Attachment A

| | Remove | | Insert |
|---|---|---|---|
| page | 3.11-2 | page | 3.11-2 |
| | 3.11-4 | | 3.11-4 |
| | 5.4-2 to 5.4-4 | | 5.4-2 to 5.4-4 |
| figure | 5.4-1 | figure | 5.4-1 |

e. Charcoal adsorbers shall be installed in the ventilation system exhaust from the spent fuel storage pit area and shall be operable.

3.11.2 Radiation levels in the spent fuel storage area shall be monitored continuously.

3.11.3 The trolley of the auxiliary building crane shall never be stationed or permitted to pass over storage racks containing spent fuel.

3.11.4 The spent fuel pool temperature shall be limited to 150°F.

3.11.5 The spent fuel shipping cask shall not be carried by the auxiliary building crane, pending the evaluation of the spent fuel cask drop accident and the crane design by RG&E and NRC review and approval.

3.11.6 The restriction of 3.11.3 above shall not apply to the movement of cannisters containing consolidated fuel rods if the spent fuel rack beneath the transported cannister contain only spent fuel that has decayed at least 60 days since reactor shutdown.

Basis:

Charcoal adsorbers will reduce significantly the consequences of a refueling accident which considers the clad failure of a single irradiated fuel assembly. Therefore, charcoal adsorbers should be employed whenever irradiated fuel is being handled. This requires that the ventilation system should be operating and drawing air through the adsorbers.

The desired air flow path, when handling irradiated fuel, is from the outside of the building into the operating floor area, toward the spent fuel storage pit, into the area exhaust ducts, through the adsorbers, and out through the ventilation system exhaust to the facility vent. Operation of a

The spent fuel pool temperature is limited to 150°F because if the spent fuel pool cooling system is not at that temperature, sufficient time (approximately 7 hours) is available to provide backup cooling, assuming the maximum anticipated heat load (full core discharge & previously stored fuel), until a temperature of 180°F is reached, the temperature at which the structural integrity of the pool was analyzed and found acceptable.

The requirement of 3.11.6 insures that should a handling accident occur during the movement of a consolidated fuel cannister (as described in 5.4.) the dose at the exclusion area boundary would satisfy the requirements of 10CFR100.

<u>References</u>

(1) FSAR - Section 9.3-1

(2) ANS-5.1 (N 18.6), October 1973

5.4.4 Cannisters containing consolidated fuel rods may be stored in either Region 1 or 2 provided that:

a. the average burnup and initial enrichment of the fuel assemblies from which the rods were removed satisfy the requirements of 5.4.2 and 5.4.3 above, and

b. the average decay heat of the fuel assembly from which the rods were removed is less than 2150 BTU/hr

5.4.5 The requirements of 5.4.4a may be excepted for those consolidated fuel assemblies of Region RGAF2.

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

Basis

The center to center spacing of Region 1 insures that Keff $\leq$ 0.95 for the enrichment limitations specified in 5.4.2[1], and for a postulated missile impact the resulting dose at the EAB would be within the guidelines of 10CFR100[2].

In Region 2, Keff $\leq$ 0.95 is insured by the addition of fixed neutron poison (boraflex) in each of the Region 2 storage locations, and a minimum burnup requirement as a function of initial enrichment for each fuel assembly design. The 60 day cooling time requirement insures that for a postulated missile impact the resulting dose at the EAB would be within the guidelines of 10CFR100.

The two curves of Figure 5.4-2 divide the fuel assembly designs into two groups. The first group is all fuel delivered prior to January 1, 1984. This incorporates all Exxon and Westinghouse HIPAR designs used at Ginna.[4] The second curve is for the Westinghouse Optimized Fuel Assembly design delivered to Ginna beginning in February 1984.[3]

The assembly average burnup is calculated using INCORE generated power sharing data and the actual plant operating history. The calculated assembly average burnup should be reduced by 10% to account for uncertainties. An uncertainty of 4% is associated with the measurement of power sharing. The additional 6% provides additional margin to bound the burnup uncertainty associated with the time between measurements and updates of core burnup. The curves of Figure 5.4-2 incorporate the uncertainties of the calculation of assembly reactivity.[3]

The calculations of fuel assembly burnup for comparison to the curves of Figure 5.4-2 to determine the acceptability for storage in Region 2 shall be independently checked. The record of these calculations shall be kept for as long as fuel assemblies remain in the pool.

The fuel storage cannisters are designed so that, normally, they can contain the equivalent number of fuel rods from two fuel assemblies in a close packed array, and can be stored in either Region 1 or Region 2 rack locations. The close packed array will insure the $K_\infty$ of the rack configuration containing any number of cannisters will be less than that for stored fuel assemblies at the same burnup and initial enrichment. The exception

of paragraph 5.4.5 is possible because the consolidated configuration is substantially less reactive than that of a fuel assembly. The maximum decay heat requirement will insure that local and film boiling will not occur between the close packed fuel rods if the pool temperature is maintained at or below 150°F. The decay heat of the assembly will be determined using ANS 5.1, ASB 9-2 or other acceptable substitute standards.

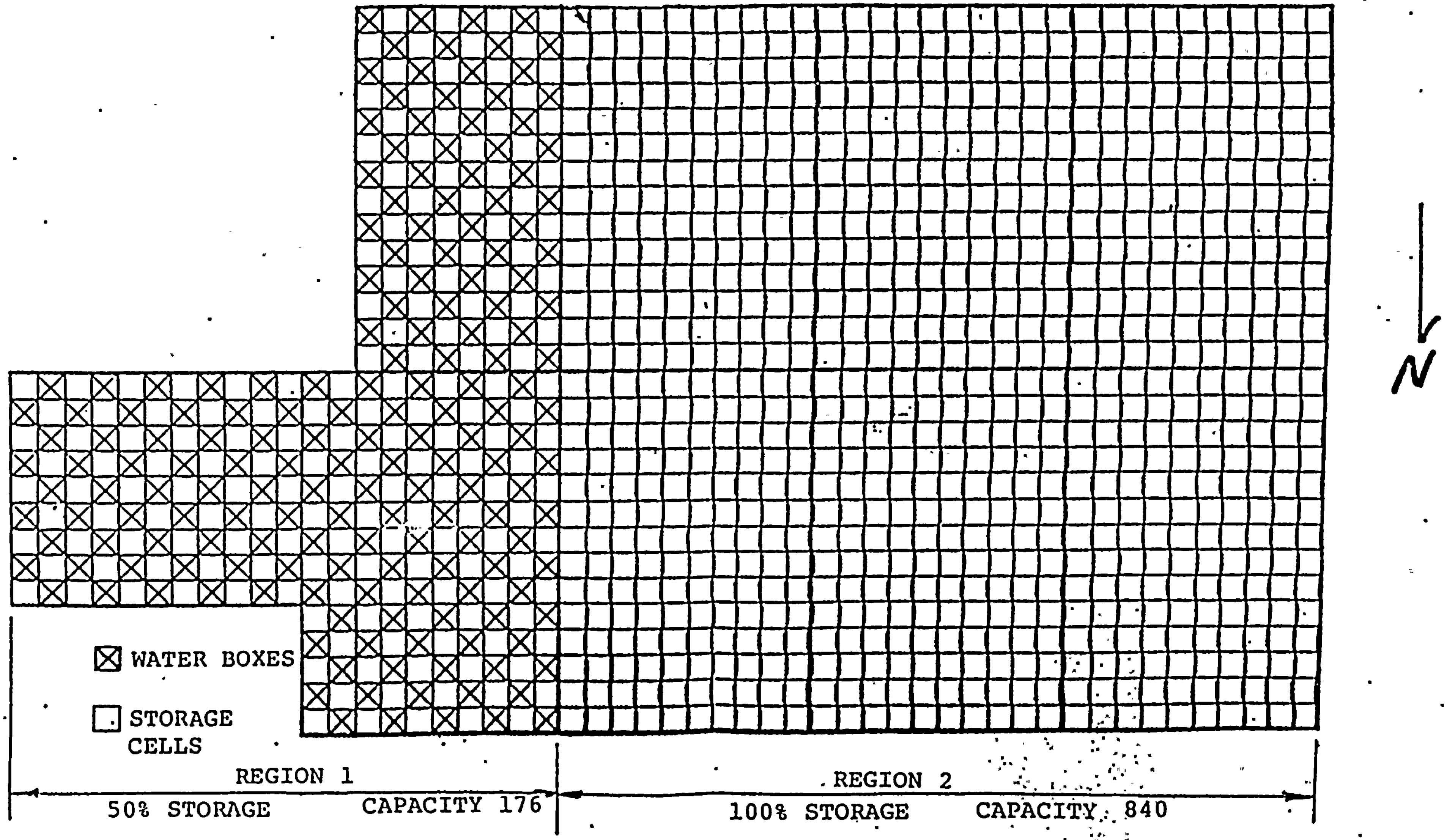
With the addition of the storage of consolidated fuel cannisters, the theoretical storage capacity of the pool would be increased to 2032 fuel assemblies (2x1016). However, due to limitation on the heat removal capability of the spent fuel pool cooling system, the storage capacity is limited to 1360 fuel assemblies.[5]

<u>References</u>


1. Letter, J.E. Maier to H.R. Denton, January 18, 1984.
2. Letter J.E. Maier to H.R. Denton, January 18, 1984.
3. Criticality Analysis of Region 2 of the Ginna MDR Spent Fuel Storage Rack, Pickard, Lowe and Garrick, Inc. March 8, 1984.
4. Letter, T.R. Robbins, Pickard, Lowe and Garrick, Inc. to J.D. Cook, RG&E March 15, 1984.
5. Letter, D.M. Crutchfield to J.E. Maier, November 5, 1981.

FI[illegible]RE 5.4-1

SPENT FUEL STORAGE RACKS




*TOTAL CAPACITY 1360 Fuel Assemblies

Total Capacity includes provisions for storage of consolidated fuel.

PROPOSED

Attachment B

In 1973 Rochester Gas and Electric (RG&E) shipped 121/spent fuel assemblies to the Nuclear Fuel Services (NFS) reprocessing facility at West Valley, New York. Of these fuel assemblies, NFS took title to 40 while for the remaining 81 title was retained by RG&E. It is the intent of RG&E to have this fuel consolidated into storage cannisters with the fuel rods from two assemblies stored in one cannister. This activity will be performed as part of a research and development project funded by RG&E, the Empire State Electric Energy Research Corporation (ESEERCO) and EPRI.

This submittal addresses the applicable safety issues of an increase in spent fuel storage capability through the storage of consolidated fuel. The general outline of the January 18, 1979 NRC guidance will be followed[3]. It should be noted that this submittal only addresses the receipt and storage of consolidated fuel, not the consolidation process itself. The fuel will be consolidated at West Valley, not at Ginna Station. Prior to any consolidation activity at Ginna, another submittal for NRC staff review will be provided.

This safety analysis is separated into 7 sections.

1. A Description of the Cannisters and Fuel Configuration
2. Nuclear
3. Thermal Hydraulic
4. Mechanical, Material, Structural
5. Cost Benefit Assessment
6. Radiological Evaluation
7. Accident Evaluation

1. Description of Cannisters and Fuel Configuration.

Reference 1, along with subsequent responses to NRC staff questions provides a complete description of the Ginna spent fuel storage racks. It is anticipated that the cannisters will be stored in Region 2 of the racks which provides for high density storage in cells incorporating a fixed neutron poison material for those fuel assemblies which satisfy specific initial enrichment and accumulated burnup criteria. There will be no modification of the racks necessary to store consolidated fuel.

The cannister is a box of 0.93" SS-304 capable of storing the equivalent number of fuel rods from two fuel assemblies (358) on a triangular array. Dimensionally the cannister will be compatible with a Region 2 storage cell. Provisions have been

made in the design of the bottom of the cannister to facilitate cooling water flow and provide chamfered lead-ins for insertion into a storage cell. The top and bottom of the cannister will allow cooling water flow in the channels between the fuel rods.

2. Nuclear

Reference 1 describes the two region storage configuration of the spent fuel storage racks at Ginna. Attached is the criticality safety analysis performed by Pickard, Lowe and Garrick for storage of consolidated fuel in the Region 2 high density configuration. This analysis shows that even for unirradiated fuel the $K_\infty$ of the rack is well below .80, and while this analysis was performed for an Exxon fuel assembly of 3.13 percent enrichment, the specification requirements of 5.4.4a will ensure that fuel assemblies of equivalent reactivity will be bounded. This analysis was performed for Region 2 only, however the Region 1 configuration being much less reactive (acceptable for storing unirradiated fuel assemblies) is also acceptable for storing cannisters of consolidated fuel.

The proposed change to the Technical Specification requires that the initial enrichment and average burnup of the fuel assemblies from which the rods were removed satisfy the requirements for storing non consolidated fuel in Region 2. The attached list of fuel at West Valley indicates that four fuel assemblies have accumulated burnups that, after the 10 percent reduction for uncertainty in the burnup determination, do not satisfy the minimum burnup requirements. However, this irradiated fuel could not achieve a $K_\infty$ of greater than the maximum of .80 for unirradiated fuel and is therefore acceptable for storage in Region 2.

3. Thermal Hydraulic

Reference 4 provides the NRC safety evaluation of the proposed spent fuel pool cooling system which assumed a storage capacity of 1360 fuel assemblies. The proposed change to the Technical Specification will allow storage of up to 1360 fuel assemblies. Therefore, the heat removal capability of the system is adequate.

Attached is an analysis by U.S. Tool & Die, Inc. of natural circulation flow through a cannister of consolidated fuel rods. Using conservative assumptions this analysis concludes that film and local boiling in the water channels between fuel rods could not occur given a fuel assembly cooling time of 2.5 years. Using the methodology of this analysis and imposing an inlet temperature of 150°F and a $\Delta Th$ of 50°F, the average heat output of an assembly would be limited to less than 3510 BTU/hr. This corresponds to a cooling time of 3.1 years per ASB 9-2. The proposed Technical Specification limits the average decay heat of an assembly to 2150 BTU/hr which corresponds to approximately 5 years of cooling time. Given the conservative assumptions and methodology, this additional decrease in decay heat generation at time of consoli-

dation will more than compensate for any uncertainties and will insure that boiling cannot occur in the water channels between rods.

4. Mechanical, Material, Structural

References 1 and 2 document the structural analysis performed for the Ginna spent fuel storage racks under the loads due to storage of consolidated fuel. This analysis determined that the structural integrity of the racks would be maintained under a seismic event.

The cannisters will be fabricated from SS304. All welding will be in accordance with ASME Section 3, subsection NF requirements. The design loads will satisfy the criteria for a seismic category 1 component.

5. Cost/Benefit Assessment

Reference 1 provides the basic information required by the January 18, 1979 NRC guidance. Table 5-1 indicates the schedule for projected fuel discharges assuming consolidation of the 81 fuel assemblies at West Valley and further consolidation when and if it is required up to the requested maximum storage capacity of 1360 fuel assemblies.

The consolidation of the fuel at West Valley is being funded as a research and development project by Rochester Gas & Electric, the Empire State Electric Energy Research Corporation and EPRI. In the future, it is anticipated that costs for fuel consolidation on a production basis will compare very well to other technologies such as dry cask storage. However, that decision and the commitment of material resources can be evaluated on an incremental basis when storage is required.

In References 3, 4 and 5, the additional heat loads that would be anticipated assuming normal discharges up to an end of plant life in 2009 (1360) were calculated. This analysis (Reference 4) assumed normal annual discharges of 36 fuel assemblies 100 hours after reactor shutdown. The resulting heat loads for normal discharges were calculated to increase incrementally from $7.07x10^6$ BTU/HR in 1981 to $9.96x10^6$ BTU/HR in the year 2010. By increasing the cooling time to 14 days in the case of a full core discharge in year 2010 the decay heat load on the spent fuel pool cooling system will remain below $16x10^6$ BTU/HR. At this maximum heat load, the analysis concluded that, assuming 80°F service water with a flow rate of 1600 gpm, the maximum pool temperature would be 150°F and the increase in service water temperature would be within the environmental guidelines of 20°F. The potential for an increase in the heat released to the environmental due to the modification is the increment from $7.07x10^6$ BTU/HR to $9.96x10^6$ BTU/HR or about $3 \times 10^6$ BTU/HR. During the assumed normal operation of the cooling system (80°F service water @ 1000 gpm) this increment represents about a 6°F increase in service water temperature

through the heat exchanger. As stated above even given the maximum heat load for a full core discharge the 20°F environmental guideline for total plant discharge would be satisfied.

6. Radiological Evaluation

Approval of this request for storage of consolidated fuel would increase the storage capacity of the pool from 1016 fuel assemblies to 1360 or an increase of about 30 percent. Reference 1 provides the Radiological Evaluation for the recent rack modification which increased storage capacity from 595 to 1016 fuel assemblies. In their evaluation[2] the NRC staff estimated that the increase in storage capacity would add less than 1 percent to the total annual occupational radiation exposure at the plant. As discussed in Reference 1, it has been our experience that dose rates show a very weak relationship to the amount of fuel stored in the pool. Therefore, this relatively small increase in storage capacity should not affect our ability to maintain individual occupational dose to ALARA levels and within the limits of 10CFR Part 20.

7. Accident Evaluation

In Reference 2, the NRC staff considered three types of accidents; a cask drop or tip, a tornado missile impact and a fuel assembly drop while handling fuel. Because fuel consolidation only involves well cooled fuel assemblies (approximately 5 years) the radiological consequences previously evaluated will remain conservative.

However, the movement of cannisters of consolidated fuel does require a change to the Technical Specifications because the cannister weight will exceed 2000 lbs (app. 2300 lbs) and therefore be classified as a heavy load. The radiological consequences of the drop of a fuel cannister can be evaluated by Reference 2. In this reference the NRC staff evaluated the consequences of a tornado missile impacting 9 fuel assemblies stored in Region 2. They concluded that the dose at the Exclusion Area Boundary would be 2 rem to the thyroid and 0.1 rem to the whole body. These consequences provide a conservative estimate of those due to the impact of a dropped cannister because of its lower kinetic energy and smaller cross section than the postulated tornado missile. The consequences are well within the guidelines of 10CFR Part 100.

The cannisters will be transported within the pool using a special tool (similar in function and size to a fuel assembly handling tool) suspended from the 5 ton hook of the auxiliary building crane. Procedural restrictions and tool design will maintain the distance above the rack and below the surface of the pool at approximately the same as those for a transported fuel assembly.

