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MONTICELLO
NUCLEAR POWER STATION

FIRST RELOAD
LICENSE SUBMITTAL

FEBRUARY 1973

PREPARED BY
GENERAL ELECTRIC COMPANY
NUCLEAR FUEL DEPARTMENT

RETURN TO REGULATORY CENTRAL FILES
ROOM 016

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I. INTRODUCTION

This document provides the technical basis of the license submittal for the first reload of Northern States Power/ Monticello Unit 1. Presented herein is a description of the new fuel and the results of the evaluation of the refueled core for the March 1973 outage.

The fuel available for the outage will be 28 Reload-1 fuel bundles, with an average bundle enrichment of 2.30 wt % U-235.

The objective of this outage is to assure the availability of the plant at high power for approximately an annual cycle while reducing the core off-gas to the lowest practical value.

II. SUMMARY

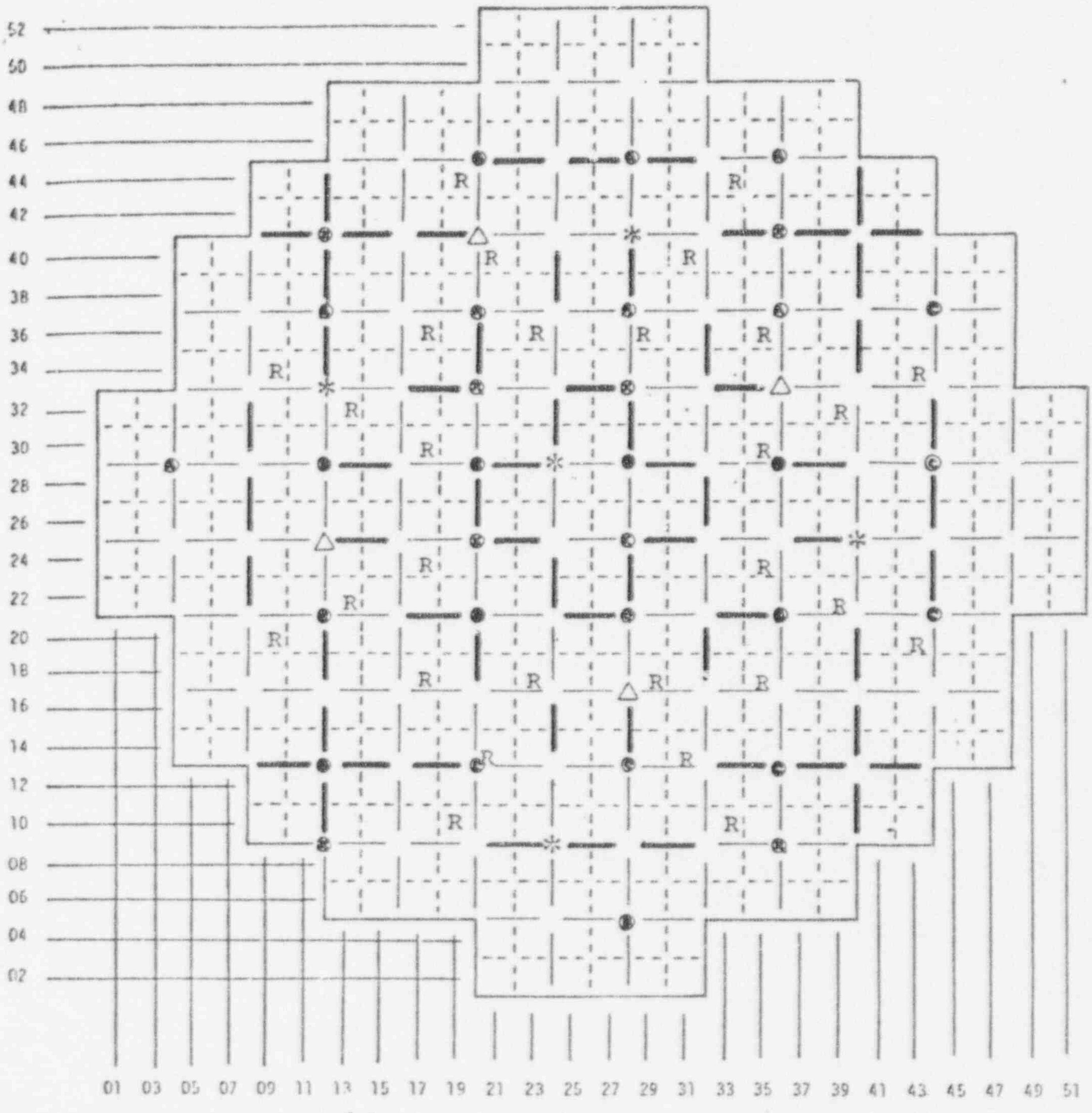
The design reference core configuration for this license submittal consists of 28 Reload-1 new fuel bundles with an average enrichment of 2.30 % U-235 and 456 initial core fuel bundles at a pre-outage core average exposure of approximately 6800 MWd/t. The reactivity control of the design reference core is augmented by leaving 64 of the original 216 temporary control curtains scattered throughout all but the peripheral area of the core in a symmetric and homogeneous pattern. The relative locations of the reload fuel and curtains in the reference core is shown in Figure II-1. The reload fuel bundle is of the same general mechanical configuration that the General Electric Company has been designing and manufacturing for the past ten to twelve years. The reload fuel uses uniformly dispersed gadolinium for reactivity control augmentation. The Reload-1 fuel also incorporates mechanical design changes recently implemented in initial core fuel of newer BWR facilities. Detailed descriptions of the designs are given in Section III of this document.

Cold shutdown calculations have been made on the full use of all Reload-1 bundles and a specified distribution of the 64 temporary control curtains. Ample cold shutdown margin throughout the cycle has been determined for the design reference case and also for the cases using 0 and 52 R1 bundles.

The reactor will be able to operate safely at 1670 MWt after the outage and satisfy all license requirements. The flexibility for fuel shuffling within specified constraints are included in this submittal.

The reload fuel is designed to operate within the same constraints on maximum linear heat generation rate (MLHGR) and minimum critical heat flux ratio (MCHFR) limits as the initial core fuel.

MONTICELLO



- LPRM Location (Letter indicates TIP machine)
- LPRM Location (Common location for all TIr machines)
- IRM Locat:ons
- Δ SRM Locations
- * Source Locations
- R Reload Bundles
- ▬ Remaining Curtains

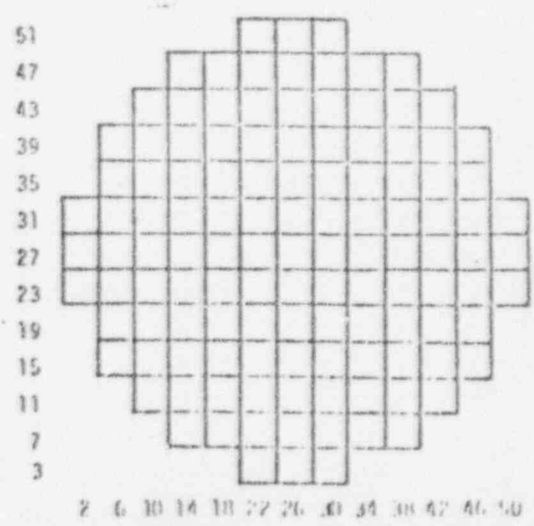


Figure II-1
Reference Design Core
Loading for Cycle 2

III. MECHANICAL DESIGN

A. FUEL DESCRIPTION

The R1 (Reload-1) fuel for the Monticello reactor is designed with the same envelope dimensions as the initial core fuel. It can, therefore, be inserted, without restriction, into all locations within the reactor core. Figure III-1 describes the general characteristics of the R1 fuel bundle and Table III-1 summarizes the significant characteristics and differences of the initial and R1 fuel. The basic lattice arrangement of 49 rods in a 7x7 array remains the same as the initial core fuel, with a centrally located spacer capture rod, and eight tie rods located symmetrically around the periphery of the fuel bundle. Figure III-2 shows the location of the enrichments within the fuel assembly as well as the location of the tie rods and spacer capture rod for the R1 fuel.

