SAXTON NUCLEAR EXPERIMENTAL CORPORATION

DOCKET NO. 50-146
LICENSE DPR-4

Change Request #20

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1. Applicant hereby submits Change Request No. 20 in compliance with paragraph 3B of License DPR-4 for change of the Technical Specifications to be authorized by the Commission as provided in 10 CFR 50.59.

SAXTON NUCLEAR EXPERIMENTAL CORPORATION

By /s/ R. E. Neidig
President

ATTEST:

/s/ R. E. Sypher
Secretary

(S E A L)

Sworn and subscribed to before me this 7th day of May, 1965

(S E A L)

/s/ Martin A. Kohr
Notary Public
Muhlenberg Township, Berks County
My Commission Expires Feb. 4, 1966

Docket No. 50-146
DPR-4

Technical Specification
Change Request No. 20

1) Description of Change

In Supplement No. 1 to Technical Specifications, page 2, change Item F.2. to read:

> Uranium oxide (UO_2) enriched to 5.7% of U-235 shall be used in the fuel assemblies, except that the test fuel assemblies listed below having enrichments and other characteristics as described may be inserted in the reactor. In test fuel assemblies the fuel rods as described may be replaced with regular fuel rods, that is, enriched to nominal 5.7% U-235 and constructed as described in Technical Specification F.3.

Test Assembly No. 1

One 61 rod assembly containing rods of the numbers and types listed in the following table:

No. of Rods	Cladding	Clad Thickness[1]	Pellet Diameter	Enrichment	Peak Power
4[4]	304 SS	80.5 mils[11]	0.294 in.	0.71 w/o	3.1 kw/ft
4[5]	304 SS	80.5	0.294	0.29 w/o	2.2
3	304 SS	15	0.357	(2)	16
3	304 SS	15	0.357	(2)	16
3	304 SS	15	0.357	(2)	16
3	304 SS	15	0.357	(2)	16
3	16-20 SS	15	0.357	(2)	16
3	348 SS	15	0.357	(2)	16
3	304 SS[9]	15	0.357	5.69 w/o	13.5
3	304 SS	15	0.357	5.69 w/o[3]	13.5
3	Zr-4[6]	23.7	0.337	5.69 w/o	12.0
5	Zr-4[6]	23.7	0.337	6.1 w/o	13.5
3	Zr-2[6]	23.7	0.337	(2)	14
3	Zr-2[6] (Ni free)	23.7	0.337	(2)	14
3	Zr-4[6]	23.7	0.337	(2)	14
3	Zr-4[10]	23.7	0.337	(2)	14
3	Zr-4[8]	23.7	0.337	(2)	14
3	Zr-4[7]	23.7	0.337	(2)	14
3	Zr-4[6]	23.7	0.337	7.3 w/o	16

Notes for Table

(1) All rods are free standing 0.391 in. O.D. nominal.
(2) First 14 pellets 5.69 w/o
next 5 pellets 6.81 w/o
next 12 pellets 6.45 w/o
next 5 pellets 6.81 w/o
next 13 pellets 5.69 w/o
(3) Contains approximately 100 ppm boron as zirconium diboride
(4) RCC element with perforated guide tube
(5) RCC element with solid guide tube
(6) Autoclave pre-oxide on O.D.
(7) Autoclave pre-oxide on O.D. and I.D.
(8) Furnace pre-oxide on O.D.
(9) Compartmented rod, 3 sections
(10) As pickled, no pre-oxide treatment
(11) RCC rod O.D. is 0.461 in. nominal.

Test Assemblies ii and iii

	Test Fuel Assembly No. ii 9-Rod Subassembly	Test Fuel Assembly No. iii 9-Rod Subassembly
First 14 pellets	5.69%	5.69%
Next 2 pellets	9.19%	7.30%
Next 3 pellets	8.57%	6.81%
Next 12 pellets	8.13%	6.46%
Next 3 pellets	8.57%	6.81%
Next 2 pellets	9.19%	7.30%
Next 14 pellets	5.69%	5.69%

NOTE: The 9-rod subassembly in the first column shall not be used at reactor power levels greater than 20 MWt.

Test Fuel Assembly No. iv

One 9-rod subassembly shall have four corner rods clad with Zircaloy-4 having a nominal thickness of 23.7 mils and shall contain uranium oxide (UO_2) enriched to 6.1% U-235. The other five rods shall be clad with Type 304 stainless steel having a nominal thickness of 9.5 mils and shall contain uranium oxide (UO_2) enriched to 5.7% U-235.