Table 5-1

Schedule of Anticipated Fuel Discharges

| Month/Year | Discharged | Total Stored | Capacity Remaining Existing | Capacity Remaining Proposed |
|---|---|---|---|---|
| March 1984 | 28 | 332 | 684 | 1028 |
| March 1985 | 28 | 360 | 656 | 1000 |
| *Sept 1985 | 81 | 441 | 575 | 919 |
| March 1986 | 28 | 469 | 547 | 891 |
| March 1987 | 28 | 497 | 519 | 863 |
| March 1988 | 28 | 525 | 491 | 835 |
| March 1989 | 28 | 553 | 463 | 807 |
| March 1990 | 28 | 581 | 435 | 779 |
| March 1991 | 28 | 609 | 407 | 751 |
| March 1992 | 28 | 637 | 379 | 723 |
| March 1993 | 28 | 665 | 351 | 695 |
| March 1994 | 28 | 693 | 323 | 667 |
| March 1995 | 28 | 721 | 295 | 639 |
| March 1996 | 28 | 749 | 267 | 611 |
| March 1997 | 28 | 777 | 239 | 583 |
| March 1998 | 28 | 805 | 211 | 555 |
| March 1999 | 28 | 833 | 183 | 527 |
| March 2000 | 28 | 861 | 155 | 499 |
| March 2001 | 28 | 889 | 127 | 471 |
| March 2002 | 28 | 917 | **99 | 443 |
| March 2003 | 28 | 945 | | 415 |
| March 2004 | 28 | 973 | | 387 |
| March 2005 | 28 | 1001 | | 359 |
| March 2006 | 28 | 1029 | | 331 |
| March 2007 | 28 | 1057 | | 303 |
| March 2008 | 28 | 1085 | | 275 |
| March 2009 | 28 | 1113 | | 247 |
| March 2010 | 28 | 1141 | | 219 |
| March 2011 | 28 | 1169 | | 191 |
| March 2012 | 28 | 1197 | | 163 |
| March 2013 | 28 | 1225 | | 135 |
| March 2014 | 28 | 1253 | | **107 |

* 81 fuel assemblies from West Valley

** Loss of full core discharge capability

Attachment C

In accordance with 10CFR 50.91 these changes to the Technical Specifications have been evaluated against three criteria to determine if the operation of the facility in accordance with the proposed amendment would:

1. involve a significant increase in the probability or consequences of an accident previously evaluated; or
2. create the possibility of a new or different kind of accident from any accident previously evaluated; or
3. involve a significant reduction in a margin of safety.

The proposed modification would increase the spent fuel storage capacity at Ginna from 1016 fuel assemblies to 1360. The safety analysis has shown that the consolidated fuel configuration satisfys NRC Staff accepted criteria for nuclear, structural and thermal hydraulic design. The discussion below examines each of the three criteria stated above and supports the finding that the proposed modification is outside the standards of 10CFR 50.91. Therefore, a no significant hazards finding is warranted.

1. The proposed modification does not involve a significant increase in the probability or the consequences of an accident previously evaluated.

Four potential accident scenarios have been identified: 1) spent fuel cask drop; 2) loss of spent fuel pool forced cooling water; 3) seismic event; 4) drop of a cannister. The probability of these events will not be affected by the amount of fuel stored in the pool.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

RG&E has evaluated the proposed storage of consolidated fuel in accordance with the NRC April 14, 1978 letter "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Application" and appropriate NRC and industry guides, codes and standards. RG&E does not consider a fuel cannister accident to be materially different from a fuel assembly accident since both assume the failure of a handling tool or system. Therefore, RG&E has found no indication that a new or different kind of accident is created.

3. The proposed modification does not involve a significant reduction in the margin of safety.

Under normal operation and accident conditions, the proposed storage of consolidated fuel must satisfy certain criteria in three areas:

1. Nuclear Criticality
2. Thermal Hydraulic
3. Structural Mechanical

In the area of nuclear criticality, the criteria established is that Keff must be less than .95. Section 2 of Attachment B of this Application indicates that this criteria is satisfied with a significantly larger margin than previous analyses.[1] The criteria itself is unchanged from previous submittals, therefore the margin of safety has not been reduced.

Section 3 of Attachment B of the Application evaluates the thermal hydraulic considerations of consolidated fuel storage. The decay heat loads that can result from consolidated fuel will

The consequences of a spent fuel cask drop accident are unchanged by the modification. The current Technical Specifications prohibit the movement of a cask in the auxiliary building. However, an Application for Amendment to the Operating License was submitted to the NRC to delete this restriction by modifying the crane to be single failure proof in accordance with the requirements of NUREG-0554. This application has been approved and will be incorporated into the Technical Specification upon completion of the modification.

The loss of spent fuel pool forced cooling water has been previously evaluated for both the current pool cooling system, and the system to be installed in 1986[4]. The decay heat loads that will be experienced due to the increased storage capacity are no greater than those assumed in these analyses. Therefore, the consequences of this accident are unchanged from those previously evaluated.

The structural response of fully loaded storage racks during a seismic event was evaluated in references 1 and 2. The results of this evaluation satisfied NRC Staff accepted design criteria. Therefore, the consequences of a seismic event are unchanged.

The proposed Technical Specification restricts the movement of a cannister to load paths which are not over racks which contain fuel that has decayed less than 60 days. This will insure that in the unlikely event a cannister is dropped the radiological releases will be well within the guidelines of 10CFR100 and less than what the NRC has previously considered acceptable.[6]

be no greater than those previously evaluated.[4,5] The required cooling times for consolidated fuel will preclude the occurrence of local or film boiling, or any condition which could lead to cladding or fuel degradation. Therefore, the margin of safety has not been reduced.

The structural consideration deal primarily with the response of racks fully loaded with consolidated fuel cannisters during a seismic event. This was evaluated by the NRC staff in reference 2, and was found to satisfy the applicable criteria. With the appropriate criteria satisfied there is no reduction in the margin of safety.

References


1. Application for Amendment to Operating License, April 2, 1984.

2. Letter, J.A. Zwolinski to R.W. Kober, November 14, 1984.

3. U.S. NRC "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Application" April 14, 1978 (revised January 18, 1979).

4. Letter, D.M. Crutchfield to J.E. Maier, November 3, 1981.

5. Letter, L.D. White to D.L. Ziemann, February 13, 1980.

6. Letter, D.M. Crutchfield to J.E. Maier, October 7, 1981.

CRITICALITY SAFETY ANALYSIS FOR THE STORAGE OF CONSOLIDATED FUEL RODS IN REGION 2 OF THE GINNA MDR SPENT FUEL STORAGE RACKS

## I. Introduction

The spent fuel storage racks at the Ginna nuclear plant utilize a two region MDR concept for spent fuel storage wherein Region 1 is for storage of unirradiated or low burnup fuel assemblies and Region 2 is for storage of irradiated fuel assemblies which have achieved some minimum designated burnup which in turn assures the criticality safety of Region 2. This fuel storage concept and the corresponding criticality safety analysis for the Ginna racks have been previously described in Reference 5.

The objective of this analysis is to show that storage cannisters, containing tightly packed fuel rods which were formerly stored as whole fuel assemblies at West Valley, NY, may be safely stored in Region 2 of the Ginna racks. The storage cannister and the planned arrangement of fuel rods within the cannister is illustrated in Figure 4. The highest initial enrichment and the lowest discharge burnup of the Ginna fuel assemblies stored at West Valley are 2.795 w/o and 15,300 MWD/MTU respectively, and these two parameters define the most limiting fuel rods from a criticality safety viewpoint. The analysis presented herein shows that this limiting fuel as well as fuel of significantly higher initial enrichment and substantially lower burnup may be safely stored as consolidated fuel rods in cannisters in Region 2 of the Ginna Spent Fuel Racks.

## II. Validation of Analytic Model Used for Criticality Safety Analysis of the Storage of Consolidated Fuel Rods

Previous criticality safety analysis for the spent fuel storage racks at Ginna utilized an analytic model which was based on validated computer codes and methods developed by or derived from those utilized in the U.S. Naval Reactor Program. This model utilized the LEOPARD,[1] PDQ07,[2] and CINDER[3] computer programs in combination with a blackness theory treatment of the Boraflex absorber in the restricted region of the racks.

The validation and accuracy of this model was previously described in detail.[4,5] For the criticality safety analysis of the storage of consolidated fuel rods in the Ginna spent fuel racks, an analytic model similar to that described above was defined and validated as described below.

For definition and validation of the analytic model, an analysis was performed of the critical experiments conducted for this purpose by the Babcock & Wilcox Company (B&W) and described in Reference 6. Comparison of the results of this analysis with the analytic results reported in Reference 6 indicates the analytic model as defined herein is at least as accurate as the KENO IV model as utilized and reported by B&W.[6]

In order to more accurately account for the large differences in the neutron energy spectrum of tightly packed assemblies of fuel rods as compared to the relatively large water gaps located between such assemblies, it was necessary to utilize the Mixed Number Density (MND) model[7] in the thermal energy group (group 4 of the 4 group LEOPARD, PDQ07 model). The MND model allows diffusion theory to more accurately predict the large thermal flux peaking which occurs in the water gaps between the assemblies consisting of tightly packed fuel rods. The effects of the water gaps on the multiplication factor of the critical assembly is greatly enahanced as the water gap is increased and/or as the fuel rods are more tightly packed within the assembly. For fuel rod lattice spacings typical of whole fuel assemblies as utilized in the reactor core, there is typically little difference between the multiplication factor predicted by the MND model and that predicted with a Wigner-Wilkins (WW) infinite medium thermal spectrum model. However for the tightly packed lattices typical of consolidated fuel rods, the MND model predicts a significantly higher multiplication factor than that predicted by the WW model.

A tightly packed lattice of fuel rods arranged on a triangular pitch results in an irregular boundary for the fuel rods comprising the assembly and, because of the limitations of diffusion theory, requires a decision with regard to how much water should be included within the boundaries of the fuel region vs how much water should be included in the water gap region. The geometric model selected was based upon wrapping a string around the outer edge of the fuel rods comprising the assemblies in the critical experiment and calculating a uniform pitch for all fuel rods comprising the assembly based upon conservation of materials included within the total area bounded by the string around the fuel rods. This geometry (i.e., string around the fuel rods) provides a unique and consistent definition of the fuel region for each of the three types of lattice used in the B&W critical experiments as illustrated in Figure 1 (for the T, S, and SO type lattices). For the fuel rack geometry, the corresponding boundary for the consolidated fuel rods would be the inner walls of the cannister containing the consolidated fuel rods.

There are five basic critical assemblies of interest for validation purposes which include three lattice spacings and three water gap spacings for the tightest triangular pitch lattice spacing. The three lattices correspond to: (1) fuel rods touching on a triangular pitch (T type), (2) fuel rods touching on a square pitch (S type), and (3) fuel rods on a square open pitch (SO type) representative of a typical reactor fuel assembly geometry. The five benchmark core geometries are summarized in Tables 1 and 2 and illustrated in Figure 2 all of which are reproduced from Reference 6.

The two-dimensional quarter core geometric model used to represent the critical assemblies in PDQ07 is shown in Figure 3 for Cores I, IV and V. The models for Cores II and III were similar but utilized more mesh points to represent the larger water gaps between assemblies for these cores. The LEOPARD code was used to generate 4 group macroscopic cross sections for each of the explicit geometric regions shown in Figure 3. The results of these two dimensional quarter core calcutions are shown in Table 3. The large effects on the $k_\infty$'s of the fuel pin cells of both the

water gaps between assemblies and the radial leakage from the core is readily apparent from these results. These results thereby demonstrate that this set of experiments represents a severe test of the individual components of the analytic model as well as the total integrated model.

The results shown in Table 3 do not take into account the neutron leakage in the axial direction from the finite critical assemblies. This axial neutron leakage effect was calculated using a one-dimensional axial model with flux weighted cross sections and a radial buckling from the two-dimensional PDQ07 results together with appropriate boundary conditions at the top of the moderator level and the bottom of the core tank. The results of these calculations, which represent the final calculated $k_{eff}$'s for the citical assemblies, are shown in Table 4. The results of the KENO IV analysis reported in Reference 6 are also shown for comparison. Although the average bias of the KENO IV results is less than that of the LEOPARD-PDQ07 model, the standard deviation of the results for the five experiments is greater with the KENO IV model. Note also that the variable bias with increasing water gap thickness for the three T type lattices that was reported in Reference 6 is essentially non-existent in the LEOPARD-PDQ07 model.

Since the LEOPARD-PDQ07 model being validated here is for the analysis of consolidated fuel rod geometries, the experiments of most interest are those represented by Cores I through IV. For these experiments the model bias is .0224, but the standard deviation is only .0011. Thus the total uncertainty for the LEOPARD-PDQ07 model corresponding to a 95-95 criterion is .0281 (bias + 5.15$\sigma$) which may be compared to the corresponding value derived from the KENO IV analysis of .0643 (bias + 5.15$\sigma$) plus the statistical uncertainty (apparently about .006).

It is also reassuring to note that for Core V, which is representive of whole fuel assemblies, the $k_{eff}$ calculated with the LEOPARD-PDQ07 MND model is .9913, which may be compared with the average calculated $k_{eff}$ previously reported for the LEOPARD-PDQ07 WW model of .9939.[4,5]

## III. Results of Criticality Safety Analysis

The analytic model used in the criticality safety analysis is the same basic model described in Section II above with the addition of the analysis model used for fuel burnup calculations which was described in Reference 5. The storage cannister and the planned arrangement of fuel rods within the cannister is illustrated in Figure 4. The geometry used for the PDQ07 calculations is a basic cell of nominal dimensions representing one-quarter of the area of a repeating array of fuel cannisters loaded into rack modules as shown in Figure 5. Any deviations of the actual rack geometry from this assumed nominal repeating array are included by adding an estimated and conservative, total incremental allowance to the calculated multiplication factor of the basic cell. Manufacturing and thermal considerations are also included by conservatively adding an estimated total incremental allowance for these effects.

In order to take advantage of calculations previously performed and reported in Reference 5, most of the calculations reported herein were based on an EXXON nuclear fuel assembly with an initial enrichment of 3.13 w/o and the physical characteristics shown in Table 5. Comparison of results obtained for the Exxon fuel with a specific calculation for fuel rods of the design stored at West Valley demonstrates that use of the Exxon fuel design parameters is conservative.

Figure 6 shows the calculated k of the spent fuel rack containing cannisters of consolidated fuel rods as a function of the average burnup of the fuel rods in the cannister. Results are shown for both the nominal capacity of the cannisters (i.e., 2 x 179 fuel rods) and a smaller number (i.e., 2 x 175 fuel rods) to cover a case wherein full packing capacity might not be utilized. These results demonstrate that, even for zero burnup and a conservatively high initial enrichment, the $k_\infty$ of the rack is well below 0.80 for both cases of 179 and 175 fuel rods loaded into one-half of the cannister.

The sensitivity of both the $k_\infty$ of an infinite lattice of fuel rods and the $k_\infty$ of the rack is shown in Figure 7 as a function of the average lattice pitch as calculated based on the indicated number of fuel rods occupying an area of one-half of the cannister. These results are presented for demonstration purposes only, since cannister contents significantly less than 179 rods per one-half cannister are not anticipated.

Table 6 shows the results of calculations which demonstrate that the use of Exxon fuel rod design parameters increases the $k_\infty$ of the fuel rack containing cannistered fuel by about .005$\Delta$k compared to results using design parameters characteristic of the fuel rods stored at West Valley. These results demonstrate that the calculations which utilize Exxon fuel rod design parameters result in a conservative evaluation of criticality safety concerns associated with the storage of consolidated fuel rods in Region 2 of the Ginna spent fuel storage racks.

The results in Table 6 also show that even with unirradiated fuel rods, the $k_\infty$ for Region 2 of the Ginna racks with consolidated fuel rods is less than 0.75. Since the minimum average burnup of the fuel rods in any assembly stored at West Valley is about 15,000 MWD/MTU, Figure 6 shows that more realistically the rack $k_\infty$ for storage of consolidated fuel from West Valley in the Ginna racks is less than 0.63.

In view of the demonstrated extremely large margin of safety associated with the storage of consolidated fuel in the Ginna racks, it was not considered to be necessary or worthwhile to perform a detailed evaluation of the effects of calculation biases, tolerances and uncertainties on the multiplication factor of the racks. Instead a conservative estimate of these effects is offered based on results previously presented in Reference 5. Table 7 presents an evaluation of the biases and uncertainties applicable to the results of the basic cell calculation which includes nominal dimensions and parameters. Where a specific evaluation of the pertubation was not performed for consolidated fuel, all pertubations which increase reactivity were taken to be twice the pertubation previously calculated for storage of whole fuel assemblies as

reported in Reference 5. As shown in Table 7, the total reactivity pertubation to be added to the $k_\infty$ calculated for the basic cell is .0560$\Delta$k for consolidated fuel rods.

Since the $k_\infty$ of the basic cell at zero burnup was conservatively evaluated to be .7422 for Exxon fuel rod design parameters, the maximum $k_\infty$ for the rack is conservatively evaluated to be: .7422 + .0560 or .7982. It should be noted that if credit for burnup is included, the maximum $k_\infty$ is further reduced by about .13$\Delta$k resulting in a maximum $k_\infty$ of about .668. It should also be noted that none of the reported results include the effects of the 2000 ppm of boron in the spent fuel pool coolant which would be expected to reduce further the $k_\infty$ by more than .20$\Delta$k.

III. Accident Analysis

The accident analysis previously reported in References 4 and 5 is conservatively applicable to the storage of consolidated fuel rods in the Ginna racks; i.e., there cannot be any significant increase in multiplication factor as the result of a dropped fuel assembly which comes to rest on the top of the racks, and the potential reactivity increase from locating a fuel assembly outside of but immediately adjacent to the rack is more than compensated for by the large reduction in multiplication factor due to the presence of 2000 ppm boron in the pool water.

Another accident that could be postulated involves the loss of containment of all the fuel rods in a single cannister (i.e., 2 x 179 or 358 fuel rods) and the subsequent relocation of these rods on a uniform and optimum pitch (i.e., the pitch which results in the maximum $k_\infty$) in the space above the racks. While the probability of such an event must be negligible, the occurrence of the accident would still not result in a criticality safety problem due to the radial leakage from the resulting finite array of rods, the burnup of the rods which may be cannistered, and the presence of a minimum of 2000 ppm boron in the coolant.

For an initial enrichment of 3.13 w/o, Figure 7 shows the optimum $k_\infty$ is about 1.42 for an infinite lattice of these fuel rods. The pitch corresponding to this optimum $k_\infty$ is .632 inches. The optimum pitch with 2000 ppm boron in the coolant will be much less than .632 inches, and therefore this value can be conservatively used to define the dimensions of an array consisting of 358 rods on a pitch of .632 inches. A square array of 361 rods on a pitch of .632 inches would be 12.0 inches on a side, and for such an arrangement of fuel rods, the neutron non-leakage probability would be appropriately 0.67. Thus radial neutron leakage effects would reduce the maximum $k_\infty$ from 1.42 to 0.95.

The reactivity loss due to a burnup of 15,000 MWD/MTU for fuel rods with an initial enrichment of 3.13 w/o is about $0.13\Delta\rho$. The effect of this reactivity loss may be conservatively evaluated as 0.95 - .13 = 0.82. The reactivity reduction of unirradiated fuel rods with an initial enrichment of 3.13 w/o due to the presence of 2000 ppm boron in the coolant is more than $0.27\Delta\rho$. The effect of this reactivity reduction may be conservatively evaluated as 0.82 - 0.27 = 0.55.