B. FUEL DESIGN

The R1 fuel is designed to meet the same stress intensity limits as the initial core fuel. The following table presents a summary of the basic stress intensity limits that are applied for Zircaloy-2 cladding:

Categories	Stress Intensity Limits in Terms of: Yield Strength (Sy)	Ultimate Tensile Strength (Su)
Primary Membrane Stress	2/3	1/2
Primary Membrane Plus Bending Stress Intensity	1	1/2 to 3/4
Primary Plus Secondary Stress Intensity	2	1.0 to 1.5

Design analyses have been performed which show that the stress intensity limits given in the table are not exceeded at continuous operation with linear heat generation rates up to the operating limit of 17.5 kW/ft.

Stresses due to external pressure, internal pressure, bending effects at end plugs, and thermal stresses have been considered. Tensile properties used in stress analyses are based on test data of irradiated fuel rod cladding for the applicable temperature.

A value of 1% of plastic strain of the Zircaloy fuel rod cladding has been defined as the limit below which fuel damage due to overstraining the fuel cladding is not expected to occur. Available data indicate that there is a small, but finite, probability that some cladding segments may have plastic elongation less than 1% at failure, but the distribution also indicates the 1% strain value to be the ~95% point in the total population. For fresh fuel, the calculated linear heat generation rate (LHGR) corresponding to 1% diametral plastic strain of the cladding is ~28 kW/ft.

Later in life the high exposure fuel has less nuclear capability, due to depletion of fissionable material, and will operate at correspondingly lower powers so that a wide margin is maintained between the operating LHGR and the LHGR calculated to cause 1% cladding diametral strain.

C. DESIGN CHANGES

The principal changes in the R1 fuel, as compared to the initial core fuel, are in the areas of supplementary

reactivity control, pellet geometry, and cladding heat treatment and dimensions. The R1 fuel will also use four enrichments as compared to three in the initial core fuel. The average enrichment of the R1 fuel is 2.30 w/o U-235 as compared to 2.25 w/o U-235 for the initial core fuel. Additional information on the design changes implemented in the R1 fuel, including the bases for these changes, can be found in the licensing topical report, NEDO-10505, "Experience with BWR Fuel Through September 1971," by H.E. Williamson and D.C. Ditmore, dated May 1972.

1. Supplementary Reactivity Control

Fuel rods containing small amounts of gadolinium oxide as supplementary reactivity control have been used in the design of reload fuel by General Electric since 1965. These fuel rods are the same design as the UO_2 rods with the exception that they contain small amounts of gadolinia with the UO_2. The mechanical design criteria applied are the same as for UO_2 rods. Performance of gadolinium containing fuel has been successfully demonstrated in over 300 assemblies in the Dresden-1 (50-10), Big Rock Point (50-155), and Humboldt (50-153) reactors. More recently, over 724 assemblies containing gadolinium were loaded in the Dresden-2 (50-237) reactor and the use of gadolinium in Quad Cities-1 (50-254) and Quad Cities-2 (50-265) reactors has been reviewed and approved by the AEC. The latest reloads in Nine Mile Point (50-220), 64 fuel bundles, and Oyster Creek (50-219), 156 fuel bundles, also include gadolinia in the fuel rods. The information in dockets 50-237, 50-254, and 50-265 is particularly

relevant because the gadolinium concentration in the R1 fuel is within the range described in these dockets. Additions of small amounts of gadolinia to UO_2 results in a reduction in the fuel thermal conductivity and melting temperature. However, at the most limiting steady-state operating condition which will be experienced by the fuel, the peak power gadolinia-urania rod will operate at a power no greater than 0.87 of the peak power UO_2 rods. Thus, if the peak UO_2 rod operates at a LHGR of 17.5 kW/ft, then the peak gadolinia-urania rod will have a LHGR no greater than 15.25 kW/ft. The LHGR calculated to cause 1% plastic diametral strain for fresh gadolinia-urania fuel is ∿25.0 kW/ft. The gadolinia is distributed uniformly over the axial length of the fuel column.

2. Pellet Geometry

Table III-1 presents a comparison of the pellet geometry for each type of fuel. The geometry of the pellet in the R1 reload fuel is the same design being used in the most recent initial core fuel designs, i.e., Browns Ferry-1, Peach Bottom-2, and Cooper Station.

In the most recent initial core fuel the pellet design is changed to reduce the pellet length-to-diameter ratio, and to chamfer the pellet ends. In addition, there will be no dished pellets, and the pellet diameter has been decreased.

The pellet length is changed to approximately 0.5 inch from the previous value of approximately

0.8 inch. The pellet diameter is reduced to 0.477 inch O.D. in order to maintain the pellet-to-cladding gap at the current value of 0.012 inch, while increasing the clad wall thickness by 0.005 inch.

The pellet design changes of reduced length-to-diameter ratio and chamfering are expected to reduce the local cladding strains associated with pellet distortion, but will not significantly affect the average diametral strains at pellet midplanes.

Data from GETR capsule irradiations, including density measurements on irradiated fuel pellets and average dimensional measurements, show no significant fuel irradiation swelling in the range of peak pellet exposures to be experienced in BWRs. All of the General Electric fuel rod data on UO_2 swelling indicates that the internal porosity in the sintered UO_2 pellet will be sufficient, without the dish, to accommodate fuel internal irradiation swelling for the range of exposures to be encountered in the BWR.

3. Cladding

A comparison of cladding dimensions is presented in Table III-1. The change in cladding thickness for the R1 fuel results from a change in the cladding heat treatment. This change is consistent with the design of the most recent initial core fuel, i.e., Browns Ferry-1, Peach Bottom-2, and Cooper Station.

The change in Zircaloy cladding heat treatment involves an increase in stress relief annealing temperature. All other aspects of the heat treatment remain unchanged. The cladding heat treatment temperature has been changed to the higher value to give less variability in mechanical properties.

This heat treatment results in an increase in ductility and an associated reduction in strength of the material. The cladding wall thickness is increased from 0.032 to 0.037 inches in order to assure that the design stress limits shown in Section B are met for the lower strength specially heat-treated cladding. The pellet-to-cladding gap will be maintained at the current value of 0.012 inch, despite the increase in the clad wall thickness, by decreasing the pellet diameter to 0.477 inch.