Test Fuel Assembly No. v

One 9-rod subassembly shall have four corner rods clad with Zircaloy-4 having a nominal thickness of 23.7 mils and shall contain uranium oxide (UO_2) enriched to 6.1% U-235. The other five rods shall be clad with Type 304 stainless steel having a nominal thickness [illegible] mils and shall contain uranium oxide (UO_2) having the same enric[illegible] Test Fuel Assembly No. iii.

Test Fuel Assembly No. vi

One four-rod subassembly shall have rods clad with Type 304 stainless steel having a nominal thickness of 23.5 mils and shall contain uranium dioxide (UO_2) fuel pellets uniformly enriched to 8.3 w/o U-235. One of these rods may contain up to 100 ppm boron initially as zirconium diboride. Two of these rods may be replaced with 23.5 mil thick Type 304 stainless steel clad rods containing vibration compacted uranium dioxide powder enriched to 6.0 w/o U-235. These latter two rods shall initially contain approximately 500 ppm of boron as boron carbide (B_4C).

Test Fuel Assembly No. vii

One 9-rod subassembly shall have four corner rods and the center rod clad with Zircaloy-4 having a nominal thickness of 23.7 mils and shall contain uranium oxide (UO_2) uniformly enriched to 7.3%. Two of the other rods shall be clad with Type 304 stainless steel having a nominal thickness of 15 mils and shall contain uranium oxide (UO_2) uniformly enriched to 5.7% U-235. One other rod shall be clad with Type 304 stainless steel having a nominal thickness of 16.1 mils, shall contain uranium oxide (UO_2) having a content of 0.29% U-235, and shall be concentrically located within a solid stainless steel guide tube. The remaining rod shall be clad with Type 304 stainless steel having a nominal thickness of 16.1 mils, shall contain uranium oxide (UO_2) having a content of 0.71% U-235 and shall be concentrically located within a perforated stainless steel guide tube.

Test Fuel Assembly No. viii

One 9-rod subassembly shall have three corner rods clad with Zircaloy-4 having a nominal thickness of 23.7 mils and shall contain vibrationally compacted uranium dioxide (UO_2) enriched to 7.2% U-235 and compacted to 86 $\pm$ 2% theoretical density. The fourth corner rod and the central rod shall be clad with Type 304 stainless having a nominal thickness of 15 mils and shall contain vibrationally compacted uranium dioxide (UO_2) enriched to 7.2% U-235 and compacted to 85 $\pm$ 2% theoretical density. Three of the remaining rods shall be clad with Zircaloy-4 and shall contain uranium dioxide (UO_2) pellets 0.337 inches in diameter which are enriched to 6.1% U-235. One of these rods shall have a previous irradiation exposure of ~7500 megawatt days per metric ton (MWD/MT) and shall contain a 15 mil diameter hole machined through the clad. The second of these rods shall have a previous irradiation exposure of ~7500 MWD/MT. The third of these rods shall have a 15 mil diameter hole machined through the clad. The final rod shall be clad with sensitized Type 304 stainless steel and shall contain uranium dioxide (UO_2) pellets enriched to 5.69% U-235 and the ten central pellets shall have 20 mil chamfers on both ends.

Following a period of irradiation, the two defected, Zircaloy-4 clad rods may be replaced by similar, defected, unirradiated Zircaloy-4 clad rods.

Test Fuel Assembly No. ix

One 9 x 9 fuel assembly shall contain 51 rods clad with Type 304 stainless steel of 15 mils thickness and shall contain uranium dioxide (UO_2) fuel pellets of 5.69% U-235 enrichment. This assembly is made by removing the central 21 rods from a normal 9 x 9 fuel assembly. The space left by removal of the central 21 rods shall be filled by a plug consisting of a stainless steel tube 0.125 inches thick and 2.75 inches in diameter welded to perforated stainless steel end plugs. The end plugs shall be designed so that flow through the plug will experience the same enthalpy rise that is experienced by flow through a normal fuel assembly. The plug shall contain three concentrically mounted stainless steel pipes 0.125 in. thick and of 2.125, 1.50 and 0.75 in. diameters, respectively. Horizontal restraint for the plug shall be provided by the grids of the fuel assembly. Vertical support for the plug shall be provided by a 1.5 in. diameter stainless steel pipe extension of the reactor head port flange. When Change Request No. 16 has been approved by the Atomic Energy Commission, this plug may be removed and replaced by a fueled supercritical loop pressure tube assembly.

