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ENCLOSURES:
(1) Attachment "A" - GE Proprietary Classification System.
(2) PROP INFO: NEDE-20179 - Monticello Segmented Test Rod Bundle(STR Bundle).

ACKNOWLEDGED

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December 14, 1973

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Mr. J. F. O'Leary, Director
Directorate of Licensing
Office of Regulation
U S Atomic Energy Commission
Washington, DC 20545

Dear Mr. O'Leary:

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

Submittal of NEDE - 20179 Monticello Segmented
Test Rod Bundle Submittal (Proprietary Information)

Attached are forty (40) copies of NEDE - 20179 which supplements the Monticello Second Reload Submittal from L. O. Mayer to J. F. O'Leary dated November 19, 1973. This report has been handled and classified proprietary by General Electric Company in accordance with the procedures and standards set forth in Attachment A to this letter. This particular information is available on a "need-to-know" basis to General Electric employees. It has not been previously released outside General Electric Company. The segmented test rods (STR) use advanced design and materials processes to improve performance and economics which may be applicable for future GE fuel designs. This technical information is of considerable commercial value and this would be lost with any public disclosure of this information. Also these design features have been obtained at considerable expense to General Electric and public disclosure of this information would allow competitors of General Electric access to this same information without incurring similar costs. It is therefore requested that this information be held in confidence and not be released for publication or otherwise disclosed in accordance with the provisions of 10CFR2.790.

We have concluded that the loading of the STR bundle described does not involve an unreviewed safety question. The Monticello Operations Committee and Safety Audit Committee have reviewed this information.

Our outage is scheduled to commence on February 23, 1974 with fuel loading shortly thereafter, and is expected to be completed within two months of that date. We shall appreciate establishment of a review schedule which will permit timely assembly and shipment of the STR bundle and resumption of plant operation as planned. The attached information is to be used to facilitate your review and approval of the insertion of one STR bundle (8x8) in the Monticello reactor. We would appreciate limited distribution consistent with proprietary information procedures.

8884

If there are any questions regarding the proprietary nature of this information, you may contact General Electric directly.

Yours very truly,

L. O. Mayer

L. O. Mayer, PE
Director of Nuclear Support Services

LOM/MHV/kn

cc: J G Keppler (without proprietary report)
G Charnoff
Minnesota Pollution Control Agency
Attn. Ken Dzugan (without proprietary report)

attachments

ATTACHMENT A

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NOVEMBER 1973
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MONTICELLO SEGMENTED TEST ROD BUNDLE SUBMITTAL

(SUPPLEMENT TO MONTICELLO SECOND RELOAD SUBMITTAL)

GENERAL ELECTRIC COMPANY

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Copy No. 1

STR BUNDLE

SUBMITTAL

MONTICELLO SEGMENTED TEST ROD BUNDLE

1.0 INTRODUCTION

A broad program for fuel performance improvement has been initiated by General Electric in concert with public power companies. Envisioned are a series of special fuel bundles placed in Boiling Water Reactors where the fuel will be incubated, that is, operated at low power for an extended period. This stage is a prelude to subsequent return to Vallecitos Nuclear Laboratory and reirradiation in GETR.* The GETR irradiations will assess the response of varying fuel concepts to power transients and cycling for direct comparison to the response of product line design. As an integral part of the cooperative program, Northern States Power Company and General Electric have planned to incubate a special bundle in the Monticello BWR.

The special fuel bundle, incorporating new design features aimed to improve performance, will be included with the Reload, R-2, fuel. Thirty-two fuel rods with refinements of geometry and materials will be segmented, that is, made in 36-inch lengths to enable subsequent exposure in the GETR. Incubation of the bundle in a large BWR is intended to be at a relatively low power level to fulfill test requirements and to minimize any possibility of interfering with operations. After exposures, ranging from 5000 to 25,000 MWD/T, selected rods are to be removed for testing at GETR. It is expected that data will be obtained indicating which design features are most beneficial.

This document provides the technical basis of the license submittal for the Monticello segmented test rod (STR) fuel bundle. Presented herein is a description of a fuel bundle which is similar to the second reload 8x8 design except for thirty-two central rods which will be segmented and incorporate special features. These will be enumerated, the prior irradiation performance of similar rods summarized, and the effects on operation outlined. This document is a supplement to the "Second Reload Submittal" for the Monticello Nuclear Generating Station.

Section 3, Mechanical Design, presents a full description of the bundle, in particular, the thirty-two segmented rods. In Sections 4 and 5 information with respect to nuclear and safety analyses is provided.

*General Electric Test Reactor

2.0 SUMMARY

Due to the reactor core location of the STR bundle in an interior position of an outer control cell, low power operation is assured. The calculated peak linear heat generation rate for the test bundle in this core location does not exceed 10.5 kW/ft at rated core output. The low power establishes that the STR bundle will always be less limiting than the peak power Reload (R-2) bundle in the core. The insertion of the STR bundle will not cause any safety-related limiting conditions in the core.