4. Other Changes

a. Hydrogen Getter

A proprietary hydrogen getter is introduced in the upper plenum of each fuel rod in the R1 fuel. The proprietary hydrogen getter is introduced as an added precaution against any process error which might result in leaving an extraneous hydrogen impurity in the fuel rod. As in previous designs, each fuel rod is also subjected to a high temperature vacuum outgas.

b. Thermal Wafers

A thermal wafer is being introduced between the bottom pellet and the lower end plug to reduce the operating temperature of the lower end plug. This change was introduced to assure satisfaction of cladding design stress limits in the weld region based on early data on cladding mechanical properties. More recent cladding property data and further analysis indicate that stress limits would not be exceeded even if the thermal wafer were not included.

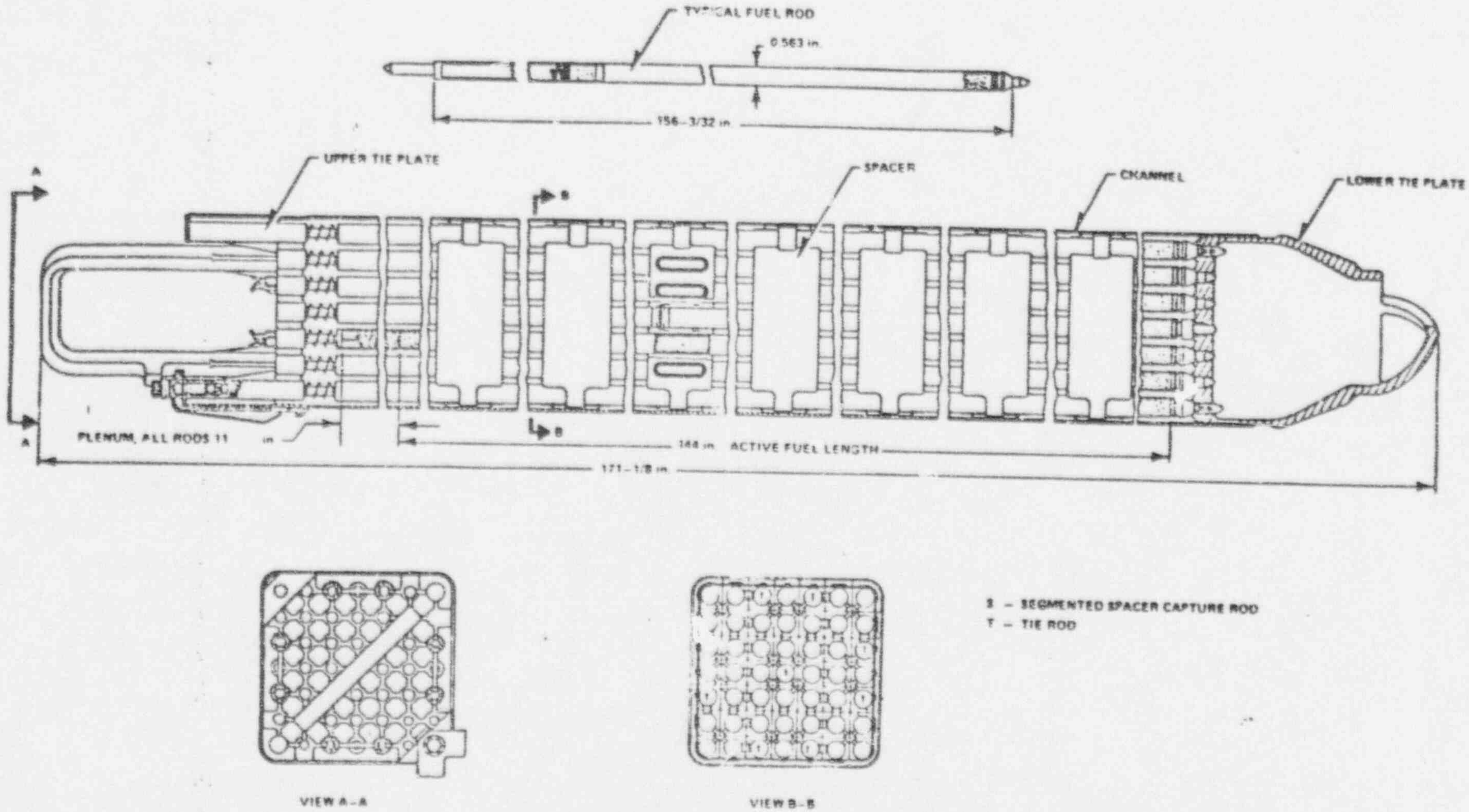


Figure III-1 MONTICELLO, RELOAD 1 FUEL ASSEMBLY

Figure III-2

Monticello R1 Fuel Lattice

wide-wide corner

4	3	T 3	2	T 2	2	3
3	2	1	1	1	1	2
T 3	1	G 1	1	1	1	T 1
2	1	1	S 1	1	G 1	1
T 2	1	1	1	1	1	T 1
2	1	1	G 1	1	1	1
3	2	T 1	1	T 1	1	2

ROD TYPE	ENRICHMENT w/o U-235	NUMBER OF RODS
1	2.56	32
2	1.94	10
3	1.69	6
4	1.33	1

S - Spacer Capture Rod
T - Tie Rods
G - Gadolinia Rods

TABLE III-1
INITIAL CORE AND RELOAD FUEL ASSEMBLY DESIGN SPECICATIONS

FUEL BUNDLE	INITIAL CORE FUEL	RELOAD 1 FUEL
Material	SINTERED UO_2 PELLETS	SINTERED UO_2 PELLETS
Initial Enrichment, w/o U^{235}		
High	2.95	2.56
Medium High	1.91	1.94
Medium Low	1.13	1.69
Low		1.33
Average for Bundle	2.25	2.30
Pellet Geometry		
Long Dished	27 rods	0 rods
Long Undished	22 rods	0 rods
Short Chamfered (undished)	0 rods	49 rods
Pellet Diameter, inches	0.487	0.477
Density, % of Theoretical		
Undished Rods	94.3	94.1
Dished Rods	90.8	N.A.
Melting Point, °F	5080	5080
CLADDING		
Material	Zr-2	Zr-2
Thickness, inches	0.032	0.037
Fuel Rod O.D., inches	0.563	0.563
FUEL RODS		
Active length, inches	144.0	144.0
Gas Plenum Length	11-1/4	11
Fill Gas	Helium	Helium
Getter	No	Yes

TABLE III-1
INITIAL CORE AND RELOAD FUEL ASSEMBLY DESIGN SPECIFICATIONS (Cont.)

	INITIAL CORE FUEL	RELOAD 1 FUEL
SPACERS		
Number per bundle	7	7
Material	Zr-4 with Inconel Springs	Zr-4 with Inconel Springs
FUEL BUNDLE		
Geometry	7x7	7x7
High Enrichment Rods	22	32
Medium High Enrichment Rods	19	10
Medium Low Enrichment Rods	8	6
Low Enrichment Rods		1
Poison Rods per Bundle	None	3
Rod Pitch, inches	0.738	0.738
Water to Fuel Volume Ratio	2.47	2.53
Heat Transfer Area, ft^2	86.5	86.5
CHANNELS		
Material	Zr-4	Zr-4
Outside Dimension, inches	5.438	5.438
Wall Thickness, inches	0.080	0.080
Channel Length, inches	162-1/8	162-1/8

IV. THERMAL-HYDRAULIC CHARACTERISTICS

The reload fuel is designed for operation within the operating limits as specified for the Monticello reactor, namely:

$$\text{MLHGR} \leq 17.5 \text{ kW/ft}$$

$$\text{MCHFR} \geq 1.9$$

The thermal-hydraulic characteristics of the reload fuel are the same as for the initial core fuel.