Test Fuel Assembly No. x

One 9-rod subassembly shall have eight rods clad with Zircaloy-4 having a nominal thickness of 23.3 mils. The fuel shall be mixed natural uranium-plutonium dioxide enriched to 6.6 w/o PuO_2. Four of the rods shall contain vibration compacted fuel and the remaining four shall contain sintered ceramic pellet fuel. The ninth rod position will be a flux thimble for use with in-core instrumentation.

Test Fuel Assembly No. xi

One 9-rod subassembly shall have the center rod and the four corner rods clad with Type 304 stainless steel having a nominal thickness of 15 mils and shall contain uranium dioxide (UO_2) uniformly enriched to 5.7% U-235. Two of the remaining rods shall be clad with Zircaloy-4 having a nominal thickness of 23 mils and shall contain uranium dioxide (UO_2) uniformly enriched to 6.45 w/o U-235. The third rod shall be clad with Zircaloy-4 having a nominal thickness of 23 mils and shall contain uranium dioxide (UO_2) uniformly enriched to 6.1 w/o U-235. This rod shall have a previous irradiation exposure in excess of 13,000 MWD/MTU. The last rod shall be clad with Zircaloy-4 having a nominal thickness of 23 mils and containing uranium dioxide (UO_2) uniformly enriched to 7.3% w/o U-235 and shall have a previous irradiation exposure in excess of 3000 MWD/MTU. The test subassembly may be inserted into the core when Change Request No. 18 is approved by the Atomic Energy Commission.

Test Capsules

Test capsules containing non-fuel material may be inserted in any of the eleven dummy fuel locations adjacent to the reactor core region or in any of the eight irradiation sample tubes on the periphery of the core provided that:

1) Prior to irradiation, the design of the test capsule has been evaluated by the SNEC Safety Committee and found acceptable with regard to physical, thermal and hydraulic performance and effect on core reactivity, neutron flux and reactivity coefficients.

2) No foreseeable failure of a test capsule could result in damage to any core component or in any manner alter the ability of the control system to function.

In Supplement No. 1 to the Technical Specifications, page 3, change N.4.e.(4).(b):

(b) 4 rod spiked subassembly, Btu/hr ft^2 577,600

In Supplement No. 1 to Technical Specifications, page 4, change N.4.e.(8).(b):

(b) In 4 rod spiked subassembly, kw/ft 26.5

2) Purpose of Change

The purpose of this change is to permit movement of the 4-rod subassembly into the central location and to permit replacement of two rods of the 4-rod subassembly with new rods containing vibration compacted loose oxide fuel and a burnable poison (boron carbide).

The scope of the second experiment involves the insertion of the unmodified test fuel assembly vi into the N-1 central location for irradiation at high specific power levels. During the irradiation, measurements will be made to correlate reactor power level and subassembly peak-specific power. Following the first period of irradiation, test fuel assembly vi will be removed and two of the fuel rods replaced by two new fuel rods which are of the same general mechanical design but are of lower enrichment and will contain a burnable poison (boron carbide) intimately mixed with vibration compacted, loose powder oxide fuel.

The objectives of this experiment are to provide information on the behavior of loose oxide fuels containing B_4C when operated at high specific power ratings. In addition, the previously irradiated pelletized rods will be subjected to higher burnup.

The present experimental fuel subassembly vi was inserted into a peripheral location in the Saxton reactor and was described in Addendum 1 dated December 20, 1962 to the Safeguards Report for the Saxton Phase 1 Research and Development Program. A detailed description of the mechanical design and thermal and hydraulic parameter for this subassembly in the peripheral location are presented in the Addendum.

3) Safety Considerations

Details of the mechanical design features of the four rod subassembly are presented in Addendum 1 to the Safeguards Report for Phase 1 of the Saxton Five-Year Research and Development Program, dated December 20, 1962. The principal design features of the old and new rods are presented in Table 2.