This analysis and the resulting large subcritical margin demonstrate that the postulated accident could not result in a criticality safety problem.

# REFERENCES


1. R.F. Barry, "LEOPARD--A Spectrum Dependent Non-Spatial Depletion Code for the IBM-7094," WCAP-3269, September 1963.

2. W.R. Caldwell, "PDQ-7 Reference Manual," WAPD-TM-678, January 1967.

3. Electric Power Research Institute, "Fission Product Data for Thermal Reactors, Part 1 and Part 2: Data Set for EPRI-CINDER and Users Manual for EPRI-CINDER Code and Data," EPRI NP-356, Final Report (1976).

4. (RGE Licensing Submittal Ginna Spent Fuel Racks)

5. (RGE Licensing Submittal Ginna Spent Fuel Rack Modification)

6. G.S. Hoovler, et. al., "Critical Experiments Supporting Underwater Storage of Tightly Packed Configurations of Spent Fuel Pins," BAW-1645-4, November 1981.

7. R.J. Breen, "A One-Group Model for Thermal Activation Calculations," Nuc. Sci. and Eng. 9, 91 (1961).

TABLE 1. BENCHMARK CORE DESCRIPTIONS

| Core designation | Description | Intermodular spacing, cm |
|---|---|---|
| I | 5 × 5 array, T-type modules | 1.778 by 1.945[(a)] |
| II | 5 × 5 array, T-type modules | 2.539 by 2.709[(a)] |
| III | 5 × 5 array, T-type modules | 3.807 by 3.976[(a)] |
| IV | 5 × 5 array, S-type modules | 1.778 |
| V | 5 × 5 array, SO-type modules | 1.792 |

(a) Because of the triangular geometry, the pins on two edges of the T-type modules were evenly aligned and those on the other two edges were in a staggered alignment (see Figure 5). The first spacing quoted is the separation between staggered row edges of adjacent modules, and the second is the distance between even-row edges of adjacent modules (see Figure 6).

TABLE 2. COMPARISON BETWEEN DESIGN AND AS-BUILT CORE DIMENSIONS

| Core | Pin pitch, cm | | Intermodular spacing, cm | |
|---|---|---|---|---|
| | Design | Actual[a] | Design | Actual[a] |
| I | 1.209±0.001 | 1.2093 | 1.778±0.025 by 1.946±0.025 | 1.778 by 1.945 |
| II | 1.209±0.001 | 1.2093 | 2.540±0.025 by 2.707±0.025 | 2.539 by 2.709 |
| III | 1.209±0.001 | 1.2093 | 3.810±0.025 by 3.978±0.025 | 3.807 by 2.976 |
| IV | 1.209±0.001 | 1.2090 | 1.778±0.025 | 1.778 |
| V | 1.410±0.013 | 1.4097 | 1.778±0.025 | 1.792 |

[a] Average derived from measurement of 100% of core hardware pieces.

TABLE 3. RESULTS OF TWO-DIMENSIONAL QUARTER CORE CALCULATIONS OF CRITICAL ASSEMBLIES-CORES I THROUGH V

| Core Number | Lattice Type | Fuel Pin Cell $k_\infty$ | Module $k_\infty$ in Core | Core $k_{eff}$ |
|---|---|---|---|---|
| I | T | .8709 | 1.0865 | .9968 |
| II | T | .8721 | 1.0756 | .9965 |
| III | T | .8750 | 1.0627 | .9962 |
| IV | S | 1.0169 | 1.0803 | .9975 |
| V | SO | 1.1539 | 1.0749 | 1.0064 |

TABLE 4. FINAL RESULTS - ANALYSIS OF CONSOLIDATED FUEL CRITICAL EXPERIMENTS

| Core Number | Lattice Type | LEOPARD-PDQ07 MODEL $k_{eff}$ | LEOPARD-PDQ07 MODEL $1-k_{eff}$ | KENO IV MODEL[6] $k_{eff}$ | KENO IV MODEL[6] $1-k_{eff}$ |
|---|---|---|---|---|---|
| I | T | .9764 | .0236 | 1.002 | -.002 |
| II | T | .9771 | .0229 | .984 | .016 |
| III | T | .9780 | .0220 | .979 | .021 |
| IV | S | .9790 | .0210 | .996 | .004 |
| V | SO | .9913 | .0087 | 1.003 | -.003 |
| I-V | | .9804 | .0196 | .9928 | .0072 |
| I-IV | | .9776 | .0224 | .9903 | .0097 |

| | LEOPARD-PDQ07 MODEL STANDARD DEVIATION | KENO IV MODEL[6] STANDARD DEVIATION |
|---|---|---|
| I-V | .0062 | .0108 |
| I-IV | .0011 | .0106 |

TABLE 5. EXXON FUEL ASSEMBLY TECHNICAL INFORMATION FOR GINNA NUCLEAR PLANT

| | |
|---|---|
| Rod Array | 14 x 14 |
| Rods Per Assembly | 179 |
| Rod Pitch, In. | 0.556 |
| Overall Dimensions, In. | 7.784 |
| | |
| Active Fuel Height, In. | 142 |
| Clad Thickness, In. | .030 |
| | |
| Fuel Rod O.D., In. | .424 |
| Pellet Diameter, In. | .3565 |
| Diametral Gap, In. | .0075 |
| Pellet Density (% theoretical) | 94 |
| | |
| Control Rod Guide Tubes | |
| Outer Diameter, In. | .540 |
| Wall Thickness, In. | .510 |
| Material | Zircaloy |
| | |
| Instrument Tube | |
| Outer Diameter, In. | .424 |
| Wall Thickness, In. | .346 |
| Material | Zircaloy |

TABLE 6. COMPARISON OF RESULTS FOR FUEL RODS OF EXXON DESIGN WITH RESULTS FOR FUEL RODS STORED AT WEST VALLEY

| Fuel Rod Parameters | $k_\infty$ of Infinite Lattice | $k_\infty$ of Rack with Cannisters |
|---|---|---|
| Exxon Design at 3.13 w/o | .9326 | .7422 |
| Design Stored at West Valley at 2.795 w/o | .9284 | .7369 |

Note:

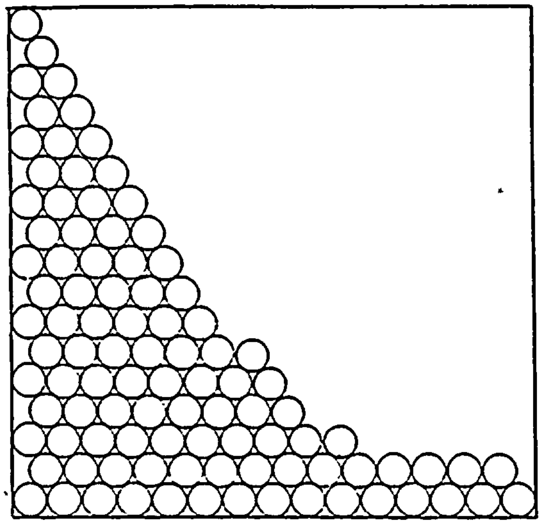
All data are based on results for unirradiated fuel rods (i.e., zero fuel burnup) and a lattice pitch derived by assuming 179 fuel rods are stored in an area corresponding to one-half of a storage cannister.

## TABLE 7. SUMMARY OF REACTIVITY BIASES AND UNCERTAINTIES FOR GINNA REGION 2 MDR

| Description | Reactivity Effects ($\Delta k_\infty$) | |
|---|---|---|
| **Calculation Biases** | Storage of Whole Fuel Assemblies[1] | Storage of Consolidated Fuel Rods |
| Leopard/PDQ model bias | +0.0031 | +0.0224 |
| Modeling Effect | +0.0005 | +0.0010 |
| Mesh Spacing Effect | +0.0002 | +0.0004 |
| Most Reactive Temperature over operating range | +0.0000 | +0.0000 |
| Most Reactive Water Density | +0.0000 | +0.0000 |
| Region 1 - Region 2 Interface Effect | +0.0123 | +0.0246 |
| Total Bias | +0.0161 | +0.0484 |
| **Tolerances and Uncertainties (95/95)** | | |
| Depleted fuel assembly reactivity uncertainties | 0.0102 | 0.0000[2] |
| Maximum error due to pitch tolerance | 0.0019 | 0.0038 |
| Maximum error due to SS thickness tolerance | 0.0002 | 0.0004 |
| Maximum error due to pellet density tolerance (± .015) | 0.0015 | 0.0030 |
| Maximum error due to pellet diameter tolerance (± .001") | 0.0005 | 0.0010 |
| Calculational Uncertainty | 0.0186 | 0.0057 |
| Total Uncertainty (statistical) | 0.0214 | 0.0076 |
| Maximum reactivity change from biases and uncertainties | 0.0375 | 0.0560 |

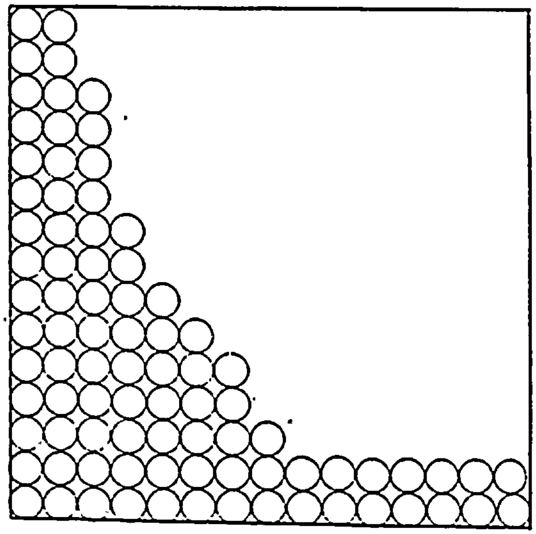
(1) Values taken from Table 9 of Reference 5.
(2) No credit taken for fuel depletion effects.

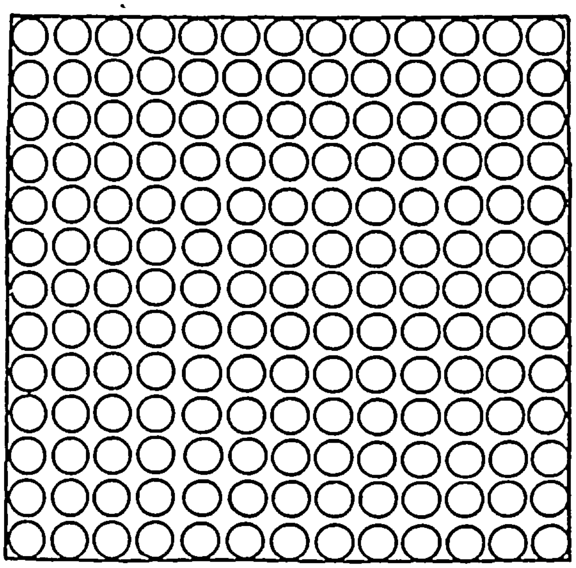
FIGURE 1. FUEL MODULES USED IN CRITICAL EXPERIMENTS



Module with Fuel Rods Touching on a Triangular Pitch (T Type)

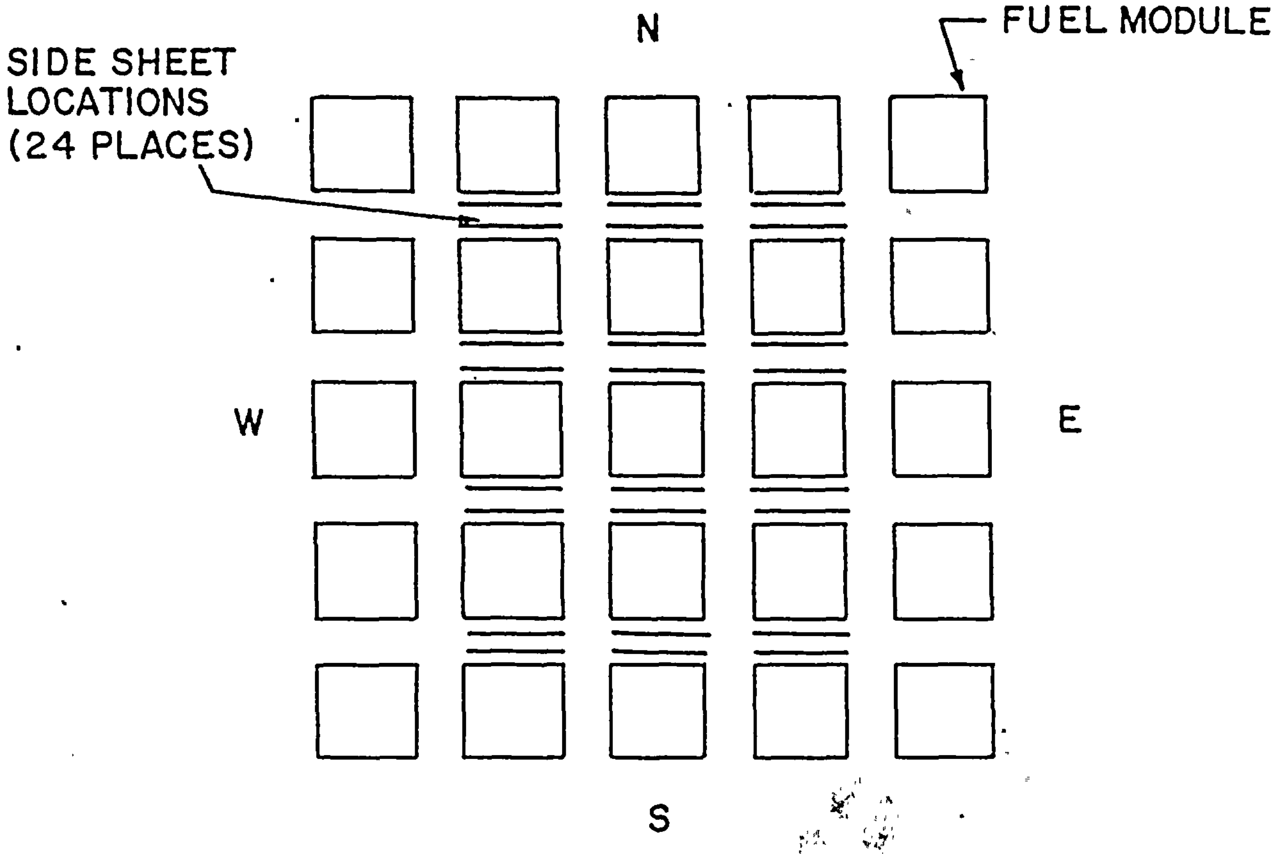
Module with Fuel Rods Touching on a Square Pitch (S Type)





Module with Fuel Rods Slightly Separated on a Square Pitch (SO Type)

FIGURE 2. PLAN VIEW OF CORE SHOWING SIDE SHEET LOCATIONS




STAGGERED-ROW EDGES ALWAYS FACED EAST - WEST
(T-TYPE MODULES ONLY)

FIGURE 3. PDQ07 MODEL OF CRITICAL ASSEMBLIES
(CORES I, IV, AND V)