V. NUCLEAR CHARACTERISTICS OF THE FUEL AND CORE

A. CORE EFFECTIVE MULTIPLICATION, CONTROL SYSTEM WORTH AND REACTIVITY COEFFICIENTS

A tabulation of the typical nuclear characteristics of the pre- and post-outage cores is provided in Table V-1. Since the geometry and materials used in construction of the replacement fuel are nearly identical to those of the initial fuel, the nuclear characteristics of the reload fuel are sufficiently close to those of the initial fuel that temperature and void dependent characteristics of the reload core will not differ significantly from the values previously reported.

B. REACTOR SHUTDOWN

The refueled core fully meets criteria established for the initial core in that it may be maintained subcritical in the most reactive condition throughout the subsequent operating cycle with the most reactive control rod in its full-out position and all other rods fully inserted.

This core loading assumes removal of 152 of the 216 temporary control curtains and insertion of 28 R1 fuel bundles. A minimum shutdown margin of 0.020 Δk has been calculated for an assumed refueling at a core average exposure of 6800 MWd/t.

Because of the relatively large shutdown margin and relatively flat exposure distribution of the reference design core, considerable lattitude exists in the establishing of the actual core loading at the time of the outage. The analysis done on the reference design

core will remain valid, in the sense that all technical and safety limits will be met, with a pre-outage core average exposure as low as 6300 MWd/t, so long as the reference core-loading, curtain-pull pattern is adhered to. At any core-average exposure above 6800 MWd/t, substantial in-core shuffling of initial core fuel can be tolerated, so long as the reference design curtain-pull is maintained with any number of R1 bundles up to 52 in a scatter pattern and no initial-core peripheral bundles are moved more than three bundle rings toward the center of the core. At a pre-outage core exposure several hundred MWd/t above that of the reference design pre-outage core, the shutdown margin corresponding to use of the design reference curtain-pull, loading pattern will be great enough to allow the removal of as many as eight additional curtains from the core so long as the basic symmetry and homogeneity of the pattern is maintained, the fuel loading pattern is followed, and sufficient reanalysis of the modified core is done to ensure compliance with the basic technical and safety limits.

C. LIQUID POISON SYSTEM

The liquid poison system is designed to provide the capability to bring the reactor from full power (1670 MWt) to a cold xenon-free shutdown condition (keff < 0.97) assuming none of the control rods can be inserted. The requirements of this system are dependent primarily on the reactor power level and on the reactivity effects of voids and temperature between full-power and cold shutdown. The liquid poison system has been examined and has been found to be adequate since the reference power level of 1670 MWt has not changed and the core reactivity effects of voids and temperature

have not been significantly altered by introduction of reload fuel.

D. REACTIVITY OF FUEL IN STORAGE

There is no new safety implication with the spent fuel storage pool and new fuel storage rack. The basic criteria for storage of fuel in the Monticello new fuel vault or spent fuel pool are satisfied provided that the reactor fuel has a $k_\infty < 1.263$ at 65°C with no control rod and no removable control augmentation of any kind. The R1 fuel satisfies the above criteria and is criticality-safe for storage in the new fuel vault or spent fuel pool.

E. SINGLE CURTAIN EFFECT ON LOCAL PEAKING

Consideration has been given to the fact that a single-curtained fuel bundle has higher local power peaking than the same bundle in either the two-curtained or uncurtained configuration. All single-curtained initial fuel bundles will have average exposures above 5000 MWd/t. In the two-curtained or uncurtained configurations, these bundles have an infinite lattice local power peaking less than 1.24; in the single-curtained configuration, they have an infinite lattice local peaking less than 1.26. Since the bundle-average power of single curtained bundles is relatively lower than that of adjacent uncurtained bundles, no problems will be experienced in meeting limits on MLHGR and MCHFR. During the cycle after the outage, the plant will take due consideration of the change in local peaking of initial bundles and its effects on plant operations and the determination of in-core power distributions.

TABLE V-1

NUCLEAR CHARACTERISTICS OF CORE

Core Effective Multiplication and Control System Worth	Pre-Outage Core[1]	Reload Core[2]
(0% Voids, 72°F)		
Keff Uncontrolled	1.13	1.12
Δk Poison Curtains	0.04	0.01
Δk Control Rods	0.16	0.16
Keff Fully Controlled	0.93	0.95
Keff Strongest Rod Out	0.95	0.98
Reactivity Coefficients		
Steam Void Coefficents at 38% Voids; Δk/k/%void Range of values during operating cycle.	-10.7×10^{-4}	-11.3×10^{-4} -9.8×10^{-4}
Power Coefficient at 1670 MWt; 523 Btu/lb Inlet Enthalapy. Δk/k/Δp/p	- 0.054	- 0.055
Fuel Temperature Coefficient at 650°C, Δk/k/ΔF°.	-1.11×10^{-5}	-1.08×10^{-5} -1.17×10^{-5}

1. End of Cycle 1 Core Average Exposure, 6800 MWd/T; 216 curtains in core.

2. Beginning of Cycle 2 Core Average Exposure, 6371 MWd/T; 64 curtains in core.

VI. SAFETY ANALYSES

A. CORE SAFETY ANALYSES

Use of the Hench-Levy correlation to determine the safety limit and to establish margins from the normal operating points to the safety limit was established in previous licensing submittals. The same considerations, margins, damage limits, and operating limits described in detail before, apply to the R1 reloaded core. A further discussion of these controlling factors in the core safety analyses is presented below:

1. Fuel Damage Limits

Fuel damage from perforation of the cladding and a subsequent release of fission products can result from overheating or excessive strain of the cladding. The former is not expected to occur before MCHFR reaches 1.0 based on the Hench-Levy correlation and the latter is assumed to occur when MLHGR reaches 28 kW/ft. The mechanical design of the R1 reload fuel, as discussed in Section III, is to the same design criteria and bases as the initial core fuel and the same damage limits are applicable with the exception of the cladding strain limit on the gadolinia-equipped rods. (As discussed previously, however, power on these rods is always lower than on maximum powered standard rods in the same bundle.)

2. Operating Limits

The R1 reload bundles are designed to operate within the same operating limits as the initial fuel

in the same environment. That is, MCHFRs will be greater than 1.9 and MLHGRs will not be greater than 17.5 kW/ft. The limiting values of MCHFR and MLHGR for the R1 reload fuel during normal operation are the same as for the initial core fuel based on the similar design conditions established for the fuels.

3. Operating Margins

With the same damage limits and operating limits as the initial fuel, operating margins between the two limits for the R1 reload fuel will be the same. However, this is based on the minimum allowable situation. Actual R1 reload fuel operating conditions of MCHFR and MLHGR are expected to be well below the design limits, as has been the experience for initial core fuel. Thus, actual operating margins will continue to be greater than the minimum allowable values used in the analyses discussed below.