TABLE 2
FUEL ROD CHARACTERISTICS
4-ROD SUBASSEMBLY vi

	Pelletized Fuel	Loose Oxide Fuel*
Number of Rods	2	2
Active Fuel Length, in.	33.2	33.2
Rod Outside Diameter, in.	0.5995	0.5995
Clad Thickness, in.	0.0235	0.0235
Pellet Diameter, in.	0.5435	-
Cold Pellet-to-Clad-Gap, in.	0.009 (diametral)	-
Pitch, in.	0.746	0.746
Enrichment, w/o U-235 (Uniform throughout rod)	8.3	6.0

The loose oxide fuel rod cladding is designed and fabricated according to the same specifications that were used for the pelletized rods. The cladding is Type 304 stainless steel and is designed to be free standing, and at the maximum reactor pressure and temperature deformation will not exceed the elastic limit.

* Contains approximately 500 ppm of boron as B_4C dispersed in the loose oxide fuel. Density of loose oxide fuel is 87 ± 2% theoretical density.

The principal thermal and hydraulic parameters calculated for the operation of the 4-rod subassembly in the peripheral location and in the central location are listed in Table 3. They are compared with the data for a spiked 3x3 in the central location.

TABLE 3
THERMAL AND HYDRAULIC DATA
4-ROD SUBASSEMBLY v1

	Peripheral Location	Central Location	
	2x2 Subassembly	2x2 Subassembly	Spiked 9-Rod Subassembly
T_{inlet}, °F	520	520	520
Core Power, MWt	23.5	20.0	23.5
Mode of Control	Chemical Shim	Chemical Shim	Chemical Shim
Maximum Thermal Output, kw/ft	20.0	26.5*	16.0
Maximum Surface Heat Flux, Btu/hr-ft^2	444,000	577,600	533,000
Hot Channel Factors			
F_q	2.76	1.72	3.31
$F_{\Delta H}$	1.82	1.22	2.72
Minimum DNB Ratios by W-2 Correlation			
100% Power - 2000 psia	q"-DNBR = 2.81	q"-DNBR = 2.59	q"-DNPR = 2.27
120% Power - 1800 psia	q"-DNBR = 2.08	q"=DNBR = 1.86	q"-DNBR = 1.65

The potential for burnout of the 4-rod subassembly in the center position is less than that for the spiked 9-rod subassembly previously approved for operation in the central location. Table 3 lists the minimum DNB ratios for the normal operation of the 4-rod subassembly as 2.59 calculated by the W-2 correlation as compared with 2.27 for the spiked 9-rod subassembly. The overpower-underpressure DNB ratios for the 4-rod and 9-rod subassemblies are 1.86 and 1.65, respectively.

The highest nominal specific power of a pelletized rod in the 2x2 subassembly with the Saxton reactor operating at 20 MWt is calculated to be 24.2 Kw/ft. A calculational uncertainty of 10% is applied to make the maximum specific power 26.5 Kw/ft. These power levels are based on the following hot channel factors: $F_q = 1.72$, $F_{\Delta H} = 1.22$. The lower enrichment, vipac fuel rods will operate at a nominal maximum of 19.2 Kw/ft and at a peak of 21.1 with the 10% calculational uncertainty factor added and based on the same hot channel factors.

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* Maximum is calculated for 8.3 w/o pelletized rods only. Maximum for new 6.0 w/o loose oxide rods is 21.1 kw/ft.

Limited center melting of the pelletized fuel is expected if the rods are operated at 26.5 Kw/ft. About 2% of the hot spot cross section will be molten under these conditions. No center melting is expected in the vipac rods at 21.1 Kw/ft because of their lower power level and because of the flux depression caused by the B_4C in the fuel. The maximum calculated center temperature for the vipac fuel at 21.1 Kw/ft is 2400 °C which is well below the melting point of 2800°C. Center melting in vipac fuel would not be expected to occur unless a linear power density of 24 Kw/ft is reached.

Experimental data from G. E.(1), Hanford(2) and Westinghouse(3) work indicate that there are no serious problems involved in operating fuels up to 30 Kw/ft with center melting. Hanford programs have irradiated low enrichment PuO_2-UO_2 fuel pellets up to 38 Kw/ft. G. E. programs in the GETR using solid fuel pellets have experienced peak surface heat fluxes of 1×10^6 Btu/hr-ft^2 (this corresponded to an average rod power of 30 Kw/ft) without failure. Westinghouse programs in the Plum Brook reactor have run successfully at more than 30 kw/ft.