The Monticello Reload-2 (R-2) fuel, scheduled for February 1974 delivery, will employ 116 fuel assemblies. One of these will be a segmented test rod (STR) bundle. It will have thirty-two special interior rods each made up of four identical segments (See Figure 3-1). However, the assembly and rod exterior dimensions remain unchanged. Two of the four remaining unsegmented interior rods will be modified with a power depressor (H_fO_2 pellets) in the axial regions corresponding to non-fuel positions of the segmented rods. The remaining two interior rods, the water, spacer-capture rod and the corner, type 2 rod (2.14% enriched), will be unchanged.

3. MECHANICAL DESIGN

3.1 GENERAL DESIGN DESCRIPTION

The 8x8 fuel bundle contains 63 fueled rods and one spacer-capture water rod which are spaced and supported in a square (8x8) array by the upper and lower tie plates. The lower tie plate has a nosepiece which has the function of supporting the fuel assembly in the reactor. The upper tie plate has a handle for transferring the fuel bundle from one location to another. The identifying assembly number is engraved on the top of the handle, and a boss projects from one side of the handle to aid in assuring proper fuel assembly orientation. Both upper and lower tie plates are fabricated from Type-304 stainless steel castings.

The special features of the thirty-two segmented rods, 2.87% enriched, are listed in Table 3-1. The segmented bundle lattice is presented in Figure 3-1 and the segmented rod drawing is shown in Figure 3-2.

Each fuel rod and segment consists of high-density UO_2 fuel pellets stacked in a Zircaloy-2 cladding tube which is evacuated, backfilled with helium, and sealed by welding Zircaloy end plugs in each end. The fuel cladding thickness is adequate to be "free-standing," i.e., capable of withstanding external reactor pressure without collapsing onto the pellets within. Although most fission products are retained within the UO_2, a fraction of the gaseous products are released from the pellet and accumulate in a plenum at the top of each segment and rod. Sufficient plenum volume is provided to prevent excessive internal pressure from these fission gases or other gases liberated over the design life of the fuel. A plenum spring, or retainer, is provided in the plenum space to prevent movement of the fuel column during fuel shipping and handling.

Four types of rods are employed in the STR fuel bundle: tie rods, a water rod, standard rods, and segmented rods. The eight tie rods in each bundle have threaded end plugs which thread into the lower tie plate casting and extend through the upper tie plate casting. A stainless steel hexagonal nut and locking tab are installed on the upper end plug to hold the assembly together. These tie rods support the weight of the assembly only during fuel handling operations when the assembly hangs by the handle; during operation, the fuel rods are supported by the lower tie plate. One rod in each fuel bundle (see Figure 3-1) is a hollow water tube used to position seven Zircaloy-4 fuel rod spacers vertically in the bundle. The water rod is a hollow Zircaloy-2 rod equipped with a square bottom end plug to prevent rotation and assure proper location of the water rod within the fuel assembly. Several holes are drilled around the circumference of the water rod at each end to allow coolant water to flow through the rod. The spacers are equipped with Inconel-X springs and maintain rod-to-rod spacing. The remaining 55 rods in the bundle are the same length as the tie rods. The end plugs at top and bottom have pins which fit into anchor holes in the tie plates. An Inconel-X expansion spring located over the top end plug pin of each fuel rod keeps the fuel rods seated in the lower tie plate and allows them to expand axially and independently by sliding within the holes of the upper tie plate.

The fuel pellets consist of high-density ceramic uranium dioxide* manufactured by compacting and sintering uranium dioxide powder into cylindrical pellets with chamfered edges. The average UO_2 pellet immersion density is as listed in Table 3-1 for the segmented rods. For the remainder of the rods it is approximately 95% of theoretical density.

Four different U-235 enrichments are used in the fuel assembly to reduce the local power peaking factor (see Figure 3-1). Fuel element design and manufacturing procedures have been developed to prevent errors in enrichment location within a fuel assembly. The fuel rods are designed with characteristic mechanical end fittings, one for each enrichment. End fittings are designed so that it is not mechanically possible to completely put together a fuel assembly with any high enrichment rods in positions specified to receive a lower enrichment. As in the 7x7 assembly design, the 8x8 bundle incorporates the use of small amounts of gadolinium as a burnable poison in selected fuel rods. The gadolinia-urania fuel rods are designed with characteristic extended end plugs. These extended end plugs permit a positive, visual check on the location of each gadolinium-bearing rod after bundle assembly.

Most aspects of the STR 8x8 bundle design are similar to the R-2 8x8 design. Specifically, the upper and lower tie plates, the fuel rod spacers, the upper and lower end plugs, and other associated bundle hardware are the same. The 8x8 fuel assembly outline dimensions are the same as the current 7x7 dimensions. Table 3-2 presents a summary of 8x8 design dimensions. These also apply to the STR bundle with the exception of: (1) the average initial enrichment which is slightly lower as a result of removal of some of the 2.87% enriched pellets in the regions where segments are joined and (2) the active fuel length and plenum length which are changed for segmented rods (per drawing Figure 3-2), and (3) also, for some segments, pellet dimensions, and densities differ, as does cladding thickness as outlined in Table 3-1.

