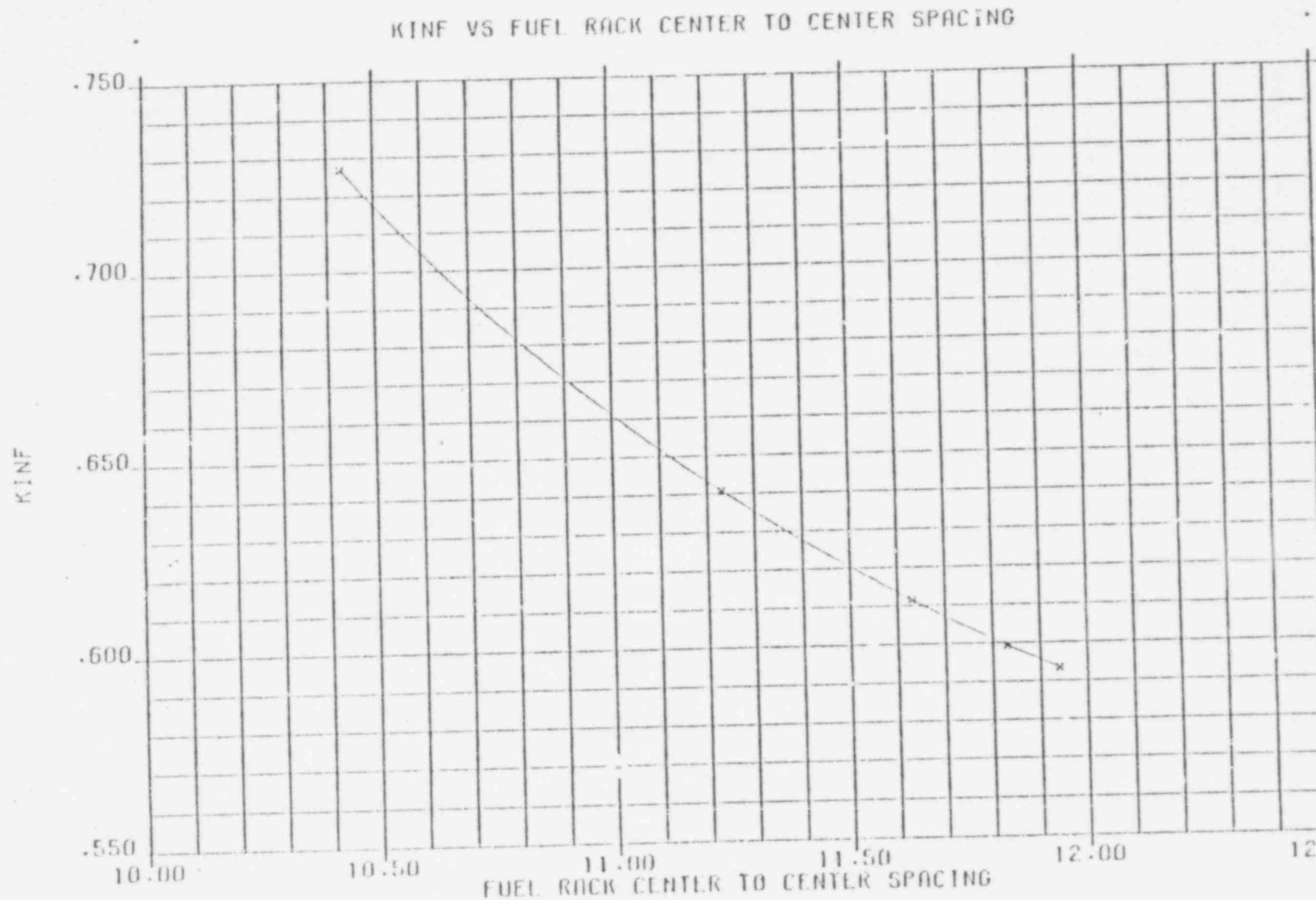

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