The maximum overpower transient in Saxton might cause the power to go to 120% of the nominal before scram would occur. This considers errors in setpoints and instrument deadband. The peak power in the 2x2 would then be about 31 Kw/ft and about 15% of the hot spot cross sectional area would be in a molten condition. The solid portion of the fuel will keep the molten fuel away from the clad and prevent any interaction between the two. The TREAT tests have shown that even in fairly rapid transients which involve only limited center melting that the molten fuel does not pass through the solid fuel and contact the cladding. These tests have also shown that in all but the most rapid transients, the temperature profile across the fuel remains parabolic. Only the very shortest period (~m sec) transients could provide the required rapid energy input into the fuel to cause the outer layer of fuel to melt and react with the clad.

Since the mechanisms required to create rapid reactivity transients are not present in the Saxton reactor, operation of the 2x2 with limited center melting is considered a safe experiment and that gross failure of the 2x2 due to the presence of center melting is not credible.

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(1) Deidenbaum, B., "High Performance UO_2 Program, Quarterly Progress Reports," GEAP-3771-1 to GEAP-3771-15.

(2) Roake, W. E., "Irradiation Alteration of Uranium Dioxide," HW-73072.

(3) Westinghouse data unpublished. To be presented at the June, 1965 ANS Meeting.

In addition to the safety margin calculated for the 4-rod subassembly, the flux wire and pitot tube thermocouple probe will provide measurements of the actual operating conditions to assure that the specific power limit of 26.5 Kw/ft will not be exceeded. The data will allow correlation of the reactor power level and the maximum specific power of the subassembly. The reactor may be operated at power levels above the limit of 20 MWt as shown in Table 3 if the measurements indicate a lower specific power than calculated in Table 3. Modification of the 4-rod assembly to permit insertion of the new rods requires the removal of the flux wire thimble from the assembly. The presence of the new, lower enrichment rods will not significantly alter the peak specific power in the old rods so that the reactor may operate with the modified 4-rod assembly up to the power level determined as limiting in the first irradiation period without exceeding the 26.5 kw/ft limit.

A study was performed using the Westinghouse W-2 DNB correlation to demonstrate that the 2x2 subassembly, when operated in the central location at specific power densities up to 26.5 kw/ft, will not be damaged under accident conditions. The accident analysis for the 2x2 assembly when operated in a peripheral location at calculated power densities of 20 kw/ft covered the two cases of loss of flow accident and the boron removal accident. Because the loss of flow accident produced a lower DNB ratio in the 2x2 than does the boron removal accident, the boron removal accident will not be repeated for this analysis.

The loss of flow accident was assumed to occur due to a loss of pump power and the flow coastdown curve of the main coolant pump is given in Figure I. The reactor and 2x2 parameters that were assumed for the accident are presented in Table 4. Reactor scram is assumed to start 1.5 seconds after the beginning of the accident and is completed at 2.4 seconds. The delay in scram initiation and completion is due to:

1) Flow decay to the low flow scram set point and error in the set point - 0.7 sec.
2) Instrumentation delay - 0.2 sec.
3) Rod motion in a region of small effectiveness - 0.6 sec.
4) Time for rods to complete motion in region of effectiveness - 0.9 sec.

The results of the loss of flow analysis are presented in Figure II and compared with the results for the previous analysis presented in Addendum 1. For the 26.5 kw/ft operation, the minimum DNB ratio is a Q-DNBR of 1.59 that occurs at 2.7 seconds after the start of the accident. The Q-DNBR was evaluated assuming a reactor coolant pressure of 1950 psia which is conservative as the Q-DNBR decreases with decreasing pressure.

In the event of a loss of flow accident with the 2x2 in the central position, an analysis would be made at the time of the accident based on the measured core and flow coastdown characteristics to ascertain if damage could have occurred to the 2x2. If this analysis indicates the possibility of damage, the 2x2 may be examined before resuming operation with the 2x2.