4. Abnormal Conditions

The minimum allowable operating margins described above are conservatively used in analyses of events such as abnormal operational transients and uncertainties concerning steady-state fuel operating conditions. Since these margins are the same with the R1 reload fuel the results of these analyses will not change with the insertion of R1 reload fuel except where dynamic changes are occurring on the reactor and its characteristics have been changed by the R1 reload fuel in such a way as to significantly affect the transient results. These

considerations involve the transient analyses; this will be covered separately below.

B. ACCIDENT ANALYSES

1. Main Steam Line Break Accident

The analysis of the main steam line break accident depends on the operating thermal-hydraulic parameters of the overall reactor, such as the pressure, and the overall factors affecting the consequences, such as primary coolant activity. Insertion of R1 reload fuel will not change any of these parameters so the previously reviewed results of this analysis will not change.

2. Refueling Accident

The analysis of the refueling accident depends on mechanical damage caused by a fuel bundle falling back onto the top of the core while it is being removed, which will not change with the use of the R1 reload fuel. The consequences depend on the fission product inventory in the fuel and various factors affecting the amount and kind of releases to the atmosphere. There will be no change in the total quantity of fission products contained in the R1 reload fuel since it will be operating at no higher power level, but there will be slight changes in the relative amounts of different constituents because of the presence of and the additional use of gadolinia in the fuel as supplementary reactivity control instead of boron-stainless steel curtains. The effects of these small differences will be inconsequential in terms of the

releases caused, and undetectable when the various reduction factors are applied to determine off-site consequences. Therefore, the previously reviewed results of this accident analysis will not change.

3. Control Rod Drop Accident

The postulated sequence of events involves an abnormally high worth rod becoming disconnected from its drive, being stuck in the full-in position, the drive being withdrawn, and the control rod falling out to the rod drive position. The analysis is done at various reactor operating states; the key reactivity feedback mechanism affecting the shutdown of the initial prompt power burst is Doppler. Final shutdown is achieved by scramming all but the dropping control rod. Given a particular set of conditions, rod drop excursions for exposed reactors will be less severe than initial cold clean cores since the Doppler reactivity feedback will be more negative. This is due to the fact the Pu-240, which has a large negative Doppler effect, builds up with exposure. However, total worth of the dropping rod also has a major effect on the accident results: this has not changed significantly from previous analyses.

The methods being used to evaluate the rod drop accident have been revised based on improved analytical techniques. These are reported in reference (1).

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(1) Topical report NEDO-10527, "Rod Drop Analysis for Large Boiling Water Reactors".

Reference (1) is a topical report on the subject applicable to Beginning-of-Life-Conditions for certain cores, similar to Monticello. This describes the effects of control rod worths on the accident results; however, exposed core conditions are to be covered in subsequent topical reports soon to be issued. Meanwhile, the rod worth minimizer backed up by operating procedures will serve to restrict rod worths to low values, well below those used in original rod drop accident analysis contained in the FSAR.

The rods containing the Gd_2O_3 will have a lower failure threshold as stated in Section III. However, as also noted in Section III, the operating power levels will be substantially lower than the maximum powered UO_2 rods. The net effect on the gadolinia-containing rods during the postulated accident is that they remain further away from the design basis limit (rapid energy transfer to the coolant) than the maximum powered UO_2 rods and, therefore, do not affect the results of the accident.

4. Loss-of-Coolant Accident

As described in Section III, the R1 reload fuel enrichment levels are slightly higher than those in the initial core bundles. Further, the R1 reload bundles incorporate a burnable gadolinia poison in three of the fuel rods in the bundle. Thus, even though the maximum design local peaking factor will not be increased, the distribution of local peaking factors and changes in peaking factors with exposure are different with the R1 reload bundles.

These are the principal differences affecting the results of the loss of coolant accident (LOCA), thereby requiring reanalysis [(2)]. Other small differences in the design of the bundle and operating characteristics were accounted for in the reanalysis but would have negligible effect on the outcome.

The same models, analytical techniques, and assumptions used in the previous analyses [(3)] in showing conformance to the AEC interim acceptance criteria were also used in this case. The results are shown in Figures VI-1 through VI-4. These show how the maximum peak clad temperature (PCT) for the design basis accident (DBA) varies with exposure and the distribution of cladding temperatures within the bundle during the course of the accident for the worst case.

PCTs for smaller break cases were shown to be less than those for the DBA in the previous submittals and the same relation would exist for this reload. Hence, calculations were not performed for smaller break cases. Calculations were also performed for the DBA, to determine the total amount of fuel surface cladding that reacts in a metal-water reaction (MWR) based on the PCTs that were obtained, and the results are also shown in Table VI-1.

For comparative purposes, the results of previous analyses referenced above [(3)] are also shown in the

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(2) See Ref. (3) below and NEDO-10329 and Suppl. 1 thereto, "Loss of Coolant Accident & Emergency Core Cooling Models for General Electric Boiling Water Reactors," for a more detailed explanation of the effect.

(3a) E.C. Ward (NSP) letter to P.A. Morris (USAEC), "ECCS Conformance to New AEC Adopted Interim Acceptance Criteria," Dated Sept. 21, 1971.

(3b) "Northern States Power Company Application for Conversion of DPR22 to Full Term," Dated June 15, 1972.

table. Thus, not only does conformance to the AEC criteria continue to be demonstrated, but calculated maximum PCTs are substantially lower than those previously calculated.

TABLE VI-1

	Max. PCT	Total MWR
AEC Criteria	2300°F	1%
Initial Core Fuel		
Two Curtains	2255°F	0.12%
One Curtain	2180°F	0.11%
R1 Reload Fuel		
No Curtains	2080°F	0.10%

C. TRANSIENT ANALYSES AND CORE DYNAMICS

A complete range of single failure caused events which are abnormal but reasonably expected during the life of the plant were analyzed as part of the original licensing of the plant. These were included in the FSAR and updated in the submittal which accounted for new information developed concerning end-of-cycle (EOC) conditions[4].

The R1 reload fuel has negligible effect on the results of the analyses of these events.

The transient analyses reviewed can be categorized as primary pressure increases, moderator temperature decreases, reactivity insertions, core coolant inventory decreases, core coolant flow increases, and core coolant flow decreases. The purpose of the

(4) L. O. Mayer (NSP) to A. Giambusso (USAEC), "Supplemental Report of a Change in the Transient Analysis as Described in the FSAR," dated February 13, 1973.

transient analyses was to show that fuel damage limits would not be reached in each of the events. These damage limits are the same as those described in Sections III and VI A.1 above, i.e., the 1% cladding strain and MCHFR = 1.0 limits. In nearly all cases, the MCHFR limit is the one most closely approached. Since the R1 reloaded core has the same fuel damage limits as the initial core, an assessment of effects on currently approved transient analyses will only depend on potential changes to the course of the transient themselves.

The updated transient analyses referenced above were performed using EOC conditions that were calculated based on available operating plant data and analytical techniques. Thus, conditions for the currently planned next fuel cycle were compared to that to determine whether any new detailed analyses or changes were required. The changes were found to be minor and no further analyses were required at this time. The original FSAR and updated transient analyses still adequately cover these cases. A summary of the reasons for this conclusion is contained below for each of the categories of events.