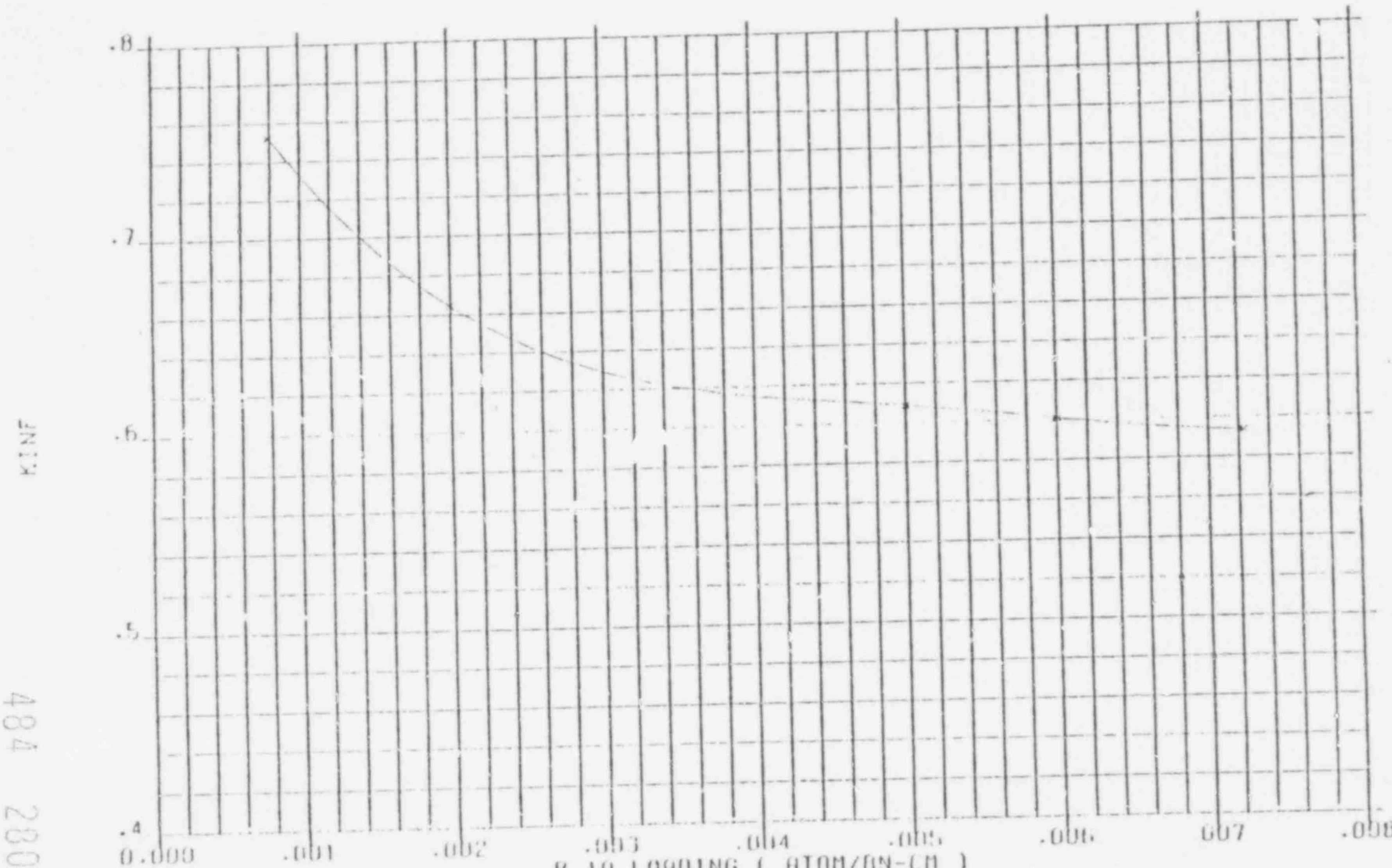
FIGURE 3.2

KINF VS B-10 LOADING



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