TABLE 4
4-ROD SUBASSEMBLY v1
ACCIDENT ANALYSIS PARAMETERS

Steady state inlet mass velocity, lb/ft^2 hr	1.157×10^6
Maximum heat flux, Btu/ft^2 hr	5.776×10^5
Average heat flux, Btu/ft^2 hr	3.36×10^5
Equivalent diameter, based on wetted perimeter, ft	0.0352
Equivalent diameter, based on heat transfer perimeter, ft	0.0685
Head loss coefficients, based on flow area in fuel region	
Bottom end plate	1.13
Spacers, total of 4	3.20
Top end plate	0.48
Steady state core pressure drop, psi	2.48

4) Health and Safety

It is our opinion that the health and safety of the public will not b endangered by this change.

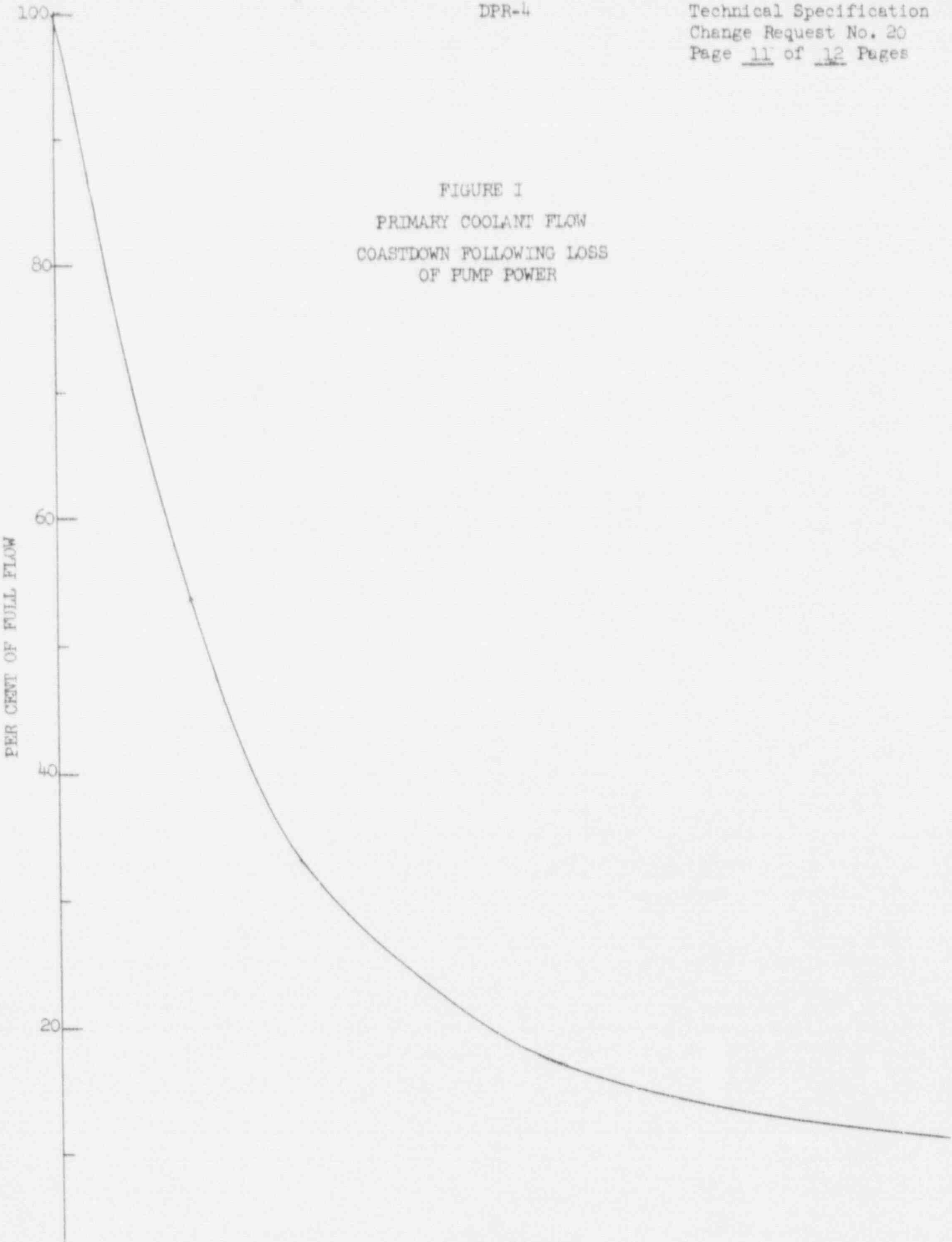
FIGURE I
PRIMARY COOLANT FLOW
COASTDOWN FOLLOWING LOSS
OF PUMP POWER
PER CENT OF FULL FLOW
100
80
60
40
20
0
TIME, SECONDS
0
1
2
3
4
5
6
7