3.2 MECHANICAL DESIGN BASES

The STR bundle is designed to assure (in conjunction with the core nuclear characteristics, the core thermal and hydraulic characteristics, the plant equipment characteristics, and the capability of the nuclear instrumentation and reactor protection system) that fuel damage limits will not be exceeded during either planned operation or abnormal operational transients caused by any single equipment malfunction or single operator error.

Thermal and mechanical analyses of the segmented test rods have been performed using the same design methods applied in normal reload fuel design analyses. The maximum operating power and exposure combinations expected for the segmented test rods throughout their lifetime were used in these analyses. In particular, assurance that the segmented rods will operate satisfactorily throughout their lifetime has three bases: (1) favorable experience with similar rods (which is covered in the following section, Fuel Operating and Development Experience), (2) analyses have shown that cladding strain will remain within damage limits, and (3) operation of the bundle in a low power region of the core.

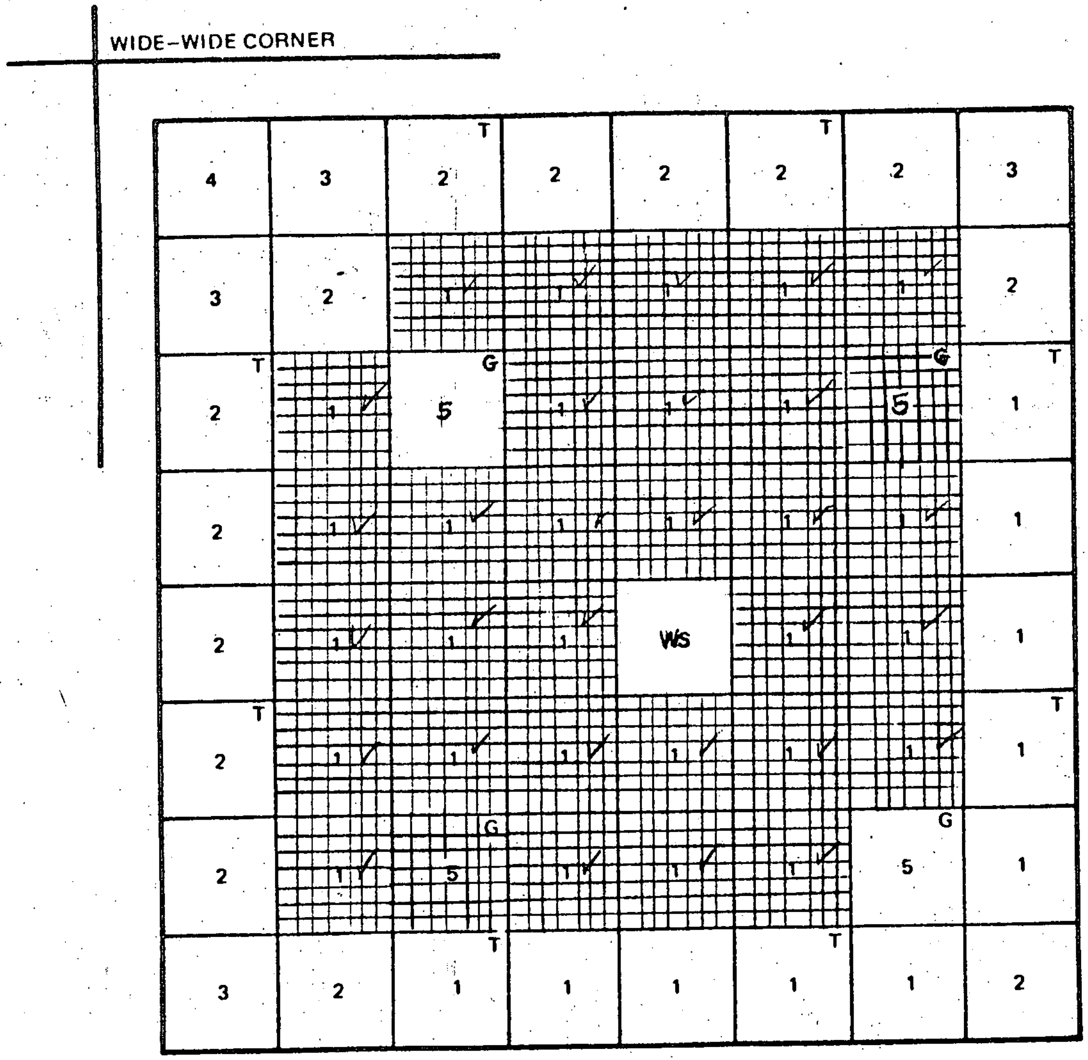
*in four rods, the pellets are Gd_2O_3-UO_2