1. Fuel Pin Cells
2. Aluminum Corner Rods
3. Aluminum Plates and Water
4. Water
5. Water

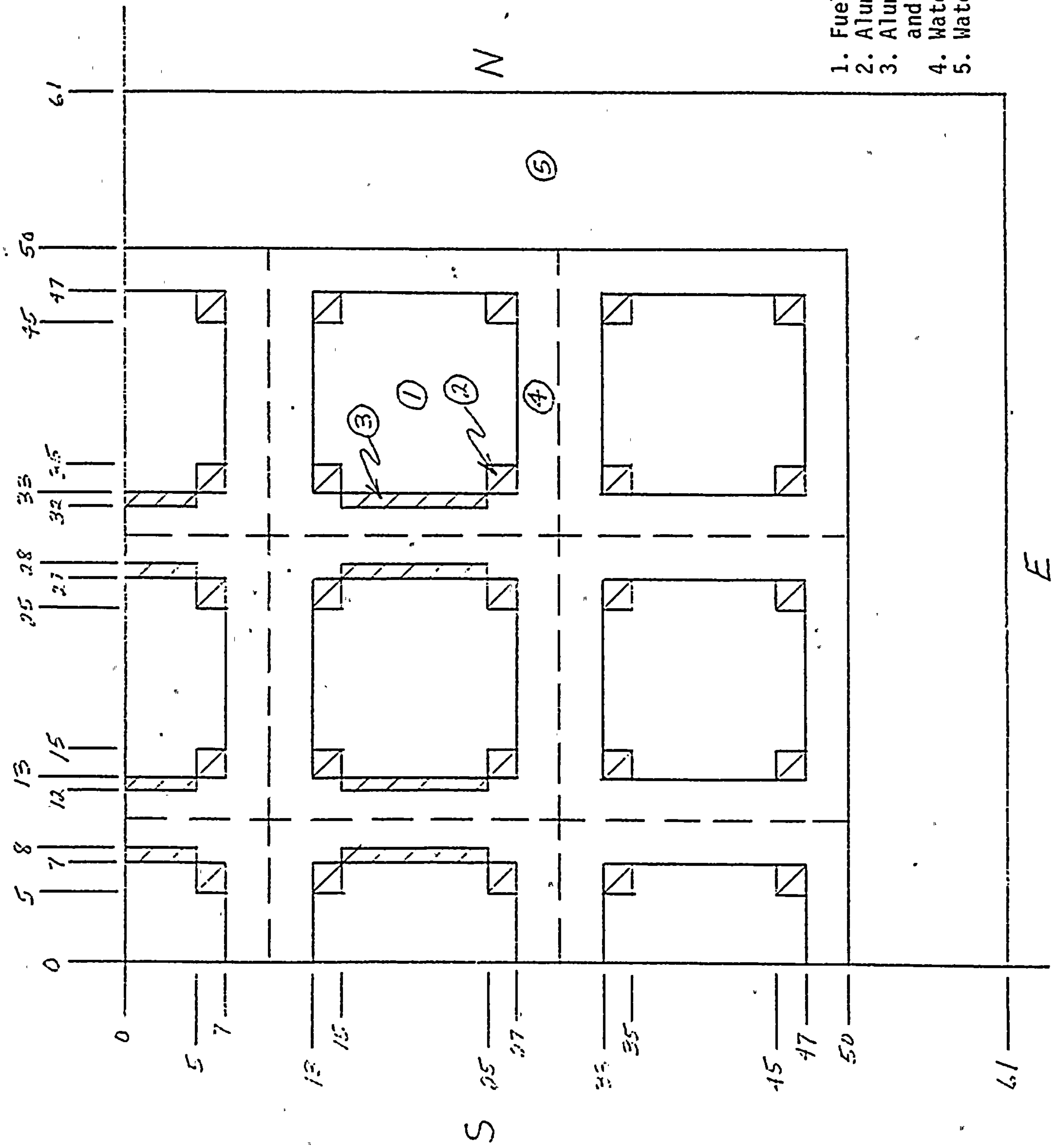


FIGURE 4. CONSOLIDATED FUEL RODS IN STORAGE CANNISTER

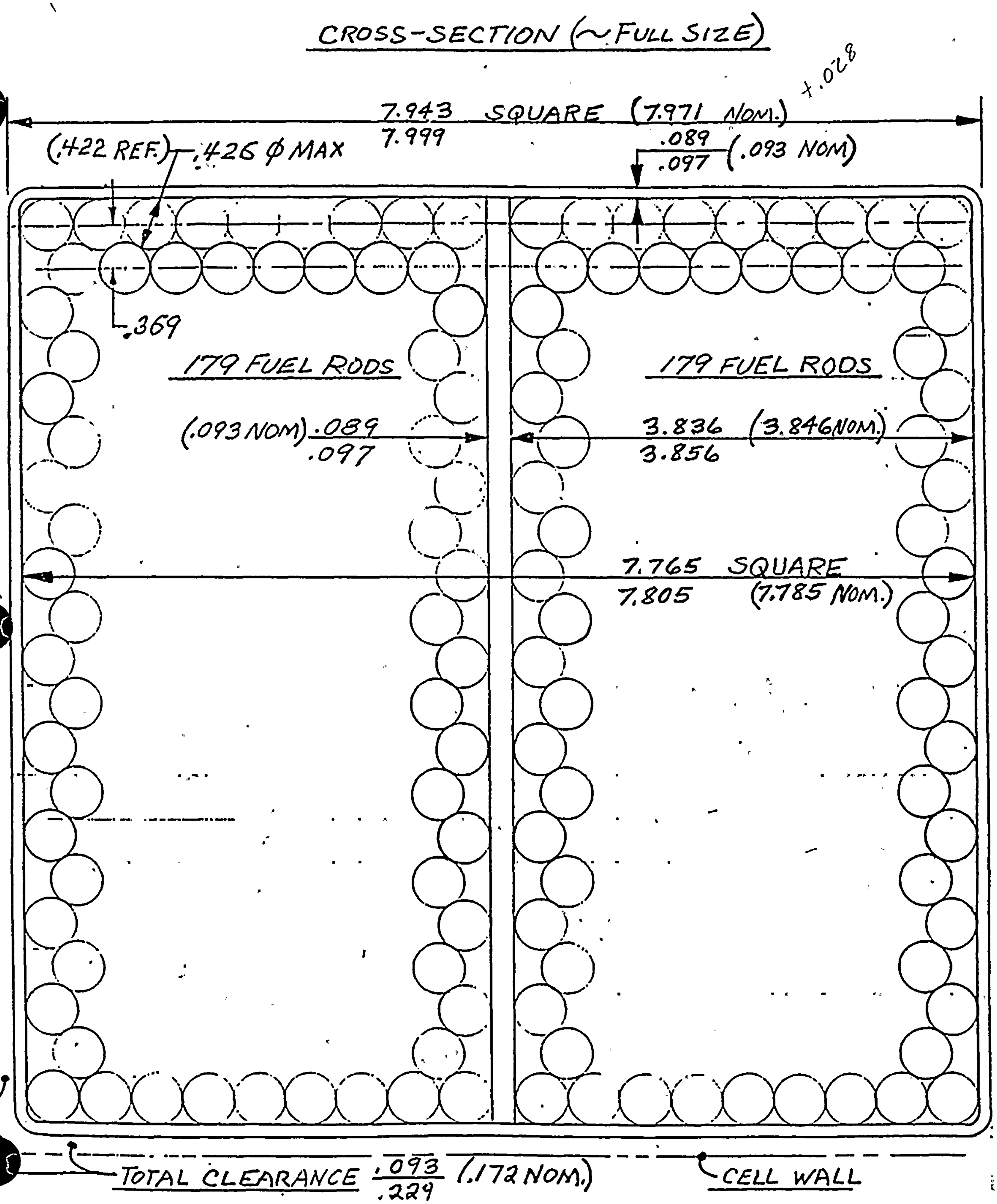
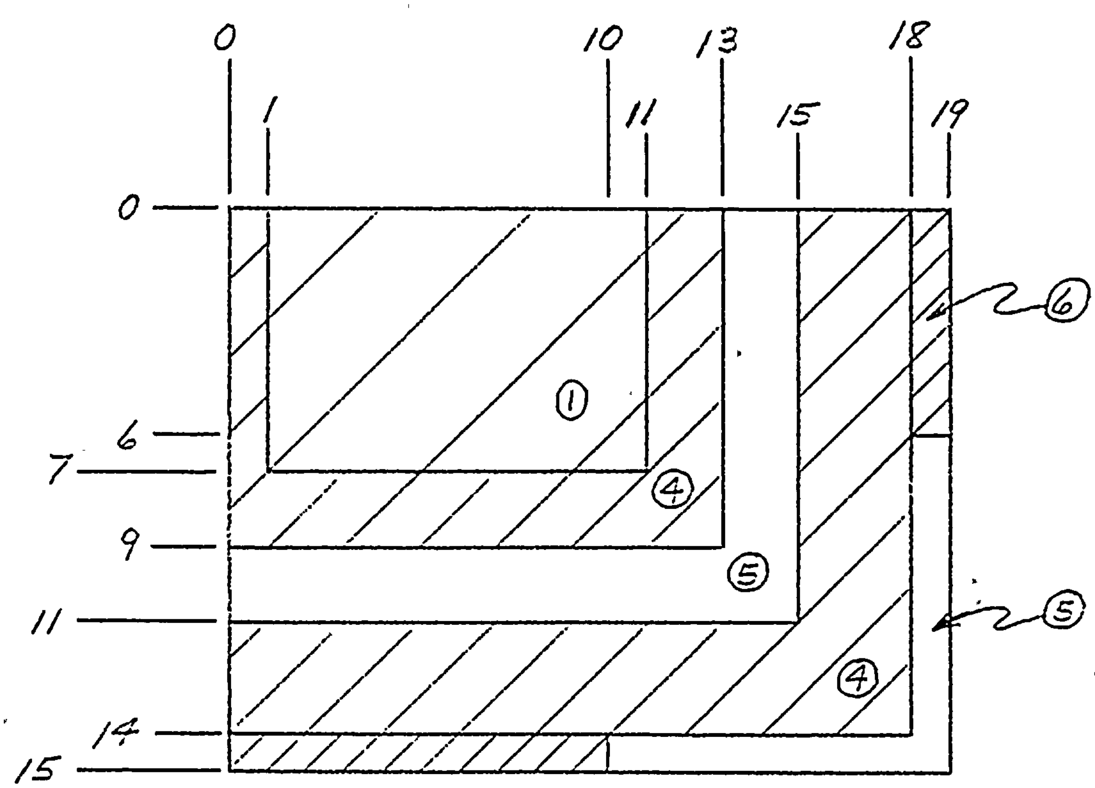


FIGURE 5. BASIC CELL MODEL USING PDQ07




① Consolidated unit fuel pin cells

④ Stainless Steel (304)

⑤ Water

⑥ Boraflex (.020 gm B10/$cm^2$)

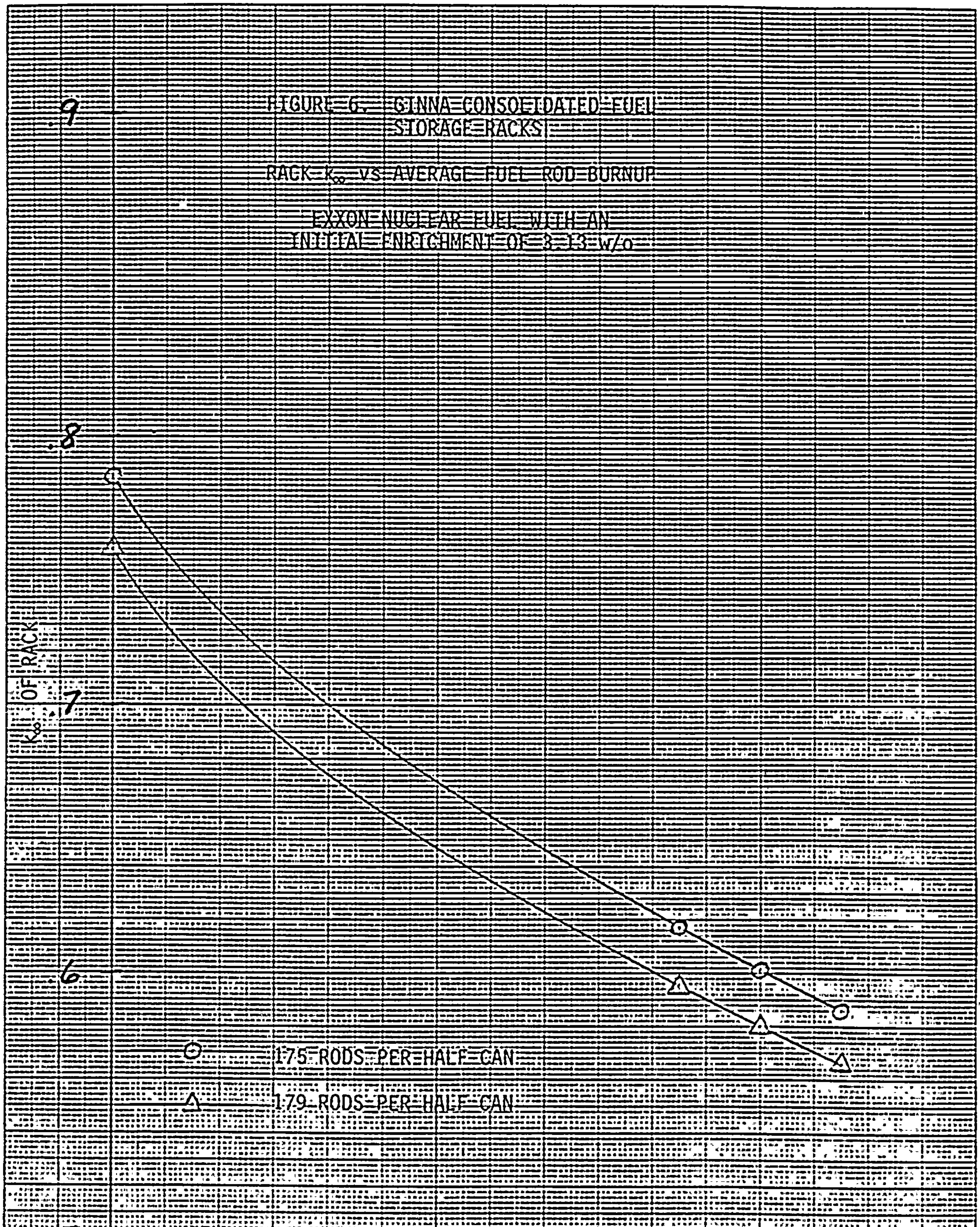


FIGURE 6. GINNA CONSOLIDATED FUEL STORAGE RACKS
RACK $k_\infty$ VS AVERAGE FUEL ROD BURNUP
EXXON NUCLEAR FUEL WITH AN INITIAL ENRICHMENT OF 3.13 w/o
.9
.8
.7
.6
.5
$k_\infty$ OF RACK
175 RODS PER HALF CAN
179 RODS PER HALF CAN
0
10
20
30
BURNUP (MWD/KGU)

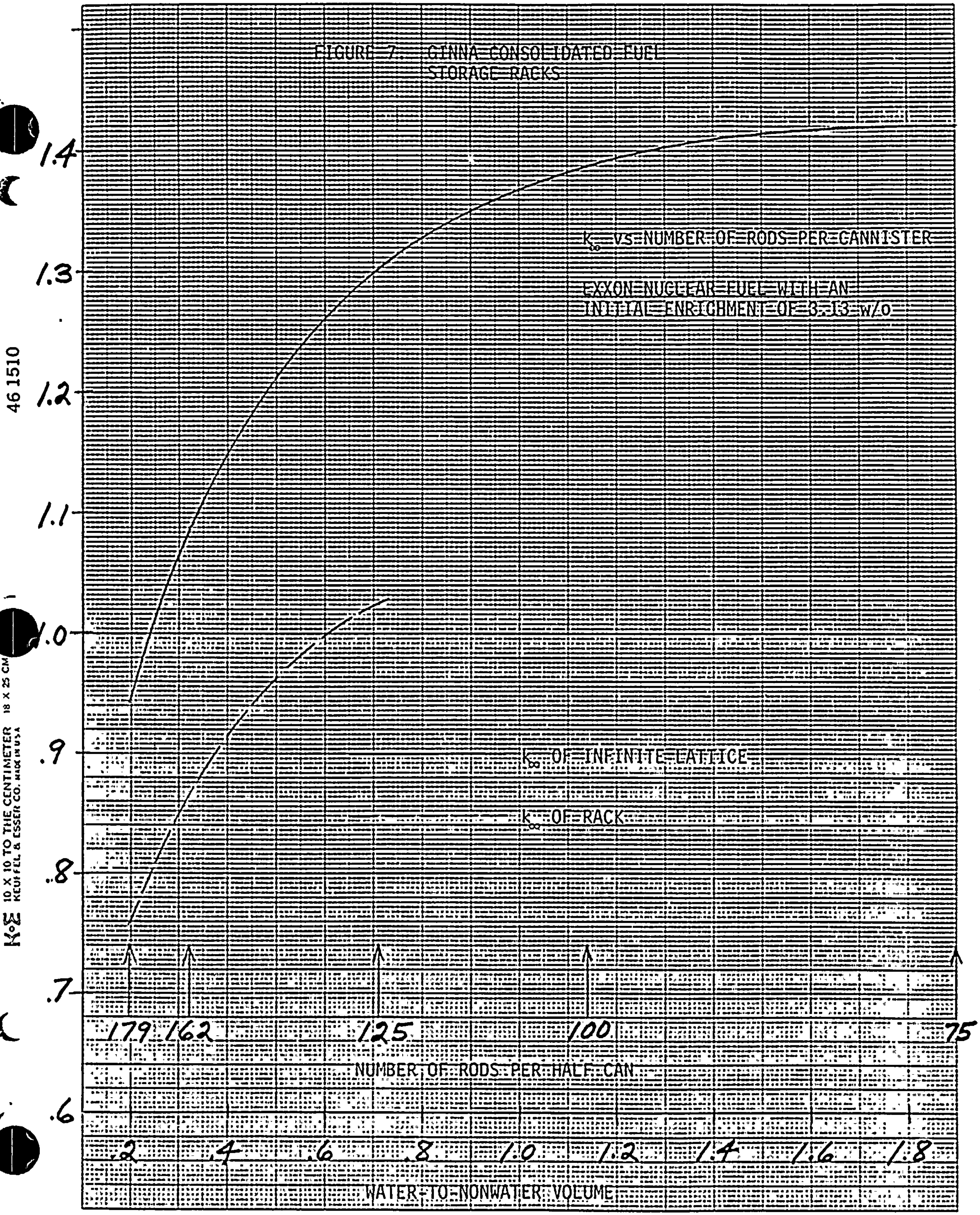


FIGURE 7 GINNA CONSOLIDATED FUEL STORAGE RACKS
$k_\infty$ vs NUMBER OF RODS PER CANNISTER
EXXON NUCLEAR FUEL WITH AN INITIAL ENRICHMENT OF 3.13 w/o
$k_\infty$ OF INFINITE LATTICE
$k_\infty$ OF RACK
1.4
1.3
1.2
1.1
1.0
.9
.8
.7
.6
179
162
125
100
75
NUMBER OF RODS PER HALF CAN
.2
.4
.6
.8
1.0
1.2
1.4
1.6
1.8
WATER-TO-NONWATER VOLUME
46 1510
K&E 10 X 10 TO THE CENTIMETER 18 X 25 CM
KEUFFEL & ESSER CO. MADE IN U.S.A.

# Rochester Gas and Electric Corporation

Inter-Office Correspondence

January 30, 1985

SUBJECT: Consolidated Fuel Storage

TO: PORC
NSARB

Attached is a proposed change to the Technical Specifications to allow the storage of consolidated fuel. This proposed change is in conjunction with the research and development project to consolidate the 81 fuel assemblies at West Valley. The consolidation process will take place at West Valley, therefore, this submittal does not address any consolidation activities.

This change would increase the spent fuel storage capacity to 1360 fuel assemblies. Consolidation involves removing the fuel rods from two assemblies and placing them in one canister. The canister could be stored in either Region 1 or 2 of the spent fuel racks. The grids, guide tubes and nozzles of the assembly will be compacted and stored in a separate canister. We anticipate that this structural material from 10 fuel assemblies can be stored in one canister.