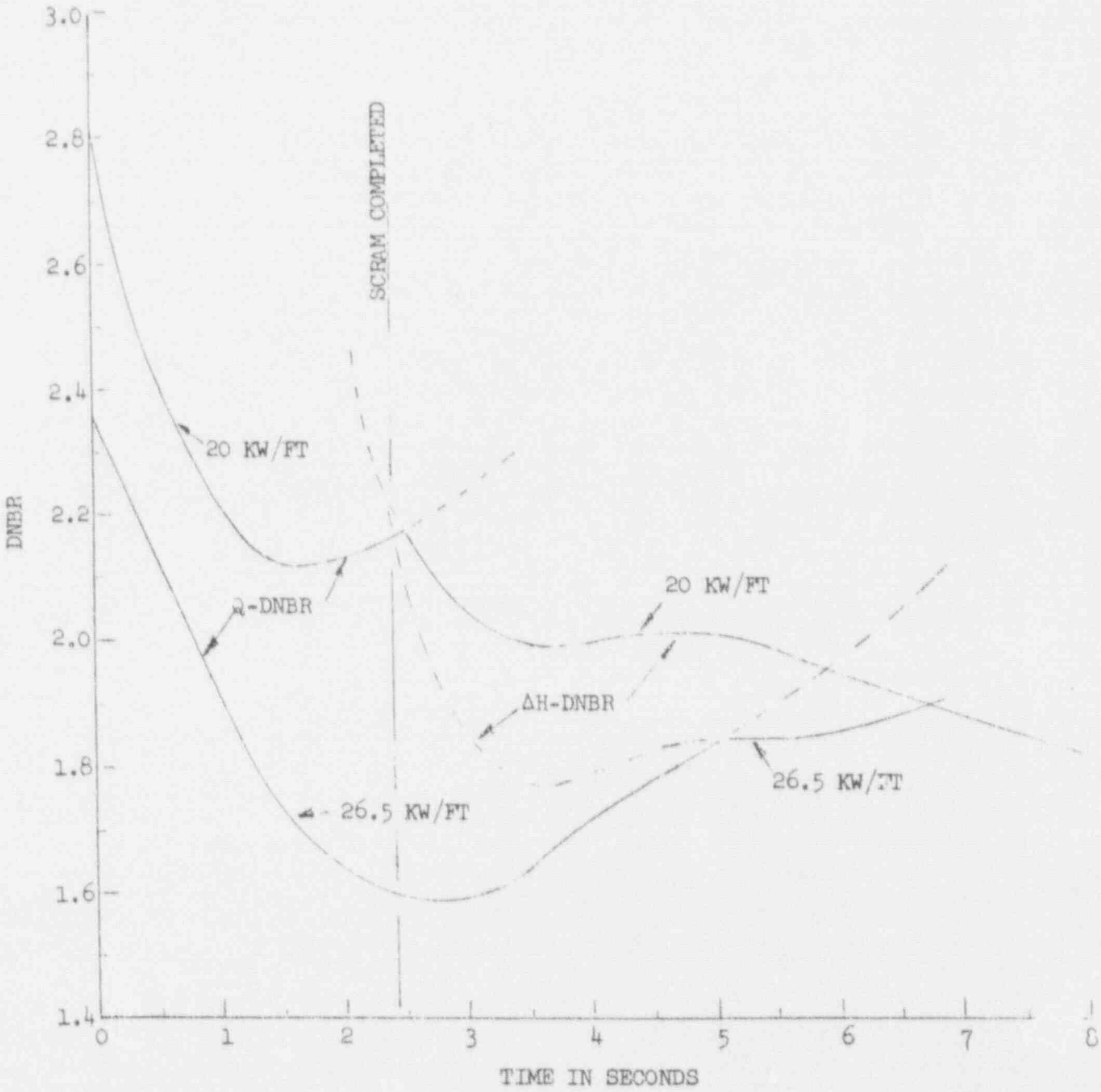

FIGURE II

DNB RATIOS DURING LOSS OF COOLANT FLOW
ACCIDENT, SAXTON 2 X 2 ASSEMBLY

For Div of Compliance

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