TABLE 3-1

SEGMENTED ROD TEST MATRIX

	TYPE ROD	NO. RODS	PARAMETERS	
1.	Reference Standards	2	Solid Pellets:	(ρ = 95%, 0.009 Gap, 0.034 Wall)
2.	Gap, Wall, UO_2 Density Matrix	2	" "	(ρ = 95%, 0.009 Gap, 0.034 Wall)
3.	" " "	2	" "	(ρ = 95%, 0.007 Gap, 0.034 Wall)
4.	" " "	2	" "	(ρ = 95%, 0.004 Gap, 0.034 Wall)
5.	" " "	2	" "	(ρ = 98%, 0.007 Gap, 0.034 Wall)
6.	" " "	2	" "	(ρ = 95%, 0.009 Gap, 0.028 Wall)
7.	" " "	2	" "	(ρ = 95%, 0.007 Gap, 0.028 Wall)
8.	" " "	2	" "	(ρ = 98%, 0.007 Gap, 0.028 Wall)
9.	" " "	2	" "	(ρ = 95%, 0.007 Gap, 0.028 Wall) pre-pressurized
10.	Surface Finish Matrix	2	" "	(ρ = 95%, 0.009 Gap, 0.028 Wall)
11.	" "	4	" "	(ρ = 95%, 0.009 Gap, 0.034 Wall)
12.	Annular Pellet Matrix	4	Hollow Pellets	0.113 Inch I.D. Pellet (ρ = 95%, 0.007 Gap, 0.028 Wall)
13.	" " "	2	" "	0.113 Inch I.D. Pellet (ρ = 95%, 0.009 Gap, 0.034 Wall)
14.	Gadolinia Rods	2	Solid Pellets 4% Gd_2O_3	(ρ = 95%, 0.009 Gap, 0.034 Wall)
		32		

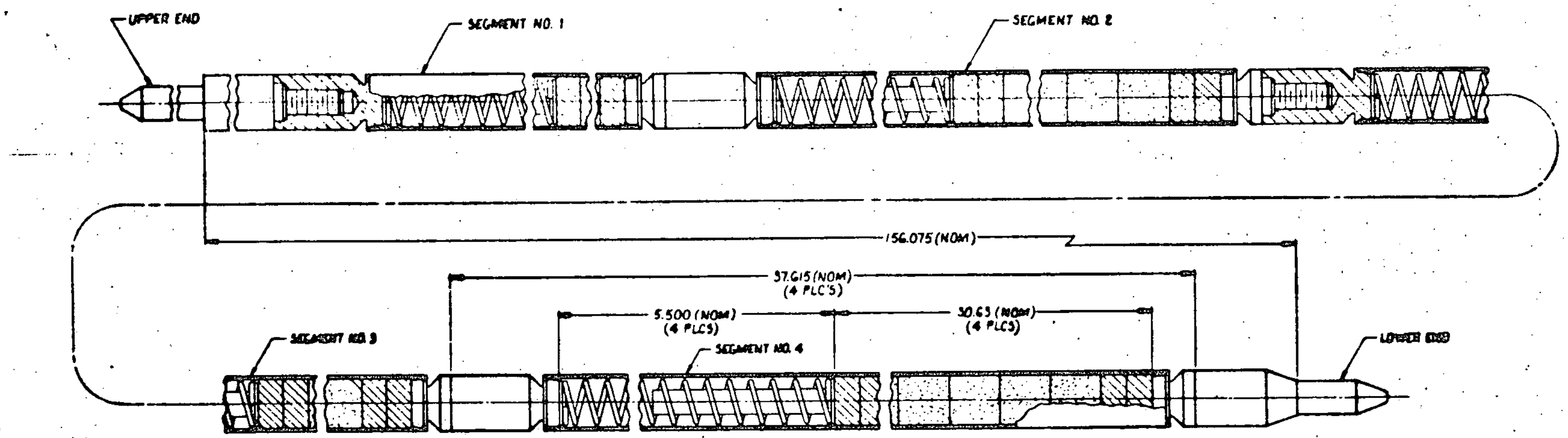
GENERAL ELECTRIC PROPRIETARY



ROD TYPE	ENRICHMENT wt % U–235	NUMBER OF RODS
1*	2.87	40
2	2.14	14
3	1.87	4
4	1.45	1
5*	2.87	4
WS	–	1

WS – SPACER CAPTURE WATER ROD
T – TIE RODS
G – GADOLINIUM RODS (4% Gd_2O_3 in segmented rods, 1.5% Gd_2O_3 in non-segmented)
* - SEGMENTED RODS ARE CROSS-HATCHED

Figure 3-1. Monticello R2 Reload Fuel Lattice FOR
STR BUNDLE



SEGMENTED TEST ROD

FIGURE 3-2

The fuel pellets consist of high-density ceramic uranium dioxide* manufactured by compacting and sintering uranium dioxide powder into cylindrical pellets with chamfered edges. The average UO_2 pellet immersion density is as listed in Table 3-1 for the segmented rods. For the remainder of the rods it is approximately 95% of theoretical density.

