Proposed Change No. 125
Supplement No. 3

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WYR 75-126

YANKEE ATOMIC ELECTRIC COMPANY

YANKEE

20 Turnpike Road Westborough, Massachusetts 01581

November 7, 1975

RECEIVED NOV 11 1975 U.S. NUCLEAR REGULATORY COMMISSION Mail Section

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

Attention: Office of Nuclear Reactor Regulation

Reference:
(1) License No. DPR-3 (Docket No. 50-29)
(2) Proposed Change No. 125 (July 14, 1975).
(3) Proposed Change No. 125, Supplement No. 1 (October 10, 1975).
(4) Proposed Change No. 125, Supplement No. 2 (October 28, 1975).
(5) Letter from R. A. Purple to R. H. Groce, dated October 30, 1975 regarding questions on Proposed Change No. 125.

Subject: Core XII Analysis

Dear Sir:

The attached information is provided in response to your letter (5) regarding Proposed Change No. 125. Answers to questions 2 and 6 require transmittal of information of the type which Exxon Nuclear maintains in confidence and withholds from public disclosure. The information has been handled and classified proprietary by Exxon Nuclear in accordance with their procedures and standards, and we hereby make application for withholding from public disclosure this information in accordance with the provisions of 10 CFR 2.790(b) for the following reasons:

1. It reveals certain distinguishing aspects of fuel design where prevention of its use by any of Exxon Nuclear's competitors without license from Exxon Nuclear constitutes a competitive economic advantage over other companies.

2. Its use by a competitor would reduce his expenditure or resources or improve his competitive position in the design and manufacture of a similar product.

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The Yankee Atomic Electric Company has a proprietary agreement with the Exxon Nuclear Company and has handled this information in accordance with that agreement.

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For yo' convenience both a proprietary response (Attachment A, fou. copies) and a non-proprietary response (Attachment B, forty copies) are provided. All pages containing proprietary information are clearly designated as such and should receive no public disclosure.

This supplement to the Proposed Change No. 125 has been reviewed by the Nuclear Safety Audit and Review Committee.

We trust you will find this information satisfactory; however, should you desire additional information feel free to contact us.

Very truly yours,

YANKEE ATOMIC ELECTRIC COMPANY

D.E. Vandenburgh

D. E. Vandenburgh
Vice President

COMMONWEALTH OF MASSACHUSETTS)
)ss.
COUNTY OF WORCESTER)

Then personally appeared before me, D. E. Vandenburgh, who being duly sworn, did state that he is a Vice President of Yankee Atomic Electric Company, that he is duly authorized to execute and file the foregoing request in the name and on the behalf of Yankee Atomic Electric Company, and that the statements therein are true to the best of his knowledge and belief.

Armand R Soucy

Armand R. Soucy Notary Public
My Commission Expires September 9, 1977

Question 1

Show either by analysis or by previous plant experience that the dissimilar materials between tie rods and fuel rods will behave satisfactorily regarding differential thermal expansion.

Response

The ENC Yankee Rowe fuel was designed and fabricated using a 304L stainless steel skeleton and Zircaloy-4 fuel cladding. The arrangement utilizing dissimilar metals for structure a[illegible]l cladding is essentially the same arrangement that is used by exi[illegible]el in Yankee Rowe, which is in its second cycle of operation with [illegible]nt performance difficulties. This material combination is also th[illegible]s previously used in two ENC fuel assemblies for the Ginna react[illegible]Ginna assemblies have performed satisfactorily to date after a[illegible]ely 18 months of reactor operation.

Column buckling of the tie rods (guide bars) due to differential thermal expansion between the stainless steel skeleton and Zircaloy-4 fuel cladding was also considered in the Yankee Rowe fuel design. Conservative analyses show that a minimum safety factor of 2.8 exists between the critical buckling load and the load that is generated by frictional forces due to differential thermal expansion.

Question 2

Provide a detailed drawing and description of the spacer grids, including the connecting scheme betw.en rods (fuel-rods-to-spacer or tie-rod-to-spacer).