John Cook

John D. Cook

Attachment

Regions RGAF1, RGAF2 and RGAF3 were shipped to West Valley. Regions RGAF1 and RGAF2 will be returned to Ginna.

| Assembly Ident. Code | | | | Initial Enrich-Ment-W/O | Initial Assembly Weight-KG | MWD To Date | Burnup MWD/T |
|---|---|---|---|---|---|---|---|
| 11 | RGAF1 | A01 | SB11 | 2.453 | 397.938 | 8121. | 20409. |
| 11 | RGAF1 | A02 | SB05 | 2.453 | 397.938 | 8048. | 20223. |
| 11 | RGAF1 | A03 | SI01 | 2.453 | 397.939 | 8432. | 21188. |
| 11 | RGAF1 | A04 | SG01 | 2.453 | 397.938 | 8481. | 21312. |
| 11 | RGAF1 | A05 | SP05 | 2.453 | 397.938 | 8170. | 20530. |
| 11 | RGAF1 | A06 | SA06 | 2.453 | 397.938 | 8135. | 20444. |
| 11 | RGAF1 | A07 | SD12 | 2.453 | 397.938 | 8544. | 21470. |
| 11 | RGAF1 | A08 | SB10 | 2.453 | 397.938 | 8712. | 21892. |
| 11 | RGAF1 | A09 | SK12 | 2.453 | 397.938 | 8589. | 21583. |
| 11 | RGAF1 | A10 | SB04 | 2.453 | 397.938 | 8615. | 21649. |
| 11 | RGAF1 | A11 | SE07 | 2.453 | 397.938 | 8160. | 20505. |
| 11 | RGAF1 | A12 | SG11 | 2.453 | 397.938 | 7203. | 18101. |
| 11 | RGAF1 | A13 | SH09 | 2.453 | 397.938 | 7436. | 18686. |
| 11 | RGAF1 | A14 | SF01 | 2.453 | 397.938 | 8448. | 21230. |
| 11 | RGAF1 | A15 | SC11 | 2.453 | 397.938 | 8562. | 21515. |
| 11 | RGAF1 | A16 | SB09 | 2.453 | 397.938 | 8593. | 21593. |
| 11 | RGAF1 | A17 | SA07 | 2.453 | 397.938 | 7198. | 18087. |
| 11 | RGAF1 | A18 | SD11 | 2.453 | 397.938 | 8048. | 20225. |
| 11 | RGAF1 | A19 | SG10 | 2.453 | 397.938 | 8556. | 21500. |
| 11 | RGAF1 | A20 | SF12 | 2.453 | 397.938 | 8552. | 21490. |
| 11 | RGAF1 | A21 | SC10 | 2.453 | 397.938 | 8103. | 20363. |
| 11 | RGAF1 | A22 | SA11 | 2.453 | 397.938 | 7975. | 20040. |
| 11 | RGAF1 | A23 | SH12 | 2.453 | 397.938 | 8570. | 21537. |
| 11 | RGAF1 | A24 | SE12 | 2.453 | 397.938 | 8622. | 21667. |
| 11 | RGAF1 | A25 | SC09 | 2.453 | 397.938 | 8065. | 20266. |
| 11 | RGAF1 | A26 | SB12 | 2.453 | 397.938 | 8545. | 21473. |
| 11 | RGAF1 | A27 | SE08 | 2.453 | 397.938 | 8577. | 21552. |
| 11 | RGAF1 | A28 | SE04 | 2.453 | 397.938 | 8391. | 21086. |
| 11 | RGAF1 | A29 | SA02 | 2.453 | 397.938 | 8274. | 20793. |
| 11 | RGAF1 | A30 | SL11 | 2.453 | 397.938 | 8319. | 20904. |
| 11 | RGAF1 | A31 | SA12 | 2.453 | 397.938 | 8283. | 20815. |
| 11 | RGAF1 | A32 | SH01 | 2.453 | 397.938 | 8748. | 21983. |
| 11 | RGAF1 | A33 | SA09 | 2.453 | 397.938 | 8211. | 20633. |
| 11 | RGAF1 | A34 | SB06 | 2.453 | 397.938 | 8359. | 21006. |
| 11 | RGAF1 | A35 | SE02 | 2.453 | 397.938 | 8369. | 21032. |
| 11 | RGAF1 | A36 | SA05 | 2.453 | 397.938 | 8602. | 21615. |
| 11 | RGAF1 | A37 | SC05 | 2.453 | 397.938 | 8657. | 21755. |
| 11 | RGAF1 | A38 | SB03 | 2.453 | 397.938 | 8501. | 21364. |
| 11 | RGAF1 | A39 | SC12 | 2.453 | 397.938 | 8597. | 21604. |
| 11 | RGAF1 | A40 | SA10 | 2.453 | 397.938 | 8215. | 20644. |
| 11 | RGAF1 | A41 | SA03 | 2.453 | 397.938 | 8404. | 21119. |
| 12 | RGAF2 | B01 | SA01 | 2.795 | 391.640 | 8112. | 20712. |
| 12 | RGAF2 | B02 | SM12 | 2.795 | 391.640 | 8185. | 20900. |
| 12 | RGAF2 | B03 | SM11 | 2.795 | 391.640 | 8159. | 20833. |
| 12 | RGAF2 | B04 | SC06 | 2.795 | 391.640 | 7540. | 19252. |
| 12 | RGAF2 | B05 | SI07 | 2.795 | 391.640 | 7511. | 19177. |
| 12 | RGAF2 | B06 | SD04 | 2.795 | 391.640 | 8806. | 22485. |
| 12 | RGAF2 | B07 | SH11 | 2.795 | 391.640 | 7217. | 18428. |
| 12 | RGAF2 | B08 | SG04 | 2.795 | 391.640 | 6126. | 15642. |

| | Assembly Ident. Code | | | Initial Enrich-Ment-W/O | Initial Assembly Weight-KG | MWD To Date | Burnup MWD/T |
|---|---|---|---|---|---|---|---|
| [illegible] | RGAF2 | B09 | SH02 | 2.795 | 391.640 | 7646. | 19524. |
| 12 | RGAF2 | B10 | SI06 | 2.795 | 391.640 | 7081. | 18080. |
| 12 | RGAF2 | B11 | SF10 | 2.795 | 391.640 | 7544. | 19262. |
| 12 | RGAF2 | B12 | SI04 | 2.795 | 391.640 | 7530. | 19227. |
| 12 | RGAF2 | B13 | SF05 | 2.795 | 391.640 | 6408. | 16363. |
| 12 | RGAF2 | B14 | SC08 | 2.795 | 391.640 | 8624. | 22021. |
| 12 | RGAF2 | B15 | SG09 | 2.795 | 391.640 | 7014. | 17909. |
| 12 | RGAF2 | B16 | SG06 | 2.795 | 391.640 | 7383. | 18852. |
| 12 | RGAF2 | B17 | SK11 | 2.795 | 391.640 | 7438. | 18991. |
| 12 | RGAF2 | B18 | SL12 | 2.795 | 391.640 | 8315. | 21230. |
| 12 | RGAF2 | B19 | SK09 | 2.795 | 391.640 | 7152. | 18262. |
| 12 | RGAF2 | B20 | SI03 | 2.795 | 391.640 | 7224. | 18445. |
| 12 | RGAF2 | B21 | SK10 | 2.795 | 391.640 | 7426. | 18962. |
| 12 | RGAF2 | B22 | SD09 | 2.795 | 391.640 | 6992. | 17854. |
| 12 | RGAF2 | B23 | SE10 | 2.795 | 391.640 | 7175. | 18320. |
| 12 | RGAF2 | B24 | SD10 | 2.795 | 391.640 | 7067. | 18045. |
| 12 | RGAF2 | B25 | SH06 | 2.795 | 391.640 | 6922. | 17674. |
| 12 | RGAF2 | B26 | SL10 | 2.795 | 391.640 | 7151. | 18260. |
| 12 | RGAF2 | B27 | SF03 | 2.795 | 391.640 | 6977. | 17814. |
| 12 | RGAF2 | B28 | SE06 | 2.795 | 391.640 | 7148. | 18252. |
| 12 | RGAF2 | B29 | SE11 | 2.795 | 391.640 | 6054. | 15458. |
| 12 | RGAF2 | B30 | SH05 | 2.795 | 391.640 | 6001. | 15323. |
| 12 | RGAF2 | B31 | SD08 | 2.795 | 391.640 | 8175. | 20874. |
| 12 | RGAF2 | B32 | SH05 | 2.795 | 391.640 | 7207. | 18402. |
| [illegible] | RGAF2 | B33 | SD02 | 2.795 | 391.640 | 7315. | 18679. |
| 1 | RGAF2 | B34 | SH10 | 2.795 | 391.640 | 7372. | 18824. |
| 12 | RGAF2 | B35 | SF03 | 2.795 | 391.640 | 6369. | 16262. |
| 12 | RGAF2 | B36 | SD01 | 2.795 | 391.640 | 8221. | 20992. |
| 12 | RGAF2 | B37 | SF11 | 2.795 | 391.640 | 6391. | 16317. |
| 12 | RGAF2 | B38 | SC04 | 2.795 | 391.640 | 7187. | 18352. |
| 12 | RGAF2 | B39 | SB08 | 2.795 | 391.640 | 7428. | 18965. |
| 12 | RGAF2 | B40 | SD07 | 2.795 | 391.640 | 7180. | 18334. |

TELEPHONE (412) 487-7030

U. S. TOOL & DIE, INC.

4030 ROUTE 8 • ALLISON PARK, PENNSYLVANIA 15101

# COMPACTED FUEL ROD THERMAL AND HYDRAULIC ANALYSIS

R. E. GINNA NUCLEAR STATION

8446-00-0008

PREPARED FOR
ROCHESTER GAS AND ELECTRIC CORPORATION
ROCHESTER, NEW YORK

PREPARED BY ______ DATE 2-25-85

REVIEWED BY William J. ______ DATE 2/25/85

APPROVED BY Ray Linder DATE 2/25/85
Ray Linder
Manager of Engineering

APPROVED BY ______ DATE 2-22-85
J. Golobic
Quality Assurance Manager

# INTRODUCTION

This Thermal-Hydraulic Analysis for Rochester Gas and Electric Corp., Ginna Nuclear Station, is for the storage of consolidation canisters which contain the fuel rods from two separate fuel assemblies arranged in a tightly-packed triangular array. This consolidated canister is stored in the spent fuel pool and is cooled by natural circulation of pool coolant water through the canister.

## 1.1 DETAILED ANALYSIS FOR NATURAL CIRCULATION COOLING

Fuel pool cooling systems are typically designed with cooling spargers located near the bottom of the pool and strainers at the top geometrically arrayed so that the spent fuel is cooled by the cold water flowing under the racks, upward through the spent fuel channels, and across the pool to the outlet. The pool bulk temperatures are established on the basis of the heat-exchanger mass flow rates and design (or off-design) characteristics, the cooling water inlet temperature, and the total amount of residual decay heat to be removed.

Within the fuel channels, it is difficult to establish accurately how much forced convection flow goes to each fuel assembly or fuel rod since geometry complicates the analysis for local cooling. It is therefore necessary to consider natural circulation cooling as the prime means of removing the decay heat in some of the spent fuel racks.

The natural circulation problem can be modeled as shown in Figure 1. Cold water (at the pool bulk temperature) exerts hydrostatic forces on heated water within the fuel rod channels. The minimum amount of driving pressure for this loop is given by equation (1).

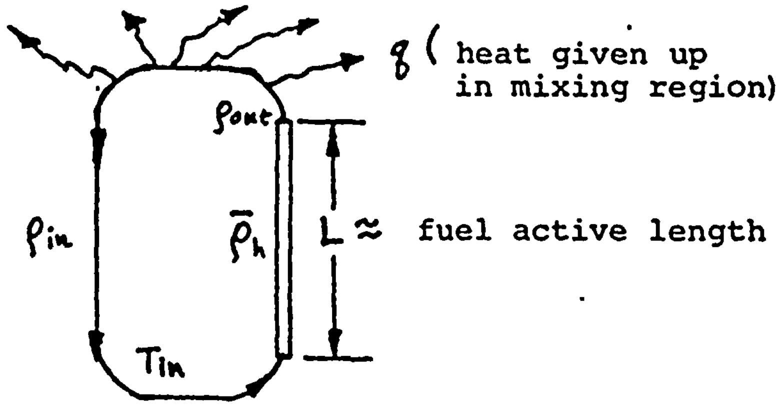



FIGURE 1
Driving Pressure Model
for Natural Circulation

$$\Delta p_d = (\rho_{in} - \overline{\rho_h}) \frac{g}{g_c} L \qquad (1)$$

where L is the fuel active length, $\rho_{in}$ is the pool bulk density, and $\overline{\rho_h}$ is the heated channel axially averaged density.

The uncertainty in equation (1) is due primarily to the lack of precision with which the mixing lengths and densities above the heated channel are known. Since the pool contains approximately 25 feet of water on top of the racks, some mixing length credit could be taken. However, local convection currents may tend to mix the outlet coolant soon after it is heated. Thus, the lower bound in equation (1) is expected to be accurate within a factor of 2.

For axially symmetric heat flux distributions, $\overline{\rho_h} = (\rho_{out} + \rho_{in})/2$. Assuming this and expanding $\rho_{out}$ in a Taylor Series about $\rho_{in}$, the driving pressure can be approximated as:

$$\Delta p_d = \frac{1}{2} \beta \rho (\Delta T_h) \frac{g}{g_c} L \qquad (2)$$

where

$\beta = \frac{-1}{\rho} \left(\frac{\partial \rho}{\partial T}\right)_p$ is the volumetric coefficient of thermal expansion for the water, a mild function of pressure

and

$\Delta T_h$ = coolant temperature increase in the heated channel.

As the temperature increases from 100° to 200°F, $\beta$ changes from $2 \times 10^{-4}$ to $4 \times 10^{-4}$ $F^{-1}$ in an approximately linear manner. Since higher order terms are neglected in equation (2), conservatism can be achieved by evaluating all fluid properties at the inlet (or pool) temperature $T_{in}$.

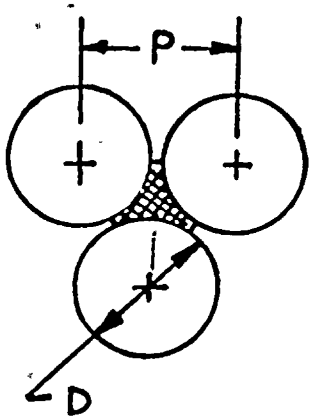
The flow driving pressure $\Delta p_d$ is to be balanced by flow losses for the circulation loop. For the closely packed configuration and long-term cooling, the only loss that requires consideration is the laminar flow loss for fuel rod friction. This loss can be estimated using an equivalent hydraulic diameter ($D_e$) for the channel.

$$\Delta p_1 = \frac{\rho v^2}{2g_c}\left(\frac{64}{R_e}\right)\frac{L'}{D_e} \tag{3}$$

where

$R_e = \frac{\rho v D_e}{\mu}$ is the Reynolds number of the flow ($R_e < 2000$ for laminar flow)

v is the average velocity in the channel, and L' is the total rod or channel length. For the packed triangular array, the flow channel as shown shaded in Figure 2 is approximately that of an equilateral triangle with



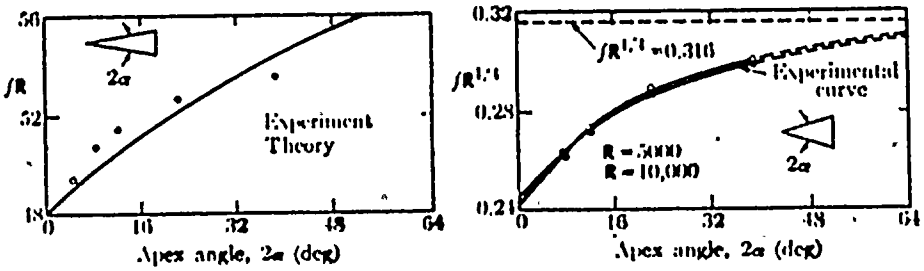

$$\text{Flow Area } A_f = \frac{\sqrt{3}}{4} P^2 - \frac{\pi}{8} D^2 \tag{4a}$$

$$\text{Wetted perimeter } P_{wet} = \frac{\pi D}{2} \tag{4b}$$

$$\text{and } D_e = \frac{4A_f}{P_{wet}} \tag{4c}$$

FIGURE 2
Flow Channel for
Closely Packed Triangular Lattice

Figure 3, (taken from reference 1) illustrates that $f \approx 57/R_e$ for the equilateral triangle. Equation (3) will then be conservative by $\sim 10\%$. Flow losses not considered here will easily be absorbed in this 10% margin. The expansion and contraction loss upon entering and leaving the tightly packed fuel region will be $\rho v^2 K/2g_c$ where the loss coefficient K is of order unity ($K \approx 2$ following reference 2, page 93). In all cases to be considered, this loss will be a small percentage of the total.




a b

FIGURE 3

Friction factors for fully developed flow in triangular ducts: (a) laminar flow ($fR_e = 64$ for circular pipes); (b) turbulent flow ($fR_e^{1/4} = 0.316$ for circular pipes)

Combining equations(2) and (3) results in expressions that can be easily solved for the velocity v, given the pool temperature, the coolant $\Delta T_h$ and the lattice geometry:

$$\Delta p_d = \Delta p_1 = \frac{1}{2}\,\beta\,\rho\,(\Delta T_h)\,\frac{g}{g_c}\,L = \frac{32\mu\,L'}{g_c\,D_e^2}\,v \qquad (5)$$

With $\Delta T_h$ specified and v found from equation (5), the decay heat to be removed by the coolant flowing in the channel (Figure 2) can be found using

$$q_{rod} = 2 \rho c_p v A_f (\Delta T_h) = 2 q_{channel} \qquad (6)$$

Since the fuel rod decay heat is now known and $q_o$, the rod thermal power is an easily calculated quantity, the resulting decay heat power fraction

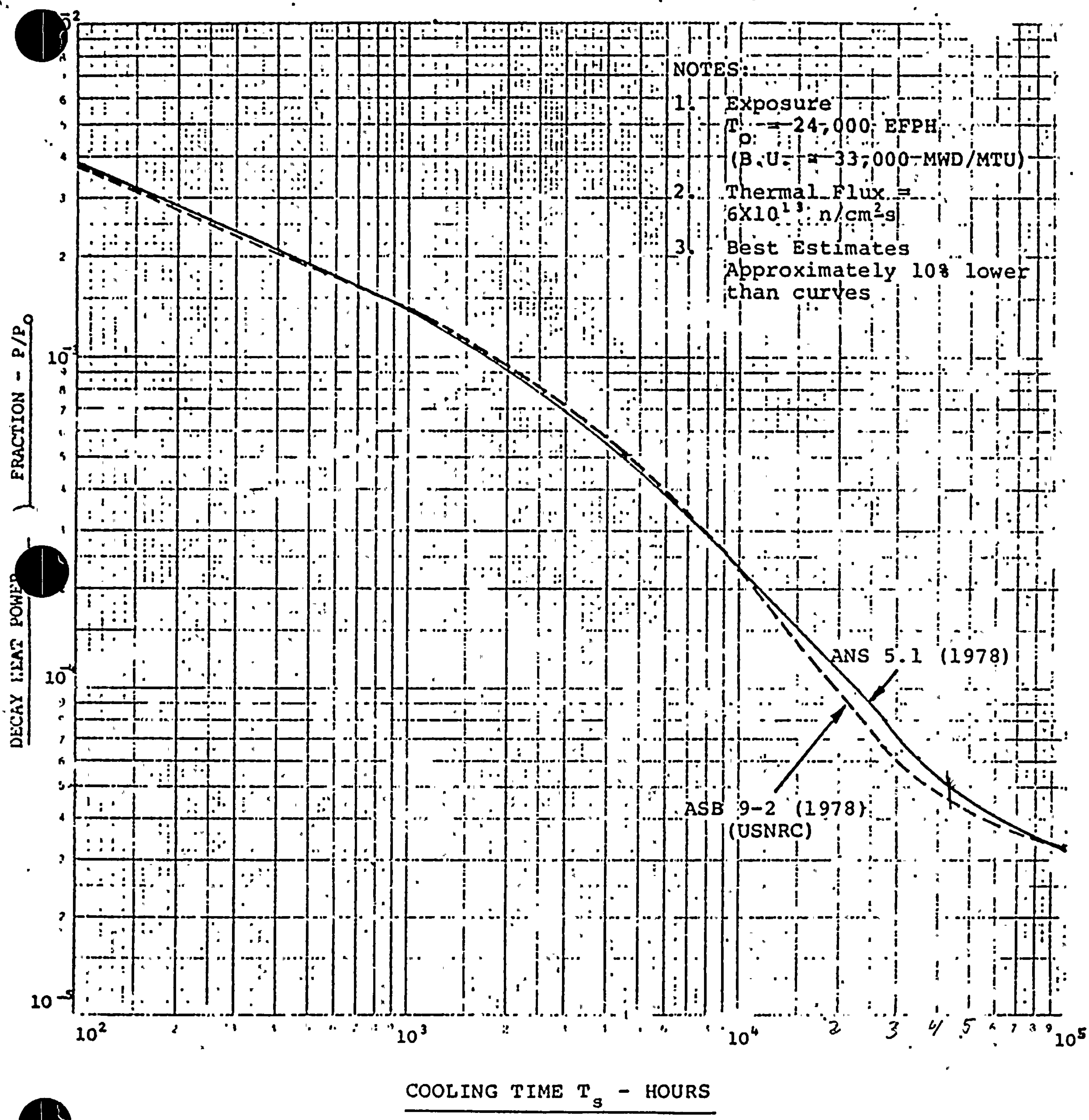
$$\frac{P}{P_o} = \frac{q_{rod}}{q_o} \qquad (7)$$

can be found. The ratio $P/P_o$ for typical LWR spent fuel is shown on the following page (Figure 4). Decay heat curves following ASB 9-2 (reference 3) and ANS 5.1 standard (reference 4) are shown. Both curves agree within the 10% margin allowed for the best estimates for most cooling times (The exception here is the interval 20,000 to 40,000 hours cooling time where ANS 5.1 includes a significant effect for neutron capture in $Cs^{133}$, thus producing the shielded nuclide $Cs^{134}$. ASB 9-2 includes no corrections for fission product absorption.)