Four different U-235 enrichments are used in the fuel assembly to reduce the local power peaking factor (see Figure 3-1). Fuel element design and manufacturing procedures have been developed to prevent errors in enrichment location within a fuel assembly. The fuel rods are designed with characteristic mechanical end fittings, one for each enrichment. End fittings are designed so that it is not mechanically possible to completely put together a fuel assembly with any high enrichment rods in positions specified to receive a lower enrichment. As in the 7x7 assembly design, the 8x8 bundle incorporates the use of small amounts of gadolinum as a burnable poison in selected fuel rods. The gadolinia-urania fuel rods are designed with characteristic extended end plugs. These extended end plugs permit a positive, visual check on the location of each gadolinium-bearing rod after bundle assembly.

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3.2 MECHANICAL DESIGN BASES

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Thermal and mechanical analyses of the segmented test rods have been performed using the same design methods and cladding stress limits as applied in normal reload fuel design analyses. The maximum operating power and exposure combinations expected for the segmented test rods throughout their lifetime were used in these analyses. In particular, assurance that the segmented rods will operate satisfactorily throughout their lifetime has three bases: (1) favorable experience with similar rods (which is covered in the following section, Fuel Operating and Development Experience), (2) analyses have shown that cladding strain will remain within damage limits, and (3) operation of the bundle in a low power region of the core.

*in four rods, the pellets are $Gd_2O_3 \cdot UO_2$

TABLE 3-2

INITIAL CORE AND RELOAD FUEL ASSEMBLY DESIGN SPECIFICATIONS

	Initial Core Fuel	Reload Fuel R1	Reload Fuel R2
Fuel Assembly			
Geometry	7 x 7	7 x 7	8 x 8
High Enrichment Rods	22	32	44
Medium High Enrichment Rods	19	10	14
Medium Low Enrichment Rods	8	6	4
Low Enrichment Rods	0	1	1
Poison Rods	0	3	4
Water-Spacer Capture Rods	0	0	1
Rod Pitch (in.)	0.738	0.738	0.640
Water to Fuel Volume Ratio	2.47	2.53	2.60
Heat Transfer Area (ft^2)	86.5	86.5	97.6
Fuel Rod			
Active Fuel Length (in.)	144.0	144.0	144.0
Gas Plenum Length (in.)	11.25	11.0	11.24
Fill Gas	helium	helium	helium
Getter	no	yes	yes
Fuel			
Material	sintered UO_2	sintered UO_2	sintered UO_2
Initial Enrichment, wt/% U-235			
Average for Bundle	2.25	2.30	2.62
High	2.95	2.56	2.87
Medium High	1.91	1.94	2.14
Medium Low	1.13	1.69	1.87
Low	–	1.33	1.45
Pellet Diameter (in.)	0.487	0.477	0.416
Pellet Immersion Density (% TD)	95.0	95.0	95.0
Cladding			
Material	Zr-2	Zr-2	Zr-2
Thickness	0.032	0.037	0.034
Outside Diameter (in.)	0.563	0.563	0.493
Fuel Channel			
Material	Zr-4	Zr-4	Zr-4
Outside Dimension (in.)	5.438	5.438	5.438
Wall Thickness (in.)	0.080	0.080	0.080
Channel Length (in.)	162-1/8	162-1/8	162-1/8
Spacers			
Material	Zr-4 with Inconel Springs	Zr-4 with Inconel Springs	Zr-4 with Inconel Springs
Number per Bundle	7	7	7

3.3 FUEL OPERATING AND DEVELOPMENT EXPERIENCE

By conservative estimate the peak linear heat generation rate will be 10.5 kW/ft for the STR bundle, which will occur in the normal production fuel pins within the bundle. The peak linear heat generation rate on the segmented test rods themselves will be somewhat lower. This condition, for the various segmented rods, is well within the bounds of available production* and developmental fuel experience. Tables 3-3 and 3-4 show the ranges of General Electric development fuel irradiations which have been completed or are in progress. Various features have been tested which are present in the STR bundle. For example, it is seen that 4 mil gaps are represented and, in some cases, even smaller pellet to clad gaps are shown.

3.3.1 Pellet to Clad Gap

Additional evidence that segments with 4 mil clad to pellet gaps will operate satisfactorily is provided by the results of the GE "swelling" tests. These consisted of the successful irradiation of 40 six-inch long fuel pins with diametral gaps from 1.4 to 8.0 mils at power levels in the range 10-22 kW/ft to burn-ups in the range 12,000 to 91,000 MWd/T.

The experience of other countries and manufacturers provides additional backup. Summaries of Halden and Westinghouse irradiations are presented in Tables 3-5 and 3-6. AECL irradiations experience[5] with rods designed with 3 mil gaps, and wall thicknesses of .015 inches also are applicable. Thousands of these rods have been irradiated successfully in both development and commercial reactors with failure rates less than 1% on a bundle basis (lower if on a fuel rod basis).

3.3.2 Clad Thickness

GE experience with rods of design comparable to the segmented rods with 28 mil cladding is shown in Tables 3-3 and 3-4. Additional supporting evidence for the mechanical stability of the 28 mil wall case is the "creep" study of fuel tubing in the Big Rock Point Reactor of Consumers Power Company. Big Rock Point operates with a coolant pressure of 1350 psi, higher temperature, and higher neutron fluxes than Monticello. Four 0.034 and two 0.027 inch wall thickness rods (0.563 inch O.D.) have been exposed for several years without fuel (or any other internal support) and have exhibited very little change in diametral or longitudinal dimensions.