For those events resulting in a primary system pressure increase, such as a turbine trip without bypass, the most important variable parameters affecting the magnitude of the pressure increase are the void coefficient and negative reactivity inserted from the control rod scram. The previously referenced analyses showed that the existing EOC case was the worst due to the change in the scram reactivity curve. For the fuel cycle covered by this submittal the same scram reactivity curve was assumed, and the void coefficient

and doppler effect were negligibly different; therefore, the change in the results of any analysis would be too small to see.

Moderator temperature decreases from such events as a feedwater controller failure in the increasing direction are also affected primarily by the void coefficient and the scram reactivity change. The same discussion from the paragraph immediately above applies except that these transients are less severe because they are slower in nature and the resulting pressure and flux increases are considerably less.

A scram does not occur as a direct result of core coolant flow decrease transients, i.e., trips of recirculation pumps, one pump seizure and others in this category, so a void coefficient change would be the principal effect changing these transients. The change in void coefficient is insignificant and does not affect the results. Start-up tests where actual recirculation pump trips were conducted demonstrated that these transients are comparatively mild.

The reactivity insertion event is exemplified by an operator inadvertantly and continuously withdrawing a control rod until stopped by the rod block monitor at a particular flux level. The results will depend primarily on the capability of the flux monitors to detect the local change in the fuel around the control rod as it is withdrawn and to stop the control rod before MCHFR reaches 1.0. This capability is a function of the geometrical arrangement of the flux monitors and control rods, control rod worths, and flux chamber response characteristics. These will

not be sufficiently affected by the insertion of this small amount of reload fuel to significantly affect the conclusions from the previous analytical results.

D. LOADING ERRORS

The possibility of loading errors consisting of misplaced pellets in a fuel rod, misplaced rods in a fuel bundle and misplaced bundles in a core have been considered. The worst case loading error for the reference core configuration occurs when a reload bundle is rotated 180 degrees in a location near the center of the core.

Proper orientation of fuel assemblies in the reactor is readily verified by visual observation and is assured by verification procedures during core loading. Five separate visual indications of proper fuel assembly orientation exist:

1. The channel fastener assemblies, including the spring and guard used to maintain clearances between channels, are located at one corner of each fuel assembly adjacent to the center of the control rod.

2. The identification boss on the fuel assembly handle points toward the adjacent control rod.

3. The channel spacing buttons are adjacent to the control rod passage area.

4. The assembly identification numbers on the fuel assembly handles are all readable from the direction of the center of the cell.

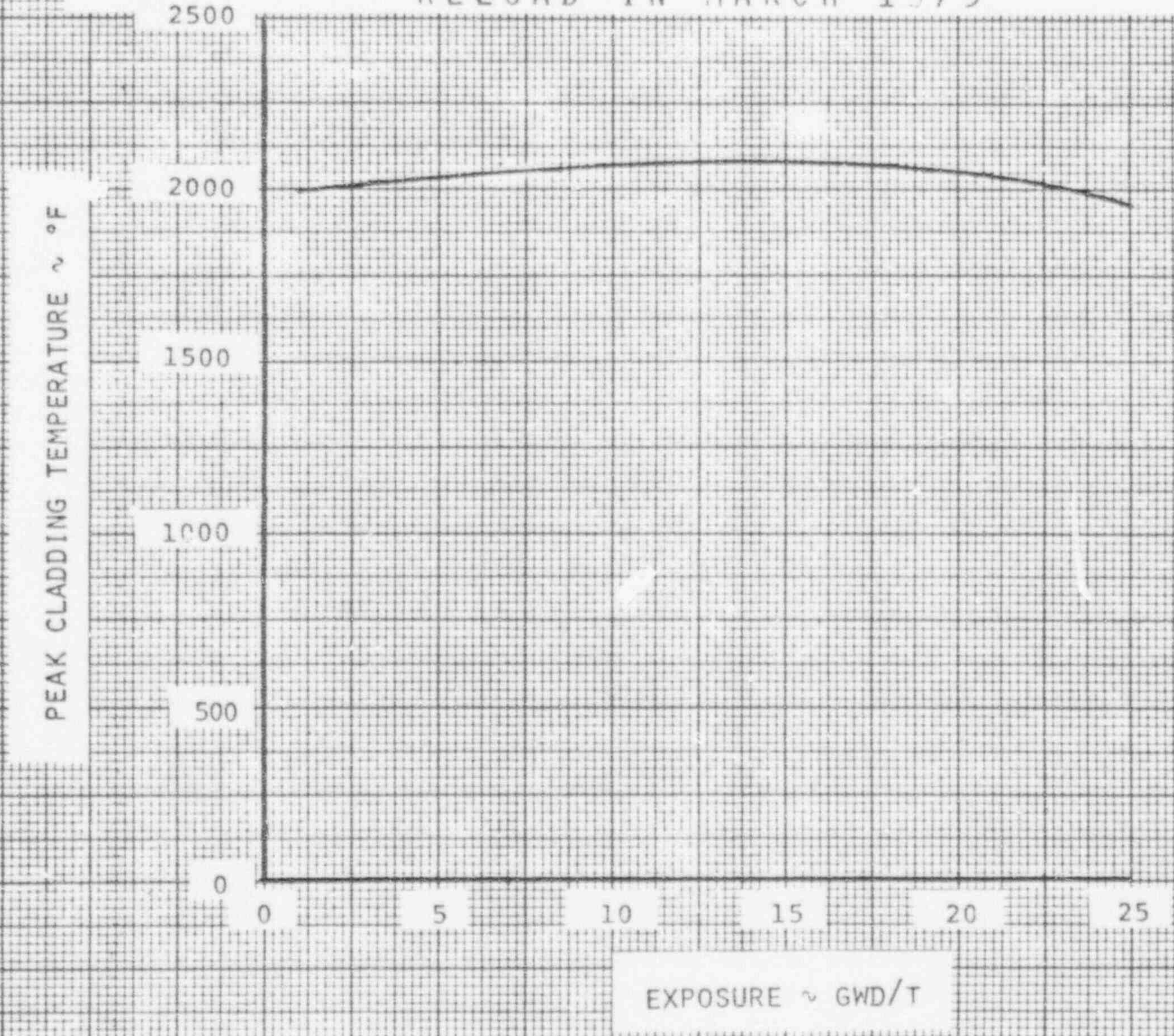
5. There is cell-to-cell replication.

Experience has demonstrated that these design features are clearly visible so that any misoriented fuel assembly would be readily distinguished during core loading verification.

If, however, through an error, fuel assembly were installed rotated 180° from the proper location which is the worst case rotational error, no fuel damage would be incurred during the subsequent power operation, even if the misoriented assembly were operating at the maximum permitted power.