The procedure outlined thus far would be sufficient for the analysis provided no limits were set on the clad surface temperature. If film boiling criteria are to be followed, the peak clad temperature must be evaluated.

DECAY HEAT CURVES FOR TYPICAL LWR SPENT FUEL
FRACTION OF OPERATING POWERS VERSUS COOLING TIME

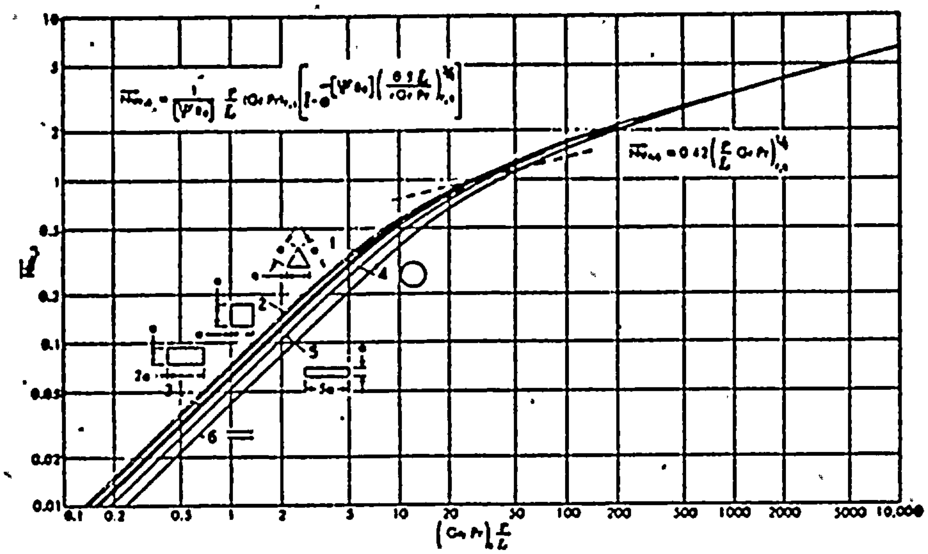
FIGURE 4



At the heated rod surface, there will be a film drop, $\Delta T_s$, related to the heat flux, $q''$, through the film coefficient, h,

$$q'' = h\,\Delta T_s \qquad (8)$$

The average h for free convection in vertical channels can be found from the following graph taken from Kreith, page 349, reference 5.




Free-convection heat transfer from the interior surfaces of vertical ducts having various cross-sectional geometries. ($\overline{Nu}_{r,r}$ as a function of $(Gr\,Pr)_{r,r}\,r/L$ for the cooling of the inner surface of vertical tubes according to Ref. 13 for different values of ($\psi$ Re). For infinitely long parallel plates (case b) ($\psi$ Re) = 24 (curve 6). For rectangular cross sections with the following proportions of the sides of the rectangles 1:1, 1:2, and 1:5, ($\psi$ Re) is 19.05, 15.55, and 14.22, respectively (curves 2, 3, and 5). For circular cross sections ($\psi$ Re) is 16 (curve 4), and for cross sections in the shape of an equilateral triangle ($\psi$ Re) = 13½ (curve 1). The points of intersection with the dashed line indicate the points where the cooling unit area of the horizontal cross-section is a maximum. (Courtesy of W. Elenbaas, N. V. Philips' Gloeilampenfabricken, Ref. 13)

FIGURE 5

In Figure 5, the Nusselt number is related to $\bar{h}$ by the average Nusselt number

$$\overline{N_u} = \frac{\bar{h}r}{k} \tag{9}$$

which depends on the Grashof number - Prandtl number product

$$G_r = \frac{\rho^2}{\mu^2} g \; \beta \quad (\Delta T_s) \; r^3 \tag{10a}$$

$$P_r = \frac{c_p \mu}{k} \tag{10b}$$

The equivalent radius of the channel is simply

$$r = \frac{D_e}{2} \tag{10c}$$

For the cases of interest in this analysis, the channel will be small enough so that

$$\overline{N}_u = \frac{\bar{h}r}{k} = \frac{1}{(\Psi R_e)} \text{[illegible]} \; r \; \text{[illegible]} \; G_r \; P_r \tag{11}$$

Where $(\Psi R_e)$ is a geometry dependent quantity. For the triangular duct $(\Psi R_e)$ = 13.333.

Assuming the heat flux varies slowly over the axial dimension so that $\overline{q''}/\bar{h}$ is nearly equal to $q''/h$ ($\overline{q''} = q_{rod}/\pi$ DL), equations (9) through (11) may be combined to yield an average film drop

$$\Delta T_s = \left[ \frac{8 \; (\Psi R_e) \; \mu \; q_{rod}}{\pi D \; (D_e{}^3) \; \rho^2 \; c_p \; g \; \beta} \right]^{\frac{1}{2}} \tag{12}$$

For Kreith's correlation, all fluid properties for $\overline{N}_u$ are to be evaluated at the surface temperature with the exception of

$\beta$, which is to be evaluated at the mixed fluid temperature. Although $\mu/\beta$ increases by a factor of 2 as the fluid temperature increases by 50°F, the weak (¼) dependence in equation (12) suggests that all properties could be evaluated at the inlet temperature $T_{in}$ for a conservative but still reasonably accurate estimate for the film drop.

Owing to the fuel rod's approximately uniform axial burnup at end of life, the location of the peak clad temperature will be near the channel exit and the peak clad temperature can be estimated as

$$T_{cladmax} = T_{out} + \Delta T_s \tag{13}$$

For heat flux distributions q" varying less rapidly than a sine curve ($F_z = \pi/2 = 1.57$), preliminary calculations indicate that the equation (13) estimate for $T_{cladmax}$ will be reasonable but yet conservatively high. Since $T_{sat}$ changes less than 10°F over half the fuel rod length, the exact location of $T_{cladmax}$ is not important and a further refinement in the model would not be necessary.

A FORTRAN computer program was written to solve the previous equations. Variations in the pool temperature (from 100 to 200°F in 10°F increments) and in the triangular ~~triangular~~ rod array pitch (from D to D+0.040" by 0.010" increments) were made for each run.

For each run, the coolant temperature increase ($\Delta T_h$) was held constant. Runs for $\Delta T_h$ = 30 F to 60 F (in 10°F increments) were made for rod diameters of .422 and .400 inches.

The output from these runs is given in the following pages. A program listing then follows.

For Ginna, the following data is needed and used as input to the program:

D = rod diameter = .422 or .400 inches

L = rod active length = 141 inches = 11.75 ft

L' = total rod length = 150 inches = 12.50 ft

$$q_o = \text{rod thermal power} = \frac{1520(3.412\times10^6)}{(121)(179)} = 240{,}000\ \frac{\text{BTU}}{\text{hr}}$$

and

$\Delta T_h$ = coolant increase (F) - changed each run

Coolant properties $\rho$ = 61 lbm/ft$^3$ and $c_p$ = 1.0 BTU/lbm-F were taken as constants over the range of temperatures of interest. The temperature varying properties were analytically fit as linear functions of temperatures using

$$\beta = [2.0 + 0.02\ (T_{in} - 100)] \times 10^{-4}\ F^{-1}$$

and

$$\frac{\beta\rho}{\mu} = 27.0 + .903\ (T_{in} - 100)\ \text{sec/ft}^2\text{-F}$$

Both are accurate at 100 and 200 F and agree within 3% and 10%, respectively, at 150 F.

NATURAL CIRCULATION IN SPENT FUEL
TRIANGULAR ARRAY - TIGHT PACKING

ROD DIAMETER IN INCHES = .422
COOLANT INCREASE IN CHANNEL = 30.00
POWER PER ROD IN REACTOR (BTU/HR)= 240000.0
ACTIVE LENGTH IN FEET = 11.75
TOTAL ROD LENGTH IN FEET = 12.50

| TIN (F) ***** | PITCH (IN) **** | EQ DIA (IN) ***** | DELTA P (LB/SF) ****** | VEL. (FT/S) ******* | ROD POWER (BTU/HR) ******** | TOUT (F) ***** | CLAD H (B/HSFF) ***** | TCLAD (F) ***** | POW. FRAC. (NONE) ********** |
|---|---|---|---|---|---|---|---|---|---|
| 100.0 | .422 | .0433 | 2.150 | .0050 | .328E 01 | 130.0 | .13 | 148.77 | .137E-04 |
| 100.0 | .432 | .0656 | 2.150 | .0115 | .114E 02 | 130.0 | .47 | 148.78 | .475E-04 |
| 100.0 | .442 | .0885 | 2.150 | .0208 | .279E 02 | 130.0 | 1.15 | 148.77 | .116E-03 |
| 100.0 | .452 | .1118 | 2.150 | .0332 | .564E 02 | 130.0 | 2.31 | 148.77 | .235E-03 |
| 100.0 | .462 | .1357 | 2.150 | .0490 | .101E 03 | 130.0 | 4.13 | 148.78 | .420E-03 |
| 120.0 | .422 | .0433 | 2.580 | .0083 | .547E 01 | 150.0 | .22 | 168.77 | .228E-04 |
| 120.0 | .432 | .0656 | 2.580 | .0191 | .190E 02 | 150.0 | .78 | 168.77 | .793E-04 |
| 120.0 | .442 | .0885 | 2.580 | .0347 | .466E 02 | 150.0 | 1.91 | 168.77 | .194E-03 |
| 120.0 | .452 | .1118 | 2.580 | .0555 | .941E 02 | 150.0 | 3.86 | 168.77 | .392E-03 |
| 120.0 | .462 | .1357 | 2.580 | .0817 | .168E 03 | 150.0 | 6.90 | 168.77 | .701E-03 |
| 140.0 | .422 | .0433 | 3.010 | .0117 | .766E 01 | 170.0 | .31 | 188.77 | .319E-04 |
| 140.0 | .432 | .0656 | 3.010 | .0268 | .266E 02 | 170.0 | 1.09 | 188.77 | .111E-03 |
| 140.0 | .442 | .0885 | 3.010 | .0486 | .653E 02 | 170.0 | 2.68 | 188.77 | .272E-03 |
| 140.0 | .452 | .1118 | 3.010 | .0777 | .132E 03 | 170.0 | 5.41 | 188.77 | .549E-03 |
| 140.0 | .462 | .1357 | 3.010 | .1145 | .236E 03 | 170.0 | 9.66 | 188.77 | .981E-03 |
| 160.0 | .422 | .0433 | 3.440 | .0150 | .985E 01 | 190.0 | .40 | 208.78 | .411E-04 |
| 160.0 | .432 | .0656 | 3.440 | .0344 | .343E 02 | 190.0 | 1.41 | 208.77 | .143E-03 |
| 160.0 | .442 | .0885 | 3.440 | .0626 | .839E 02 | 190.0 | 3.44 | 208.77 | .350E-03 |
| 160.0 | .452 | .1118 | 3.440 | .1000 | .169E 03 | 190.0 | 6.95 | 208.77 | .706E-03 |
| 160.0 | .462 | .1357 | 3.440 | .1472 | .303E 03 | 190.0 | 12.43 | 208.77 | .126E-02 |
| 180.0 | .422 | .0433 | 3.870 | .0183 | .120E 02 | 210.0 | .49 | 228.78 | .502E-04 |
| 180.0 | .432 | .0656 | 3.870 | .0421 | .419E 02 | 210.0 | 1.72 | 228.77 | .175E-03 |
| 180.0 | .442 | .0885 | 3.870 | .0765 | .103E 03 | 210.0 | 4.21 | 228.77 | .427E-03 |
| 180.0 | .452 | .1118 | 3.870 | .1222 | .207E 03 | 210.0 | 8.50 | 228.77 | .863E-03 |
| 180.0 | .462 | .1357 | 3.870 | .1800 | .370E 03 | 210.0 | 15.19 | 228.77 | .154E-02 |
| 200.0 | .422 | .0433 | 4.300 | .0217 | .142E 02 | 230.0 | .58 | 248.77 | .593E-04 |
| 200.0 | .432 | .0656 | 4.300 | .0498 | .495E 02 | 230.0 | 2.03 | 248.77 | .206E-03 |
| 200.0 | .442 | .0885 | 4.300 | .0904 | .121E 03 | 230.0 | 4.98 | 248.77 | .505E-03 |
| 200.0 | .452 | .1118 | 4.300 | .1444 | .245E 03 | 230.0 | 10.05 | 248.77 | .102E-02 |
| 200.0 | .462 | .1357 | 4.300 | .2127 | .438E 03 | 230.0 | 17.96 | 248.77 | .182E-02 |

NATURAL CIRCULATION IN SPENT FUEL
TRIANGULAR ARRAY - TIGHT PACKING

ROD DIAMETER IN INCHES = .422
COOLANT INCREASE IN CHANNEL = 40.00
POWER PER ROD IN REACTOR (BTU/HR)= 240000.0
ACTIVE LENGTH IN FEET = 11.75
TOTAL ROD LENGTH IN FEET = 12.50

| TIN (F) ***** | PITCH (IN) **** | EQ DIA (IN) ***** | DELTA P (LB/SF) ****** | VEL. (FT/S) ******* | ROD POWER (BTU/HR) ******** | TOUT (F) ***** | CLAD H (B/HSFF) ***** | TCLAD (F) ***** | POW. FRAC. (NONE) ********** |
|---|---|---|---|---|---|---|---|---|---|
| 100.0 | .422 | .0433 | 2.867 | .0067 | .583E 01 | 140.0 | .18 | 165.03 | .243E-04 |
| 100.0 | .432 | .0656 | 2.867 | .0153 | .203E 02 | 140.0 | .62 | 165.03 | .844E-04 |
| 100.0 | .442 | .0885 | 2.867 | .0277 | .496E 02 | 140.0 | 1.53 | 165.03 | .207E-03 |
| 100.0 | .452 | .1118 | 2.867 | .0443 | .100E 03 | 140.0 | 3.08 | 165.03 | .418E-03 |
| 100.0 | .462 | .1357 | 2.867 | .0653 | .179E 03 | 140.0 | 5.51 | 165.03 | .746E-03 |
| 120.0 | .422 | .0433 | 3.440 | .0111 | .972E 01 | 160.0 | .30 | 185.03 | .405E-04 |
| 120.0 | .432 | .0656 | 3.440 | .0255 | .338E 02 | 160.0 | 1.04 | 185.03 | .141E-03 |
| 120.0 | .442 | .0885 | 3.440 | .0463 | .828E 02 | 160.0 | 2.55 | 185.03 | .345E-03 |
| 120.0 | .452 | .1118 | 3.440 | .0740 | .167E 03 | 160.0 | 5.15 | 185.03 | .697E-03 |
| 120.0 | .462 | .1357 | 3.440 | .1089 | .299E 03 | 160.0 | 9.20 | 185.03 | .125E-02 |
| 140.0 | .422 | .0433 | 4.014 | .0156 | .136E 02 | 180.0 | .42 | 205.03 | .567E-04 |
| 140.0 | .432 | .0656 | 4.014 | .0357 | .474E 02 | 180.0 | 1.46 | 205.03 | .197E-03 |
| 140.0 | .442 | .0885 | 4.014 | .0649 | .116E 03 | 180.0 | 3.57 | 205.03 | .483E-03 |
| 140.0 | .452 | .1118 | 4.014 | .1036 | .234E 03 | 180.0 | 7.21 | 205.03 | .976E-03 |
| 140.0 | .462 | .1357 | 4.014 | .1526 | .419E 03 | 180.0 | 12.89 | 205.03 | .174E-02 |
| 160.0 | .422 | .0433 | 4.587 | .0200 | .175E 02 | 200.0 | .54 | 225.03 | .730E-04 |
| 160.0 | .432 | .0656 | 4.587 | .0459 | .609E 02 | 200.0 | 1.87 | 225.03 | .254E-03 |
| 160.0 | .442 | .0885 | 4.587 | .0834 | .149E 03 | 200.0 | 4.59 | 225.03 | .622E-03 |
| 160.0 | .452 | .1118 | 4.587 | .1333 | .301E 03 | 200.0 | 9.27 | 225.03 | .126E-02 |
| 160.0 | .462 | .1357 | 4.587 | .1963 | .539E 03 | 200.0 | 16.57 | 225.03 | .224E-02 |
| 180.0 | .422 | .0433 | 5.161 | .0244 | .214E 02 | 220.0 | .66 | 245.03 | .892E-04 |
| 180.0 | .432 | .0656 | 5.161 | .0561 | .745E 02 | 220.0 | 2.29 | 245.03 | .310E-03 |
| 180.0 | .442 | .0885 | 5.161 | .1020 | .182E 03 | 220.0 | 5.61 | 245.03 | .760E-03 |
| 180.0 | .452 | .1118 | 5.161 | .1629 | .368E 03 | 220.0 | 11.34 | 245.03 | .153E-02 |
| 180.0 | .462 | .1357 | 5.161 | .2399 | .658E 03 | 220.0 | 20.26 | 245.03 | .274E-02 |
| 200.0 | .422 | .0433 | 5.734 | .0289 | .253E 02 | 240.0 | .78 | 265.03 | .105E-03 |
| 200.0 | .432 | .0656 | 5.734 | .0663 | .880E 02 | 240.0 | 2.71 | 265.03 | .367E-03 |
| 200.0 | .442 | .0885 | 5.734 | .1205 | .216E 03 | 240.0 | 6.63 | 265.03 | .898E-03 |
| 200.0 | .452 | .1118 | 5.734 | .1926 | .435E 03 | 240.0 | 13.40 | 265.03 | .181E-02 |
| 200.0 | .462 | .1357 | 5.734 | .2836 | .778E 03 | 240.0 | 23.95 | 265.03 | .324E-02 |