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*The production experience is presented in "The Second Reload License Submittal" for the Monticello Nuclear Power Station, Section 3.4.1 and Tables 3-3 and 3-4.

TABLE 3-3

GENERAL ELECTRIC DEVELOPMENTAL IRRADIATIONS ZIRCALOY-CLAD 95% TD UO_2 PELLET FUEL RODS

Name	Reactor	No. of Rods	Fuel Rod Dia. (in.)	Clad Wall Thickness (in.)	Pellet-to-Clad Gap (mils)	Peak Heat Flux ($Btu/h\text{-}ft^2$)	Peak LHGR (kW/ft)	Peak Exposure (MWd/Te)	Status
Dresden Prototype	VBWR	9	0.565	0.030	3.0-16.0	460,000	19.94	12,000	Completed
Fuel Cycle R & D[a]	VBWR	144	0.424	0.022	2.0-8.0	509,000	16.6	13,800	Completed
Dresden Prototypes	VBWR	52	0.565	0.023	5.0-8.0	407,000	17.64	10,000	Completed
High Performance	GETR	12	0.565	0.030	4.0-6.0	630,000	27.0	1,500	Completed[h]
UO_2[a]						1,126,000	49.0		
High Performance	GETR	2	0.565	0.030	4.0-11.0	1,355,000	58.0	14,000	Completed[e]
UO_2[b]									
SA-1[c]	Dresden 1	96	0.424	0.022	4.0-8.0	400,000	13.0	40,000	Completed
D-1,2,3[c]	Consumers	363	0.424	0.030	7.0	434,000	14.2	30,000	Completed
D-50[f]	Consumers	36	0.570	0.035	12.0	507,000	22.0	15,400	g,i
D-52,53	Consumers	58	0.700	0.040	13.0	525,000	27.0	4,600	i
GE-Halden	Halden	21	0.563	0.032-0.060	7.0-14.0	510,000	22.0	6,300	Continuing

a USAEC Contract AT(04-3) – 189 Project Agreement 11
b USAEC Contract AT(04-3) – 189 Project Agreement 17
c USAEC Contract AT(04-3) – 189 Project Agreement 41
d USAEC Contract AT(04-3) – 361
e Hollow Pellet
f USAEC Contract AT(04-3) – 189 Project Agreement 50
g Eight fuel rods failed during second operating cycle due to abnormal crud and scale deposition
h One rod failure @ 40 kW/ft
i Fuel assemblies presently out of reactor pending approval for reinsertion

GENERAL ELECTRIC PROPRIETARY

TABLE 3-4

GENERAL ELECTRIC DEVELOPMENTAL IRRADIATIONS ZIRCALOY-CLAD 95% TD UO_2 PELLET CAPSULES GENERAL ELECTRIC TEST REACTOR

Capsule	Number of Rods	Fuel Rod Dia. (in.)	Clad Wall Thickness (in.)	Pellet-to-Clad Gap (mils)	Peak Heat Flux (Btu/h-ft²)	Peak LHGR (kW/ft)	Peak Exposure (MWd/Te)	Status
A	3	0.425	0.024-0.032	1.4-10.2	750,000	24.5	88,000	Complete
	1	0.488	0.032	11.2	785,000	29.4	34,000	Complete
B	6	0.489	0.034	7.8-11.6	504,000	18.9	65,000	Complete
C	5	0.557	0.036	2.0-15.0	475,000	20.3	59,000	Complete
D	5	0.557	0.036	2.0-14.0	540,000	23.0	36,500	Complete
E	5	0.250	0.015	6.5	735,000	14.1	100,000	Complete
F	3	0.443	0.030	3.0-13.0	480,000	16.3	29,000	Complete