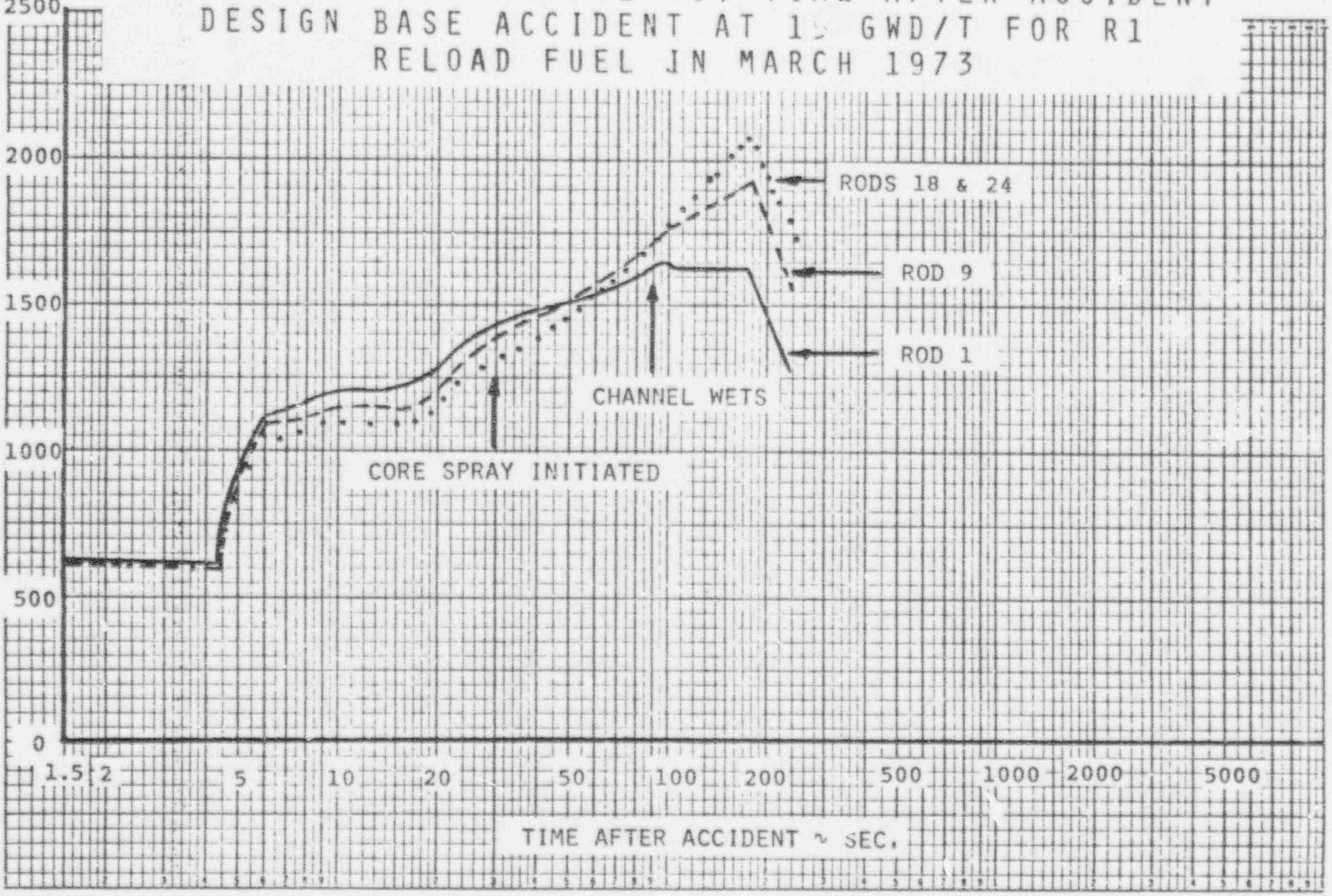
FIGURE VI-1

MONTICELLO
PEAK CLAD TEMPERATURE vs. EXPOSURE FOR
DESIGN BASE ACCIDENT WITH R1
RELOAD IN MARCH 1973



F URE VI-2

MONTICELLO
PEAK CLAD TEMPERATURE vs. TIME AFTER ACCIDENT
DESIGN BASE ACCIDENT AT 1 GWD/T FOR R1
RELOAD FUEL IN MARCH 1973



VI-13

FIGURE VI-3

MONTICELLO
PEAK CLAD TEMPERATURE vs. EXPOSURE FOR DESIGN BASE ACCIDENT WITH INITIAL FUEL IN BUNDLE WITH SINGLE CURTAIN-
MARCH 1973