NATURAL CIRCULATION IN SPENT FUEL
TRIANGULAR ARRAY - TIGHT PACKING

ROD DIAMETER IN INCHES = .422
COOLANT INCREASE IN CHANNEL = 50.00
POWER PER ROD IN REACTOR (BTU/HR)= 240000.0
CTIVE LENGTH IN FEET = 11.75
OTAL ROD LENGTH IN FEET = 12.50

| TIN (F) ***** | PITCH (IN) **** | EQ DIA (IN) ***** | DELTA P (LB/SF) ****** | VEL. (FT/S) ******* | ROD POWER (BTU/HR) ******** | TOUT (F) ***** | CLAD H (B/HSFF) ***** | TCLAD (F) ***** | POW. FRAC. (NONE) ********** |
|---|---|---|---|---|---|---|---|---|---|
| 100.0 | .422 | .0433 | 3.584 | .0083 | .910E 01 | 150.0 | .22 | 181.29 | .379E-04 |
| 100.0 | .432 | .0656 | 3.584 | .0191 | .317E 02 | 150.0 | .78 | 181.29 | .132E-03 |
| 100.0 | .442 | .0885 | 3.584 | .0347 | .775E 02 | 150.0 | 1.91 | 181.29 | .323E-03 |
| 100.0 | .452 | .1118 | 3.584 | .0554 | .157E 03 | 150.0 | 3.86 | 181.29 | .652E-03 |
| 100.0 | .462 | .1357 | 3.584 | .0816 | .280E 03 | 150.0 | 6.89 | 181.29 | .117E-02 |
| 120.0 | .422 | .0433 | 4.300 | .0139 | .152E 02 | 170.0 | .37 | 201.29 | .633E-04 |
| 120.0 | .432 | .0656 | 4.300 | .0319 | .528E 02 | 170.0 | 1.30 | 201.29 | .220E-03 |
| 120.0 | .442 | .0885 | 4.300 | .0579 | .129E 03 | 170.0 | 3.19 | 201.29 | .539E-03 |
| 120.0 | .452 | .1118 | 4.300 | .0925 | .261E 03 | 170.0 | 6.43 | 201.29 | .109E-02 |
| 120.0 | .462 | .1357 | 4.300 | .1362 | .467E 03 | 170.0 | 11.50 | 201.29 | .195E-02 |
| 140.0 | .422 | .0433 | 5.017 | .0194 | .213E 02 | 190.0 | .52 | 221.29 | .887E-04 |
| 140.0 | .432 | .0656 | 5.017 | .0446 | .740E 02 | 190.0 | 1.82 | 221.29 | .308E-03 |
| 140.0 | .442 | .0885 | 5.017 | .0811 | .181E 03 | 190.0 | 4.46 | 221.29 | .755E-03 |
| 140.0 | .452 | .1118 | 5.017 | .1295 | .366E 03 | 190.0 | 9.01 | 221.29 | .153E-02 |
| 140.0 | .462 | .1357 | 5.017 | .1908 | .654E 03 | 190.0 | 16.11 | 221.29 | .273E-02 |
| 160.0 | .422 | .0433 | 5.734 | .0250 | .274E 02 | 210.0 | .67 | 241.29 | .114E-03 |
| 160.0 | .432 | .0656 | 5.734 | .0574 | .952E 02 | 210.0 | 2.34 | 241.29 | .397E-03 |
| 160.0 | .442 | .0885 | 5.734 | .1043 | .233E 03 | 210.0 | 5.74 | 241.29 | .971E-03 |
| 160.0 | .452 | .1118 | 5.734 | .1666 | .471E 03 | 210.0 | 11.59 | 241.29 | .196E-02 |
| 60.0 | .462 | .1357 | 5.734 | .2453 | .841E 03 | 210.0 | 20.72 | 241.29 | .351E-02 |
| 180.0 | .422 | .0433 | 6.451 | .0306 | .335E 02 | 230.0 | .82 | 261.29 | .139E-03 |
| 180.0 | .432 | .0656 | 6.451 | .0702 | .116E 03 | 230.0 | 2.86 | 261.29 | .485E-03 |
| 180.0 | .442 | .0885 | 6.451 | .1275 | .285E 03 | 230.0 | 7.02 | 261.29 | .119E-02 |
| 180.0 | .452 | .1118 | 6.451 | .2037 | .576E 03 | 230.0 | 14.17 | 261.29 | .240E-02 |
| 180.0 | .462 | .1357 | 6.451 | .2999 | .103E 04 | 230.0 | 25.32 | 261.29 | .429E-02 |
| 200.0 | .422 | .0433 | 7.167 | .0361 | .395E 02 | 250.0 | .97 | 281.29 | .165E-03 |
| 200.0 | .432 | .0656 | 7.167 | .0829 | .138E 03 | 250.0 | 3.39 | 281.29 | .573E-03 |
| 200.0 | .442 | .0885 | 7.167 | .1507 | .337E 03 | 250.0 | 8.29 | 281.29 | .140E-02 |
| 200.0 | .452 | .1118 | 7.167 | .2407 | .680E 03 | 250.0 | 16.75 | 281.29 | .283E-02 |
| 200.0 | .462 | .1357 | 7.167 | .3545 | .122E 04 | 250.0 | 29.93 | 281.29 | .507E-02 |

NATURAL CIRCULATION IN SPENT FUEL
TRIANGULAR ARRAY - TIGHT PACKING

ROD DIAMETER IN INCHES = .422
COOLANT INCREASE IN CHANNEL = 60.00
POWER PER ROD IN REACTOR (BTU/HR)= 240000.0
CTIVE LENGTH IN FEET = 11.75
OTAL ROD LENGTH IN FEET = 12.50

| TIN (F) ***** | PITCH (IN) **** | EQ DIA (IN) ***** | DELTA P (LB/SF) ****** | VEL. (FT/S) ******* | ROD POWER (BTU/HR) ******** | TOUT (F) ***** | CLAD H (B/HSFF) ***** | TCLAD (F) ***** | POW. FRAC. (NONE) ********** |
|---|---|---|---|---|---|---|---|---|---|
| 100.0 | .422 | .0433 | 4.300 | .0100 | .131E 02 | 160.0 | .27 | 197.55 | .546E-04 |
| 100.0 | .432 | .0656 | 4.300 | .0229 | .456E 02 | 160.0 | .94 | 197.55 | .190E-03 |
| 100.0 | .442 | .0885 | 4.300 | .0416 | .112E 03 | 160.0 | 2.29 | 197.55 | .465E-03 |
| 100.0 | .452 | .1118 | 4.300 | .0665 | .225E 03 | 160.0 | 4.63 | 197.55 | .940E-03 |
| 100.0 | .462 | .1357 | 4.300 | .0979 | .403E 03 | 160.0 | 8.27 | 197.55 | .168E-02 |
| 120.0 | .422 | .0433 | 5.161 | .0167 | .219E 02 | 180.0 | .45 | 217.55 | .912E-04 |
| 120.0 | .432 | .0656 | 5.161 | .0382 | .761E 02 | 180.0 | 1.56 | 217.55 | .317E-03 |
| 120.0 | .442 | .0885 | 5.161 | .0694 | .186E 03 | 180.0 | 3.82 | 217.55 | .776E-03 |
| 120.0 | .452 | .1118 | 5.161 | .1110 | .376E 03 | 180.0 | 7.72 | 217.55 | .157E-02 |
| 120.0 | .462 | .1357 | 5.161 | .1634 | .673E 03 | 180.0 | 13.80 | 217.55 | .280E-02 |
| 140.0 | .422 | .0433 | 6.021 | .0233 | .306E 02 | 200.0 | .63 | 237.55 | .128E-03 |
| 140.0 | .432 | .0656 | 6.021 | .0535 | .107E 03 | 200.0 | 2.19 | 237.55 | .444E-03 |
| 140.0 | .442 | .0885 | 6.021 | .0973 | .261E 03 | 200.0 | 5.35 | 237.55 | .109E-02 |
| 140.0 | .452 | .1118 | 6.021 | .1554 | .527E 03 | 200.0 | 10.81 | 237.55 | .220E-02 |
| 140.0 | .462 | .1357 | 6.021 | .2289 | .942E 03 | 200.0 | 19.33 | 237.55 | .393E-02 |
| 160.0 | .422 | .0433 | 6.881 | .0300 | .394E 02 | 220.0 | .81 | 257.55 | .164E-03 |
| 160.0 | .432 | .0656 | 6.881 | .0689 | .137E 03 | 220.0 | 2.81 | 257.55 | .571E-03 |
| 160.0 | .442 | .0885 | 6.881 | .1251 | .336E 03 | 220.0 | 6.89 | 257.55 | .140E-02 |
| 160.0 | .452 | .1118 | 6.881 | .1999 | .678E 03 | 220.0 | 13.91 | 257.55 | .282E-02 |
| 60.0 | .462 | .1357 | 6.881 | .2944 | .121E 04 | 220.0 | 24.86 | 257.55 | .505E-02 |
| 180.0 | .422 | .0433 | 7.741 | .0367 | .482E 02 | 240.0 | .99 | 277.55 | .201E-03 |
| 180.0 | .432 | .0656 | 7.741 | .0842 | .168E 03 | 240.0 | 3.44 | 277.55 | .698E-03 |
| 180.0 | .442 | .0885 | 7.741 | .1530 | .410E 03 | 240.0 | 8.42 | 277.55 | .171E-02 |
| 180.0 | .452 | .1118 | 7.741 | .2444 | .829E 03 | 240.0 | 17.00 | 277.55 | .345E-02 |
| 180.0 | .462 | .1357 | 7.741 | .3599 | .148E 04 | 240.0 | 30.39 | 277.55 | .617E-02 |
| 200.0 | .422 | .0433 | 8.601 | .0433 | .569E 02 | 260.0 | 1.17 | 297.55 | .237E-03 |
| 200.0 | .432 | .0656 | 8.601 | .0995 | .198E 03 | 260.0 | 4.06 | 297.55 | .825E-03 |
| 200.0 | .442 | .0885 | 8.601 | .1808 | .485E 03 | 260.0 | 9.95 | 297.55 | .202E-02 |
| 200.0 | .452 | .1118 | 8.601 | .2889 | .980E 03 | 260.0 | 20.10 | 297.55 | .408E-02 |
| 200.0 | .462 | .1357 | 8.601 | .4254 | .175E 04 | 260.0 | 35.92 | 297.55 | .730E-02 |

NATURAL CIRCULATION IN SPENT FUEL
TRIANGULAR ARRAY - TIGHT PACKING

ROD DIAMETER IN INCHES = .400
COOLANT INCREASE IN CHANNEL = 30.00
POWER PER ROD IN REACTOR (BTU/HR)= 240000.0
ACTIVE LENGTH IN FEET = 11.75
TOTAL ROD LENGTH IN FEET = 12.50

| TIN (F) ***** | PITCH (IN) **** | EQ DIA (IN) ***** | DELTA P (LB/SF) ****** | VEL. (FT/S) ******* | ROD POWER (BTU/HR) ******** | TOUT (F) ***** | CLAD H (B/HSFF) ***** | TCLAD (F) ***** | POW. FRAC. (NONE) ********** |
|---|---|---|---|---|---|---|---|---|---|
| 100.0 | .400 | .0411 | 2.150 | .0045 | .265E 01 | 130.0 | .11 | 148.77 | .110E-04 |
| 100.0 | .410 | .0634 | 2.150 | .0107 | .973E 01 | 130.0 | .42 | 148.78 | .406E-04 |
| 100.0 | .420 | .0863 | 2.150 | .0198 | .245E 02 | 130.0 | 1.06 | 148.77 | .102E-03 |
| 100.0 | .430 | .1097 | 2.150 | .0320 | .504E 02 | 130.0 | 2.18 | 148.77 | .210E-03 |
| 100.0 | .440 | .1337 | 2.150 | .0475 | .913E 02 | 130.0 | 3.95 | 148.78 | .380E-03 |
| 120.0 | .400 | .0411 | 2.580 | .0075 | .441E 01 | 150.0 | .19 | 168.77 | .184E-04 |
| 120.0 | .410 | .0634 | 2.580 | .0178 | .162E 02 | 150.0 | .70 | 168.77 | .677E-04 |
| 120.0 | .420 | .0863 | 2.580 | .0330 | .409E 02 | 150.0 | 1.77 | 168.77 | .171E-03 |
| 120.0 | .430 | .1097 | 2.580 | .0534 | .842E 02 | 150.0 | 3.64 | 168.77 | .351E-03 |
| 120.0 | .440 | .1337 | 2.580 | .0793 | .152E 03 | 150.0 | 6.59 | 168.77 | .635E-03 |
| 140.0 | .400 | .0411 | 3.010 | .0105 | .618E 01 | 170.0 | .27 | 188.77 | .258E-04 |
| 140.0 | .410 | .0634 | 3.010 | .0250 | .228E 02 | 170.0 | .98 | 188.77 | .948E-04 |
| 140.0 | .420 | .0863 | 3.010 | .0463 | .573E 02 | 170.0 | 2.48 | 188.77 | .239E-03 |
| 140.0 | .430 | .1097 | 3.010 | .0748 | .118E 03 | 170.0 | 5.10 | 188.77 | .491E-03 |
| 140.0 | .440 | .1337 | 3.010 | .1111 | .213E 03 | 170.0 | 9.24 | 188.77 | .889E-03 |
| 160.0 | .400 | .0411 | 3.440 | .0135 | .795E 01 | 190.0 | .34 | 208.77 | .331E-04 |
| 160.0 | .410 | .0634 | 3.440 | .0321 | .293E 02 | 190.0 | 1.27 | 208.77 | .122E-03 |
| 160.0 | .420 | .0863 | 3.440 | .0595 | .738E 02 | 190.0 | 3.19 | 208.77 | .307E-03 |
| 160.0 | .430 | .1097 | 3.440 | .0962 | .152E 03 | 190.0 | 6.56 | 208.77 | .632E-03 |
| 160.0 | .440 | .1337 | 3.440 | .1428 | .274E 03 | 190.0 | 11.88 | 208.77 | .114E-02 |
| 180.0 | .400 | .0411 | 3.870 | .0165 | .972E 01 | 210.0 | .42 | 228.77 | .405E-04 |
| 180.0 | .410 | .0634 | 3.870 | .0393 | .358E 02 | 210.0 | 1.55 | 228.77 | .149E-03 |
| 180.0 | .420 | .0863 | 3.870 | .0727 | .902E 02 | 210.0 | 3.90 | 228.77 | .376E-03 |
| 180.0 | .430 | .1097 | 3.870 | .1176 | .185E 03 | 210.0 | 8.03 | 228.77 | .772E-03 |
| 180.0 | .440 | .1337 | 3.870 | .1746 | .336E 03 | 210.0 | 14.52 | 228.77 | .140E-02 |
| 200.0 | .400 | .0411 | 4.300 | .0195 | .115E 02 | 230.0 | .50 | 248.77 | .479E-04 |
| 200.0 | .410 | .0634 | 4.300 | .0464 | .423E 02 | 230.0 | 1.83 | 248.77 | .176E-03 |
| 200.0 | .420 | .0863 | 4.300 | .0860 | .107E 03 | 230.0 | 4.61 | 248.77 | .444E-03 |
| 200.0 | .430 | .1097 | 4.300 | .1390 | .219E 03 | 230.0 | 9.49 | 248.77 | .913E-03 |
| 200.0 | .440 | .1337 | 4.300 | .2064 | .397E 03 | 230.0 | 17.17 | 248.77 | .165E-02 |

NATURAL CIRCULATION IN SPENT FUEL
TRIANGULAR ARRAY - TIGHT PACKING

ROD DIAMETER IN INCHES = .400
COOLANT INCREASE IN CHANNEL = 40.00
POWER PER ROD IN REACTOR (BTU/HR)= 240000.0
\CTIVE LENGTH IN FEET = 11.75
TOTAL ROD LENGTH IN FEET = 12.50

| TIN (F) ***** | PITCH (IN) **** | EQ DIA (IN) ***** | DELTA P (LB/SF) ****** | VEL. (FT/S) ******* | ROD POWER (BTU/HR) ******** | TOUT (F) ***** | CLAD H (B/HSFF) ***** | TCLAD (F) ***** | POW. FRAC. (NONE) ********** |
|---|---|---|---|---|---|---|---|---|---|
| 100.0 | .400 | .0411 | 2.867 | .0060 | .470E 01 | 140.0 | .15 | 165.03 | .196E-04 |
| 100.0 | .410 | .0634 | 2.867 | .0142 | .173E 02 | 140.0 | .56 | 165.03 | .721E-04 |
| 100.0 | .420 | .0863 | 2.867 | .0264 | .436E 02 | 140.0 | 1.42 | 165.03 | .182E-03 |
| 100.0 | .430 | .1097 | 2.867 | .0427 | .897E 02 | 140.0 | 2.91 | 165.03 | .374E-03 |
| 100.0 | .440 | .1337 | 2.867 | .0633 | .162E 03 | 140.0 | 5.27 | 165.03 | .676E-03 |
| 120.0 | .400 | .0411 | 3.440 | .0100 | .785E 01 | 160.0 | .25 | 185.03 | .327E-04 |
| 120.0 | .410 | .0634 | 3.440 | .0238 | .289E 02 | 160.0 | .94 | 185.03 | .120E-03 |
| 120.0 | .420 | .0863 | 3.440 | .0440 | .728E 02 | 160.0 | 2.36 | 185.03 | .303E-03 |
| 120.0 | .430 | .1097 | 3.440 | .0712 | .150E 03 | 160.0 | 4.86 | 185.03 | .624E-03 |
| 120.0 | .440 | .1337 | 3.440 | .1057 | .271E 03 | 160.0 | 8.79 | 185.03 | .113E-02 |
| 140.0 | .400 | .0411 | 4.014 | .0140 | .110E 02 | 180.0 | .36 | 205.03 | .458E-04 |
| 140.0 | .410 | .0634 | 4.014 | .0333 | .404E 02 | 180.0 | 1.31 | 205.03 | .169E-03 |
| 140.0 | .420 | .0863 | 4.014 | .0617 | .102E 03 | 180.0 | 3.31 | 205.03 | .425E-03 |
| 140.0 | .430 | .1097 | 4.014 | .0997 | .210E 03 | 180.0 | 6.81 | 205.03 | .873E-03 |
| 140.0 | .440 | .1337 | 4.014 | .1481 | .379E 03 | 180.0 | 12.32 | 205.03 | .158E-02 |
| 160.0 | .400 | .0411 | 4.587 | .0180 | .141E 02 | 200.0 | .46 | 225.03 | .589E-04 |
| 160.0 | .410 | .0634 | 4.587 | .0428 | .520E 02 | 200.0 | 1.69 | 225.03 | .217E-03 |
| 160.0 | .420 | .0863 | 4.587 | .0793 | .131E 03 | 200.0 | 4.26 | 225.03 | .546E-03 |
| 160.0 | .430 | .1097 | 4.587 | .1282 | .270E 03 | 200.0 | 8.75 | 225.03 | .112E-02 |
| 160.0 | .440 | .1337 | 4.587 | .1904 | .488E 03 | 200.0 | 15.84 | 225.03 | .203E-02 |
| 180.0 | .400 | .0411 | 5.161 | .0220 | .173E 02 | 220.0 | .56 | 245.03 | .720E-04 |
| 180.0 | .410 | .0634 | 5.161 | .0523 | .636E 02 | 220.0 | 2.06 | 245.03 | .265E-03 |
| 180.0 | .420 | .0863 | 5.161 | .0970 | .160E 03 | 220.0 | 5.20 | 245.03 | .668E-03 |
| 180.0 | .430 | .1097 | 5.161 | .1568 | .330E 03 | 220.0 | 10.70 | 245.03 | .137E-02 |
| 180.0 | .440 | .1337 | 5.161 | .2328 | .596E 03 | 220.0 | 19.36 | 245.03 | .249E-02 |
| 200.0 | .400 | .0411 | 5.734 | .0260 | .204E 02 | 240.0 | .66 | 265.03 | .851E-04 |
| 200.0 | .410 | .0634 | 5.734 | .0619 | .752E 02 | 240.0 | 2.44 | 265.03 | .313E-03 |
| 200.0 | .420 | .0863 | 5.734 | .1146 | .189E 03 | 240.0 | 6.15 | 265.03 | .789E-03 |
| 200.0 | .430 | .1097 | 5.734 | .1853 | .390E 03 | 240.0 | 12.65 | 265.03 | .162E-02 |
| 200.0 | .440 | .1337 | 5.734 | .2752 | .705E 03 | 240.0 | 22.89 | 265.03 | .294E-02 |