TABLE 3-5

HALDEN TESTS OF ZIRCALOY CLAD, PELLETED UO_2 FUEL
HEAVY WATER-BOILING WATER REACTOR
400 PSI COOLANT PRESSURE

Source	Assy. No.	No. of Rods	Pellet-to-Cladding Gap (mils)	Wall Thickness (in.)	O. D. (in.)	Length (in.)	Peak Heat Flux (Btu/hr-ft^2)	Peak kW/ft	Peak Exposure (MWd/T)	Comments
UKAEA	IFA-6	5	3.5-6.5	0.019±0.002	0.613	65.7	350,000	17.7	3,900	Failed--longitudinal crack, and end plug hydrided
UKAEA	IFA-7	5	3.5-6.5	0.019±0.002	0.613	65.7	350,000	17.7	4,100	
UKAEA	IFA-8	5*	3.5-6.5	0.019±0.002	0.613	65.7	350,000	17.7	4,400	Failed--top end plug
UKAEA	IFA-19	5*	3.5-6.5	0.019±0.002	0.613	65.7	350,000	17.7		
			6.5-11.5	0.019±0.002	0.613	65.7	350,000	17.7	3,200	Failed--top end plug
UKAEA	IFA-20	5*	3.5-6.5	0.019±0.002	0.613	65.7	350,000	17.7	5,100	
UKAEA	IFA-22	5	3.5-6.5	0.019	0.613	65.7	370,000	17.7		
				0.014	0.613	65.7	370,000	18.9	7,600	Failed--longitudinal and circumferential cracks
UKAEA	IFA-23	5	3.5	0.019	0.613	65.7	370,000	18.9	5,700	
				0.014	0.613	65.7	370,000	18.9		
UKAEA	IFA-25	5*	3.5	0.020	0.613	65.7	370,000	18.9	4,800	
				0.015	0.613	65.7	370,000	18.9		
UKAEA	IFA-29	5	3.5	0.019	0.613	65.7	495,000	24.8	15,000	
UKAEA	IFA-41	5	3.5	0.017	0.61	65.7	370,000	18.9	730	
				0.020		65.7				
Norway	IFA-4	14	4.3-10.0	0.018	0.54	31.4	160,000	8.0	14,000	
Sweden	IFA-10	6	6	0.025	0.547	67.5	314,000	14.1	3,800	
Sweden	IFA-11	5	2.0-6.0	0.021-0.025	0.55	67.5	365,000	16.4	9,000	
Sweden	IFA-21	5	2.0-7.0	0.019-0.025	0.55	67.5	300,000	15.2	11,100	Failed--longitudinal splits
Germany	IFA-26	7	5.5-12.0	0.024	0.596	43.5	435,000	21.5	6,000	
Italy	IFA-137	21		0.020	0.786		281,000	16.9	290	Failed
				0.039						

* Assemblies contain some (1 to 4) Zr-Nb clad rods.

TABLE 3-6

WESTINGHOUSE - CAROLINAS VIRGINIA TUBE REACTOR
ZIRCALOY CLADDING 2000 PSI HEAVY WATER

Source	Material	Assembly	Rod No	Cladding Thickness (in.)	O. D. (in.)	Pellet-to-Cladding Gaps (mils)	Peak Q/A (Btu/h - ft^2)	Peak kW/ft	Peak Exposure (MWd/T)	Comments
Westinghouse	Zircaloy-4	G-1	33,831	0.0222	0.4839	6.6	~475,000	~17.6	8,000	
			W13,831	0.0222	0.4839	6.6	~475,000	~17.6	8,000	
			44,731	0.0222	0.4839	7.8	570,000	21.2	9,600	
			44,732	0.0222	0.4839	7.8	570,000	21.2	9,600	Failed
			29,732	0.0222	0.4839	7.8	570,000	21.2	9,600	
			33,833	0.0222	0.4839	6.6	~475,000	~17.6	8,000	
		G-2	13,836	0.0222	0.4839	6.6	430,000	16.0	11,400	
			13,834	0.0222	0.4839	6.6	430,000	16.0	11,400	
			13,833	0.0222	0.4839	6.6	430,000	16.0	11,400	
			13,831	0.0222	0.4839	6.6	430,000	16.0	11,400	
			13,835	0.0222	0.4839	6.6	430,000	16.0	11,400	
			13,832	0.0222	0.4839	6.6	430,000	16.0	11,400	
		G-3	53,831	0.0222	0.4839	6.6	465,000	17.2	6,200	Failed
			13,837	0.0222	0.4839	6.6	465,000	17.2	6,200	
			64,761	0.0222	0.4839	7.8	560,000	20.6	7,400	
			64,762	0.0222	0.4839	7.8	560,000	20.6	7,400	
			24,734	0.0222	0.4839	7.8	560,000	20.6	7,400	
			53,862	0.0222	0.4839	6.6	465,000	17.2	6,200	
		G-4	83,832	0.016	0.4855	6.0	475,000	17.6	11,600	
			13,838	0.0222	0.4839	6.6	475,000	17.6	11,600	
			13,839	0.0222	0.4839	6.6	475,000	17.6	11,600	
			83,831	0.016	0.4855	6.0	475,000	17.6	11,600	Failed
		Standard Assemblies	1366 Rods	0.02575	0.4875	6.0	~370,000	~13.6	~18,000	1366 rods no failures
		[illegible]-2-31	1549	0.02575	0.4875	6.0	330,000	12.3	17,200	Failed
		[illegible]-1-15	1475	0.02575	0.4875	6.0	374,000	13.6	17,400	Failed

3.3.3 Hollow Pellets

The favorable irradiation experience with hollow (or annular) pellets is summarized in Table 3-7. Diametral expansion and circumferential ridge heights have been less than or equal to those experienced with solid pellet fuel. The success of the extensive program at Big Rock Point with annular PuO_2-UO_2 fuel firmly establishes the viability of the hollow pellet concept.

3.3.4 Fuel Density

Uranium dioxide fuel at 98% of theoretical density has been exposed successfully in GE sponsored irradiations at Halden as well as in development capsules in the GETR. Commercial fuel produced by the Canadians and the British has a nominal density of 97% and has demonstrated successful performance.