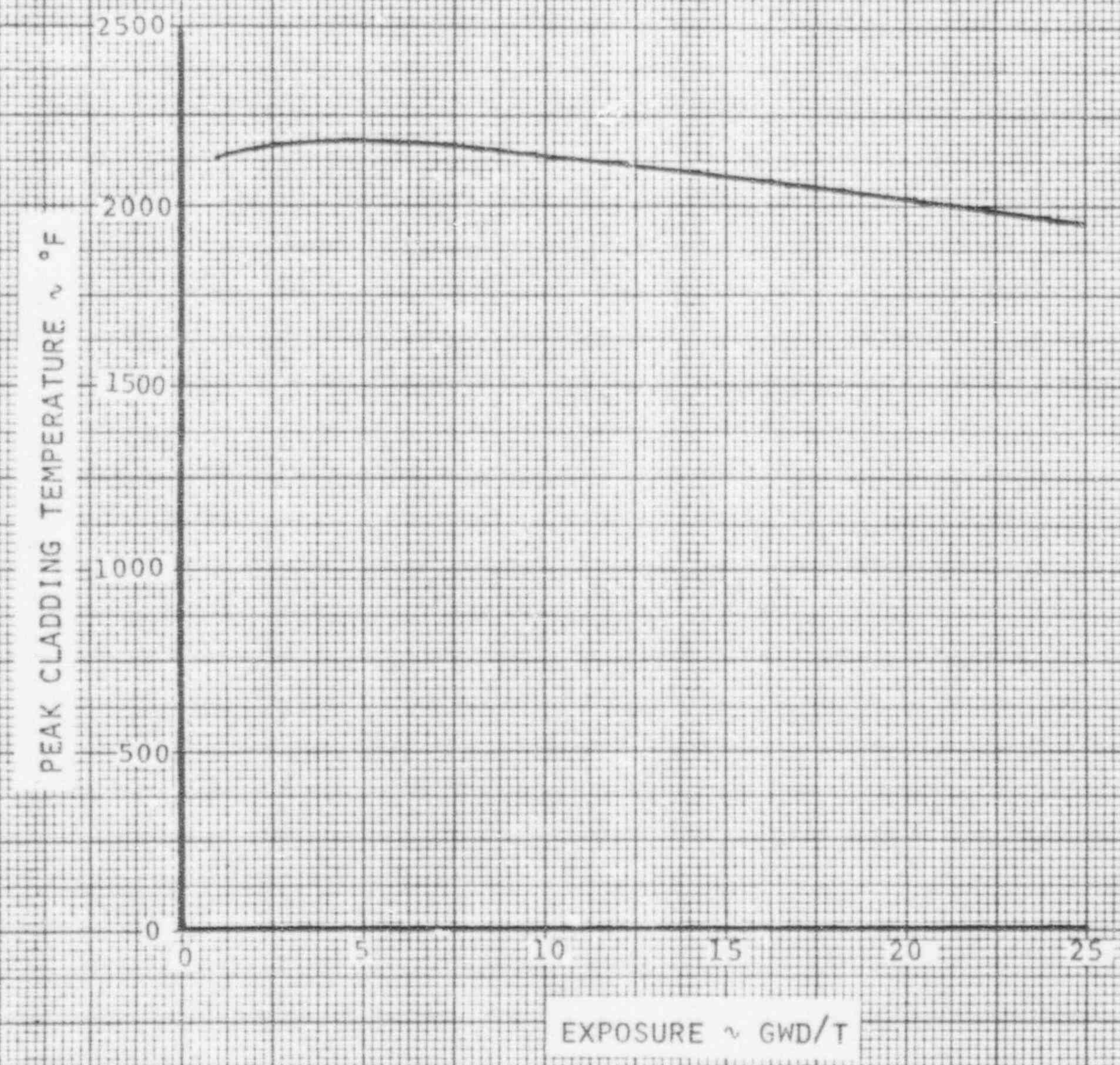
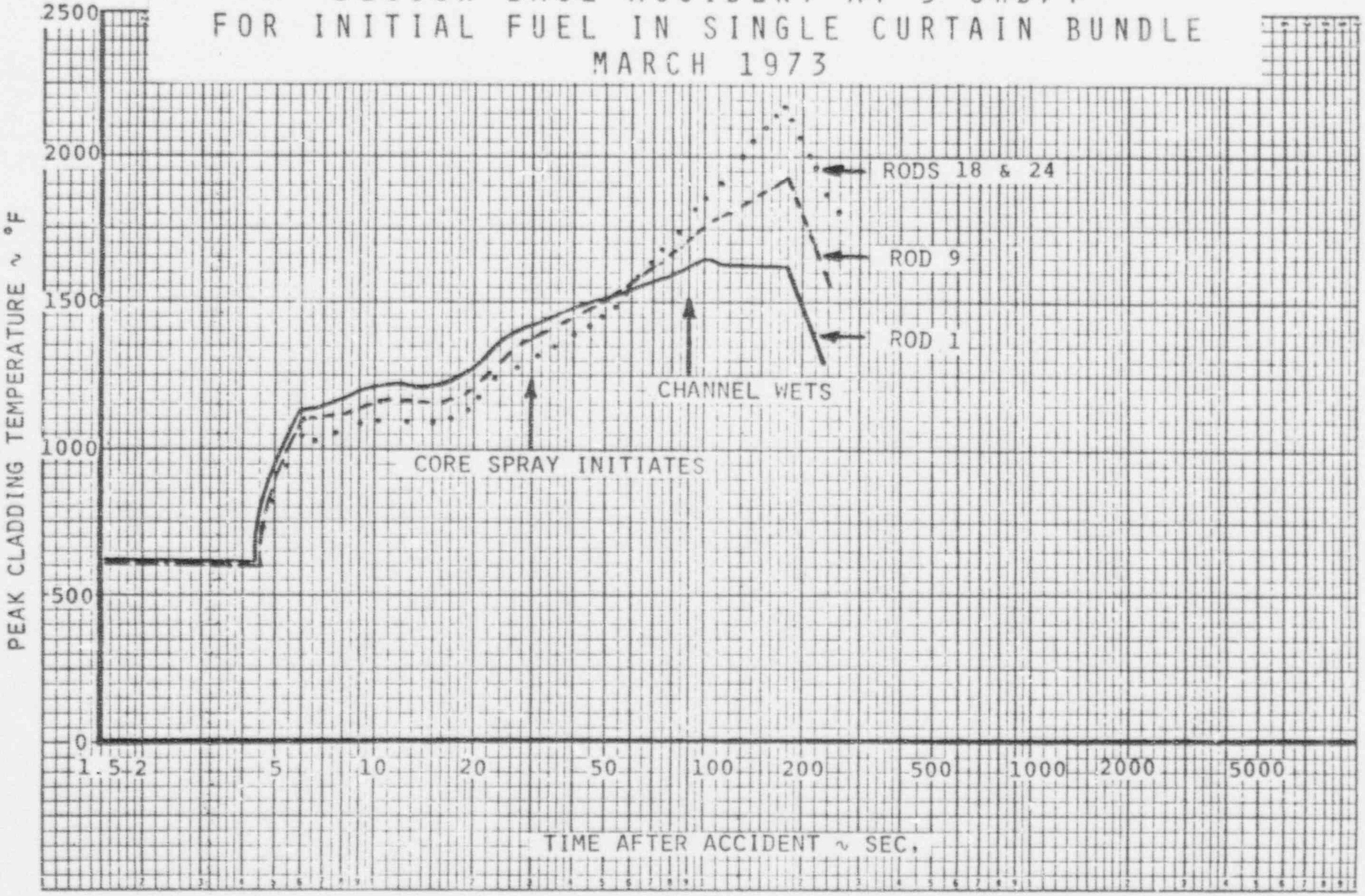


FIGURE VI-4

MONTICELLO
PEAK CLAD TEMPERATURE vs. TIME AFTER ACCIDENT
DESIGN BASE ACCIDENT AT 5 GWD/T
FOR INITIAL FUEL IN SINGLE CURTAIN BUNDLE
MARCH 1973



VII. RECONSTITUTION OF FUEL BUNDLES

A. GENERAL

Reconstitution of a number of failed fuel bundles is planned during the outage. Fuel bundle reconstitution is accomplished by replacing the leaker rods in a leaker bundle with sound fuel rods from another leaker bundle. A sound rod is defined as one which has been inspected by non-destructive testing techniques, both eddy current and ultrasonic, and results in a signal response which is acceptable when compared to results of known sound rods. Sound rods used for such replacement are again non-destructively tested immediately prior to insertion into the reconstituted bundle to confirm the soundness of the rod. Reconstituted bundles will not differ significantly in local power distribution, reactivity or exposure capability from the original fuel bundles. Controls are established in the nuclear criteria for reconstitution, Section C. A reconstituted bundle has the same inherent mechanical integrity as an as-built bundle. The same procedures and safeguards used for proper positioning of the original fuel are applicable for the loading of the reconstituted fuel in the core.

B. ACCOUNTABILITY

Procedures and check sheets are provided to control all rod movements to prevent any interchange of rod locations within the bundle during inspection and reconstitution. The mechanical design of the bundle upper tie plate precludes the possibility of a higher enrichment rod being inserted into a low enrichment rod location. This is accomplished by the end plug

shank diameter being larger for the higher enrichment rods which will not allow placement of these rods into the smaller upper tie plate h les which receive the lower enrichment rods. Some tie rods may have the same end plug size as another rod of different enrichment. In this case, rod interchange could be possible. However, it is impossible to assemble the tie plate with this condition existing. During actual reconstitution, each rod movement is specified in a detailed procedure incorporating checkoffs for each move. The actual movements are documented in a final report for record purposes which also include uranium weight interchanges for repaired bundles.

C. NUCLEAR REQUIREMENTS

The criteria used in development of bundle reconstitution rules are given below:

1. The maximum local power peaking will be less than the design value of 1.24.

2. The maximum reactivity change of any reconstituted bundle will be within $\pm$ 0.005 Δk of that existing for the original bundle.

3. Movement of rods will be restricted with regard to enrichment. Replacement rods must be of the same initial enrichment level as the replaced rod.

4. Replacement of failed rods and operation of the repaired fuel bundle will be done in a manner which will result in the peak clad temperature during a loss of coolant accident being less then 2300°F.

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