NATURAL CIRCULATION IN SPENT FUEL
TRIANGULAR ARRAY - TIGHT PACKING

ROD DIAMETER IN INCHES = .400
COOLANT INCREASE IN CHANNEL = 50.00
POWER PER ROD IN REACTOR (BTU/HR)= 240000.0
ACTIVE LENGTH IN FEET = 11.75
TOTAL ROD LENGTH IN FEET = 12.50

| TIN (F) ***** | PITCH (IN) **** | EQ DIA (IN) ***** | DELTA P (LB/SF) ****** | VEL. (FT/S) ******* | ROD POWER (BTU/HR) ******** | TOUT (F) ***** | CLAD H (B/HSFF) ***** | TCLAD (F) ***** | POW. FRAC. (NONE) ********** |
|---|---|---|---|---|---|---|---|---|---|
| 100.0 | .400 | .0411 | 3.584 | .0075 | .735E 01 | 150.0 | .19 | 181.29 | .306E-04 |
| 100.0 | .410 | .0634 | 3.584 | .0178 | .270E 02 | 150.0 | .70 | 181.29 | .113E-03 |
| 100.0 | .420 | .0863 | 3.584 | .0330 | .681E 02 | 150.0 | 1.77 | 181.29 | .284E-03 |
| 100.0 | .430 | .1097 | 3.584 | .0533 | .140E 03 | 150.0 | 3.64 | 181.29 | .584E-03 |
| 100.0 | .440 | .1337 | 3.584 | .0792 | .254E 03 | 150.0 | 6.59 | 181.29 | .106E-02 |
| 120.0 | .400 | .0411 | 4.300 | .0125 | .123E 02 | 170.0 | .32 | 201.29 | .511E-04 |
| 120.0 | .410 | .0634 | 4.300 | .0297 | .451E 02 | 170.0 | 1.17 | 201.29 | .188E-03 |
| 120.0 | .420 | .0863 | 4.300 | .0550 | .114E 03 | 170.0 | 2.95 | 201.29 | .474E-03 |
| 120.0 | .430 | .1097 | 4.300 | .0890 | .234E 03 | 170.0 | 6.07 | 201.29 | .974E-03 |
| 120.0 | .440 | .1337 | 4.300 | .1321 | .423E 03 | 170.0 | 10.99 | 201.29 | .176E-02 |
| 140.0 | .400 | .0411 | 5.017 | .0175 | .172E 02 | 190.0 | .45 | 221.29 | .716E-04 |
| 140.0 | .410 | .0634 | 5.017 | .0416 | .632E 02 | 190.0 | 1.64 | 221.29 | .263E-03 |
| 140.0 | .420 | .0863 | 5.017 | .0771 | .159E 03 | 190.0 | 4.14 | 221.29 | .664E-03 |
| 140.0 | .430 | .1097 | 5.017 | .1246 | .328E 03 | 190.0 | 8.51 | 221.29 | .136E-02 |
| 140.0 | .440 | .1337 | 5.017 | .1851 | .593E 03 | 190.0 | 15.40 | 221.29 | .247E-02 |
| 160.0 | .400 | .0411 | 5.734 | .0225 | .221E 02 | 210.0 | .57 | 241.29 | .921E-04 |
| 160.0 | .410 | .0634 | 5.734 | .0535 | .813E 02 | 210.0 | 2.11 | 241.29 | .339E-03 |
| 160.0 | .420 | .0863 | 5.734 | .0991 | .205E 03 | 210.0 | 5.32 | 241.29 | .854E-03 |
| 160.0 | .430 | .1097 | 5.734 | .1603 | .421E 03 | 210.0 | 10.94 | 241.29 | .176E-02 |
| 160.0 | .440 | .1337 | 5.734 | .2381 | .762E 03 | 210.0 | 19.80 | 241.29 | .318E-02 |
| 180.0 | .400 | .0411 | 6.451 | .0275 | .270E 02 | 230.0 | .70 | 261.29 | .113E-03 |
| 180.0 | .410 | .0634 | 6.451 | .0654 | .994E 02 | 230.0 | 2.58 | 261.29 | .414E-03 |
| 180.0 | .420 | .0863 | 6.451 | .1212 | .250E 03 | 230.0 | 6.51 | 261.29 | .104E-02 |
| 180.0 | .430 | .1097 | 6.451 | .1960 | .515E 03 | 230.0 | 13.38 | 261.29 | .215E-02 |
| 180.0 | .440 | .1337 | 6.451 | .2910 | .932E 03 | 230.0 | 24.20 | 261.29 | .388E-02 |
| 200.0 | .400 | .0411 | 7.167 | .0325 | .319E 02 | 250.0 | .83 | 281.29 | .133E-03 |
| 200.0 | .410 | .0634 | 7.167 | .0773 | .117E 03 | 250.0 | 3.05 | 281.29 | .489E-03 |
| 200.0 | .420 | .0863 | 7.167 | .1433 | .296E 03 | 250.0 | 7.69 | 281.29 | .123E-02 |
| 200.0 | .430 | .1097 | 7.167 | .2316 | .609E 03 | 250.0 | 15.81 | 281.29 | .254E-02 |
| 200.0 | .440 | .1337 | 7.167 | .3440 | .110E 04 | 250.0 | 28.61 | 281.29 | .459E-02 |

NATURAL CIRCULATION IN SPENT FUEL
TRIANGULAR ARRAY - TIGHT PACKING

ROD DIAMETER IN INCHES = .400
COOLANT INCREASE IN CHANNEL = 60.00
POWER PER ROD IN REACTOR (BTU/HR)= 240000.0
ACTIVE LENGTH IN FEET = 11.75
TOTAL ROD LENGTH IN FEET = 12.50

| TIN (F) ***** | PITCH (IN) **** | EQ DIA (IN) ***** | DELTA P (LB/SF) ****** | VEL. (FT/S) ******* | ROD POWER (BTU/HR) ******** | TOUT (F) ***** | CLAD H (B/HSFF) ***** | TCLAD (F) ***** | POW. FRAC. (NONE) ********** |
|---|---|---|---|---|---|---|---|---|---|
| 100.0 | .400 | .0411 | 4.300 | .0090 | .106E 02 | 160.0 | .23 | 197.55 | .441E-04 |
| 100.0 | .410 | .0634 | 4.300 | .0214 | .389E 02 | 160.0 | .84 | 197.55 | .162E-03 |
| 100.0 | .420 | .0863 | 4.300 | .0396 | .981E 02 | 160.0 | 2.12 | 197.55 | .409E-03 |
| 100.0 | .430 | .1097 | 4.300 | .0640 | .202E 03 | 160.0 | 4.37 | 197.55 | .841E-03 |
| 100.0 | .440 | .1337 | 4.300 | .0950 | .365E 03 | 160.0 | 7.90 | 197.55 | .152E-02 |
| 120.0 | .400 | .0411 | 5.161 | .0150 | .177E 02 | 180.0 | .38 | 217.55 | .736E-04 |
| 120.0 | .410 | .0634 | 5.161 | .0357 | .650E 02 | 180.0 | 1.41 | 217.55 | .271E-03 |
| 120.0 | .420 | .0863 | 5.161 | .0660 | .164E 03 | 180.0 | 3.54 | 217.55 | .682E-03 |
| 120.0 | .430 | .1097 | 5.161 | .1068 | .337E 03 | 180.0 | 7.29 | 217.55 | .140E-02 |
| 120.0 | .440 | .1337 | 5.161 | .1586 | .609E 03 | 180.0 | 13.19 | 217.55 | .254E-02 |
| 140.0 | .400 | .0411 | 6.021 | .0210 | .247E 02 | 200.0 | .54 | 237.55 | .103E-03 |
| 140.0 | .410 | .0634 | 6.021 | .0499 | .910E 02 | 200.0 | 1.97 | 237.55 | .379E-03 |
| 140.0 | .420 | .0863 | 6.021 | .0925 | .229E 03 | 200.0 | 4.96 | 237.55 | .956E-03 |
| 140.0 | .430 | .1097 | 6.021 | .1496 | .472E 03 | 200.0 | 10.21 | 237.55 | .197E-02 |
| 140.0 | .440 | .1337 | 6.021 | .2221 | .854E 03 | 200.0 | 18.47 | 237.55 | .356E-02 |
| 160.0 | .400 | .0411 | 6.881 | .0270 | .318E 02 | 220.0 | .69 | 257.55 | .133E-03 |
| 160.0 | .410 | .0634 | 6.881 | .0642 | .117E 03 | 220.0 | 2.53 | 257.55 | .488E-03 |
| 160.0 | .420 | .0863 | 6.881 | .1190 | .295E 03 | 220.0 | 6.39 | 257.55 | .123E-02 |
| 160.0 | .430 | .1097 | 6.881 | .1924 | .607E 03 | 220.0 | 13.13 | 257.55 | .253E-02 |
| 160.0 | .440 | .1337 | 6.881 | .2857 | .110E 04 | 220.0 | 23.76 | 257.55 | .457E-02 |
| 180.0 | .400 | .0411 | 7.741 | .0329 | .389E 02 | 240.0 | .84 | 277.55 | .162E-03 |
| 180.0 | .410 | .0634 | 7.741 | .0785 | .143E 03 | 240.0 | 3.10 | 277.55 | .596E-03 |
| 180.0 | .420 | .0863 | 7.741 | .1454 | .361E 03 | 240.0 | 7.81 | 277.55 | .150E-02 |
| 180.0 | .430 | .1097 | 7.741 | .2352 | .742E 03 | 240.0 | 16.05 | 277.55 | .309E-02 |
| 180.0 | .440 | .1337 | 7.741 | .3492 | .134E 04 | 240.0 | 29.05 | 277.55 | .559E-02 |
| 200.0 | .400 | .0411 | 8.601 | .0389 | .460E 02 | 260.0 | .99 | 297.55 | .192E-03 |
| 200.0 | .410 | .0634 | 8.601 | .0928 | .169E 03 | 260.0 | 3.66 | 297.55 | .705E-03 |
| 200.0 | .420 | .0863 | 8.601 | .1719 | .426E 03 | 260.0 | 9.23 | 297.55 | .178E-02 |
| 200.0 | .430 | .1097 | 8.601 | .2780 | .877E 03 | 260.0 | 18.97 | 297.55 | .365E-02 |
| 200.0 | .440 | .1337 | 8.601 | .4128 | .159E 04 | 260.0 | 34.33 | 297.55 | .661E-02 |

```
C       FORTRAN PROGRAM
C       NATURAL RECIRCULATION IN SPENT FUEL
C       TRIANGULAR ARRAY-TIGHT PACKING
C       ****
C       PRE-INPUT DATA, MAY BE CHANGED FOR OTHER CONDITIONS
C       THE FOLLOWING VALUES ARE USED AS DEFAULT VALUES IF AN ERROR
C       OCCURS WHILE ATEMPTING TO READ VALUES FROM FOR01.DA
        XL=11.75
        XLP=12.50
        G=32.174
        SR=13.3333
        D=0.422
        Q0=240000.
        DT=60.
        RO=61.
C       ****
C       INPUT DATA IS READ FROM FILE FOR01.DA FREE FORMAT AND MUST BE
C       IN THE FOLLOWING ORDER:

C       XL=ACTIVE LENGTH OF FUEL ROD
C       XLP=    TOTAL LENGTH OF ROD
C       D=      DIA. OF FUEL ROD
C       Q0=     INITIAL POWER LEVEL PER ROD IN REACTOR
C       DT=     COOLANT INCREASE IN CHANNEL
C
C       IF AN ERROR OCCURS WHEN READING DATA FROM FOR01.DA
C       THE PROGRAM SENDS AN ERROR MESSAGE TO THE SCREEN AND
C       THE DEFAULT VALUES LISTED PREVIOUSLY ARE USED
        COMPILER STATIC
        CALL FDELETE ("PINSTORAGE.DA")
        CALL FOPEN(2,"FOR01.DA")
        READ FREE (2,ERR=3)XL,XLP,D,Q0,DT
        GO TO 4
3       TYPE"**************************************************"
        TYPE"****       ERROR ON READING FROM FILE FOR01.DA    ***"
        TYPE"***        EXECUTION PROCEDING WITH DEFAULT VALUES ***"
        TYPE"**************************************************"
        TYPE""
4       CALL FOPEN(1,"PINSTORAGE.DA")
        WRITE(1,99)D,DT,Q0,XL,XLP
99      FORMAT("NATURAL CIRCULATION IN SPENT FUEL",/,"TRIANGULAR ARRAY
     1 - TIGHT PACKING",//,"ROD DIAMETER IN INCHES = ",F6.3,/,"COOLANT
     2 INCREASE IN CHANNEL = ",F5.2,/,"POWER PER ROD IN REACTOR (BTU/HR)="
     3,F10.1,/,"ACTIVE LENGTH IN FEET = ",F6.2,/,"TOTAL ROD LENGTH IN
     4 FEET = ",F6.2,///)
        WRITE(1,200)
200     FORMAT(" TIN   PITCH   EQ DIA DELTA P   VEL.  ROD POWER  TOUT
     1  CLAD H  TCLAD   POW. FRAC.",/," (F)    (IN)    (IN)   (LB/SF)
     2  (FT/S)  (BTU/HR)    (F) (B/HSFF)  (F)     (NONE)    ",/
     3,"*****  ****  *****  ******  *******  ********  *****
     4  *****  *****  **********"/)
        DO 100 TIN=100.,200.,20.
        DO 90 P=D,D+.040,.010
        BETA=(2.+.02*(TIN-100.))*1.0E-04
        DP=BETA*RO*DT*XL/2.
        DE=8.*(0.433013*P*P-0.392699*D*D)/3.141593/D
        XMU=BETA*RO/(27.0+0.903*(TIN-100.))
        V=DP*G*DE*DE/XMU/XLP/144./32.
        AF=0.392699*D*DE/144.
        QR=2.*DT*RO*AF*V*3600.
        PF=QR/Q0
        X=QR*SR*XMU*8.*20736./(3.141593*D*BETA*RO*RO*G*DE*DE*DE*3600.)
        CT=SQRT(X)
        HC=12.*QR/(CT*3.141593*D*XL)
        TOUT=TIN+DT
        TCLAD=TIN+DT+CT
        WRITE(1,300)TIN,P,DE,DP,V,QR,TOUT,HC,TCLAD,PF
300     FORMAT(F5.1,F6.3,F7.4,F8.3,F9.4,E10.3,F7.1,F7.2,F7.2,E11.3)
90      CONTINUE
        WRITE(1,400)
400     FORMAT(/)
100     CONTINUE
        STOP
        END
```

The following table has been constructed from the decay heat power fraction data printouts of the previous nine pages and the higher decay heat curve of Figure 4. Suppression of film boiling criteria at the clad surface ($T_{sat} \leq 250°F$) is the basis of this comparison tabulation.

| | | | Required Cooling Times (YRS) | | | |
|---|---|---|---|---|---|---|
| $T_{in} \approx T_{pool}$ | $T_{out}$ | $T_{cladmax}$ | P = D = .422 | P = D + .01=.432 | P = D = .400 | P = D + .01=.410 |
| 200 | 230 | 249 | 4.00 | 1.26 | 5.71 | 1.48 |
| 180 | 220 | 245 | 2.90 | 0.89 | 3.20 | 1.03 |
| 160 | 210 | 241 | 2.40 | 0.68 | 2.74 | 0.80 |
| 140 | 200 | 238 | 2.00 | 0.59 | 2.51 | 0.74 |
| (1) 140 | 200 | 238 | 1.77 | 0.52 | 2.22 | 0.65 |

TABLE 1. Required cooling times for spent fuel rods stored in a tightly packed triangular array.

(1) Reduced burn-up conditions (16,000 EFPH, 22.000 MWD/MTU)

## 1.2 DISCUSSION AND CONCLUSIONS FOR THE NATURAL CIRCULATION ANALYSIS

The following conclusions are based on the analysis of the previous section and the required cooling times of TABLE 1. Natural circulation is assumed to be the prime means of heat removal in some of the tightly-packed triangular arrays of spent fuel rods. Suppression of film and local boiling in the water channels between rods is imposed as a criterium.

1.2.1 A 10 mil gap between rods reduces the required cooling by more than a factor of three. Uncertainties in the rod packing scheme may lead to such a gap since 18 rods placed side by side comprise a distance of 7.60 inches without gaps and 7.60 + 17(.010) = 7.77 inches with 10 mil gaps. The difference, 0.17 inch, is less than one-half rod diameter. However, since some areas in the array may contain no gaps, it may not be acceptable to assume a credit for this.

1.2.2 Imposing a condition of pool bulk temperature ≈140°F, the required cooling time will exceed two years but not three years. Storage time prior to consolidation for this condition (Case 4 in TABLE 1) is 2.0 years for fuel rods with 0.422 inch diameter and 2.51 years for fuel rods with 0.400 inch diameter. Neither a factor of 2 increase in the flow driving pressure nor a 33% decrease in fuel burn-up will significantly reduce these times. The first two or three cases in TABLE 1 constitute extreme conditions that typically are not met when a normal refueling or full core off-load discharge are first placed in the pool. In addition, these conditions may be somewhat artificial since the bulk pool is

already near boiling. Therefore, it will not be necessary to wait three years before the fuel rods are stored in the tightly-packed configuration.

1.2.3 The recommended cooling time, based on the assumptions, criteria, and analysis presented here, is 2.5 years. After this time, the spent fuel rods may be packed into a tightly-packed (triangular) array and adequately cooled.

## 2.0 REFERENCES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