TABLE 3-7

HOLLOW PELLET FUEL IRRADIATION EXPERIENCE

(OPERATION BELOW MELTING)

SOURCE	REACTOR	EXPOSURE	PEAK POWER KW/FT	COMMENTS
G.E.	GETR - Capsule -5*	2 Hours	12 - 20	--
G.E.	GETR - Capsule -3*	30,000 MWD/T	14 - 20	One failure
G.E.	GETR Loop High Perf.	2,000 MWD/T	26	--
G.E.	Big Rock Point -PuO_2-UO_2 24 Rods -10 Meas*	25,000 MWD/T	11.6	Irradiation Still in Progress
G.E.	Big Rock Point PuO_2-UO_2 3 Bundles (204 Rods)	25,000 MWD/T	11.7	Examined in pool after one one year exposure and returned to service. No problems related to annular pellet rods.
G.E.	GETR - Capsule -3*	10,000 MWD/T	9	Irradiation Still in Progress
G.E.	Halden			
	a) IFA 214-5*	14,000 MWD/T	13 - 19	Irradiation Still in Progress
	b) IFA 238-1*	10,000 MWD/T	13 - 19	
	c) IFA 236-2*	6,000 MWD/T	12 - 16	
	d) IFA 408-1*	3,000 MWD/T	12 - 16	

*Number of Data Points

4.0 NUCLEAR CHARACTERISTICS

The impact of this bundle on neighboring bundles will be negligible since its nuclear characteristics have been maintained as essentially the same as a standard reload bundle. All nonsegmented rods are standard reload rods and all segmented rods in the bundle will incorporate the same enrichment (2.87 w/o U-235) as the rods which they replace in the standard reload bundle. The reactivity of the segmented rod bundle has been slightly reduced due to the lower inventory of UO_2 and the addition of a small amount of hafnium between the fuel rod segments.

In order to assure that no adverse power peaking will occur in the STR bundle during the bundle lifetime, 10-mil H_f cylinders will be placed in the plenum regions and 0.5 inches of H_fO_2 pellets will be loaded at both ends of every segment in the 32 segmented fuel rods. Hafnia pellets will also be loaded in the two unsegmented gadolinia-bearing fuel rods spanning the plenum end-plug regions of the segmented rods. The H_f content was chosen to conservatively simulate the uranium absorption at the beginning of the bundle life, with an allowance made for the depletion of the hafnium isotopes. The location of the two segmented gadolinia rods containing 4 w/o of Gd_2O_3 was chosen so that the effect of the increased gadolinia content on the local power peaking factor will be minimal. With these design features the maximum linear heat generation rate in the bundle will be 10.5 kW/ft throughout its lifetime. At approximately 5000 MWD/T it is planned to remove the two segmented gadolinia rods as a part of a pellet densification study.

5.0 SAFETY ANALYSIS

The segmented test rod bundle will have no effect on the results of the safety analyses previously reported for the Reload (R-2) fuel for Monticello. Included are the analyses of normal operation, abnormal operational transients, and highly unlikely accidents. There are two basic reasons for this.

First, there is but one bundle, and it has essentially the same nuclear characteristics as Reload (R-2) bundles. Thus, there would be no change to core characteristics which determine the outcome of safety analyses affecting the overall core. An example of this type is the transient analyses associated with primary system pressurization, such as on MSIV closure. Second, the location of the bundle in the edge of the core assures that lower peak LHGR's and higher MCHFR's will be maintained (regardless of any postulated reactivity imbalance) than are seen or are applied as operating limits in the center of the core, i.e. MCHFR's will be substantially greater than 1.9 and MLHGR's will be substantially less than 13.4 KW/FT for this STR bundle. Thus, margins between the maximum operating point and applicable damage limits for the STR bundle will be the same or greater than center loaded bundles. An example of the safety evaluations conducted follows for the control rod drop accident.

The worth of the control blade in the control cell containing the segmented rod bundle is less than the delay neutron fraction since it is located on the periphery of the core and neutron leakage effects will reduce its reactivity worth. Therefore, any postulated drop of the control blade immediately adjacent to this bundle would not result in a super-prompt-critical transient and hence no fuel rod stored energy which would approach the fuel failure threshold. A postulated drop of a control blade one or two control cells away (toward the central region of the core) may result in a super-prompt-critical transient which is no different than for a regular fuel bundle. This could result in some fuel segments of the STR bundle exceeding the energy of the fuel cladding perforation threshold of 170 cal/gm. However, due to the segmented construction of these fuel rods, the resulting amount of fission gas released due to a cladding perforation would be significantly less than that released due to the perforation of a full length standard fuel rod.

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[3]Williamson, H. E., and Ditmore, D. C., "Experience with BWR Fuel Through September 1971," May 1972 (NEDO-10505).

[4]Ditmore, D. C., and Elkins, R. B., "Densification Considerations in BWR Fuel Design and Performance," December 1972 (NEDM-10735).

[5]AECL-4366, "Improved Performance for UO_2 Fuel," January 1973, J. A. L. Robertson.

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