Response

The following proprietary drawings show the connection between the tie rods (guide bars) and the spacer grids:

XN-302010,	spacer assembly, type A
XN-302011,	spacer assembly, type B
XN-302014,	guide bar (corner)
XN-302015,	guide bar (side)
XN-302004,	fuel bundle skeleton assembly (type A)
XN-302005,	fuel bundle skeleton assembly (type B)

The spacer grid assemblies are attached to the guide bars in the skeleton assembly by weld joints made by tungsten inert gas arc welding methods. In addition to quality control requirements, the welding process is controlled to assure that each joint exhibits a minimum of 500 lbs ultimate strength in shear. Conservative analysis and testing demonstrates a joint strength requirement of no more than 145 lbs.

Question 3

Discuss out-of-pile proof tests, if any, in support of fuel assembly design verification.

Response

Listed below are the out-of-pile proof tests which were performed to verify the adequacy of the Yankee Rowe fuel design:

A. Yankee-Rowe locking system strength test

This test was performed to verify that the locking devices used to attach the upper nozzle assembly to the guide bars was sufficiently strong to withstand design handling loads. Test results indicated that minimum safety factor of 2.0 exists between the yield point load and the design load with only four of the eight total guide bars carrying the load.

B. Spacer grid springs and dimples

Tests were performed on production spacer springs to verify force deflection characteristics. In addition, tests were conducted on spacer dimples to determine the support stiffness which was used in design calculations for contact stress in the fuel cladding.

C. Cladding to Spacer contact friction

Tests were performed to determine the friction loading between the spacer grid contact points and the fuel rod cladding. This information was used to verify the adequacy of assumed friction coefficients in calculations to establish the spacer grid and guide bar loads due to differential thermal expansion.

D. Spacer Assembly strength test

Tests were performed on a typical Yankee Rowe Spacer Assembly to verify its structural integrity. Data obtained from these tests were used to verify the adequacy of the spacer design from the static and fatigue standpoints.

E. Lower Nozzle Strength Test

Strength tests were performed on the lower nozzle assembly simulating loading during reactor operation. Test results showed that the design was adequate by a large margin.

F. Upper Tie Plate Strength Test

Strength tests were performed on a Yankee Rowe production upper tie plate. Data from this test confirmed that the upper nozzle design was adequate.

G. Pressure Drop Tests

Tests were performed on a prototypic Yankee Rowe fuel assembly utilizing production parts to verify calculated component pressure loss coefficients. Loss coefficients were obtained for the inlet hardware grid spacers, bare rod friction, and outlet hardware over a Reynolds Number range of ~100,000 to ~450,000. Test results provided the basis for determination of the assembly and assembly component pressure drops which was used to confirm the thermal-hydraulic compatibility of the ENC fuel and the existing fuel.

H. Fretting Corrosion Tests

A test was performed on a prototypic fuel assembly as in section G above to demonstrate the adequacy of Yankee Rowe fuel design with respect to corrosion, fretting corrosion, and mechanical wear under reactor hydraulic conditions for approximately 276 hours. Test results showed no evidence of corrosion or fretting corrosion. Some mechanical wear was observed on the upper nozzle hold down springs and bolts due to lateral vibration of the hold down springs. Further testing was directed at evaluation of this wear and resulted in the determination that the wear was self limiting at approximately .008 inch total which is not detrimental to fuel assembly performance.

Question 4

Discuss the response of fuel assemblies regarding seismic and LOCA conditions.

Response

The column buckling strength and other structural strength characteristics of the ENC assemblies are at least equivalent to the existing assemblies. Dynamic loading tests simulating LOCA loads using a prototype ENC assembly are scheduled for the first half of 1976.

Question 5

Provide input values and results for the cladding creep collapse calculation.

Response

Listed below are the input values used in the creep collapse calculations for the Yankee Rowe Fuel:

o	Design exposure (Assembly Average)	30,000 MWD/MTU
o	Design basis power history	Table 1
o	Design basis fast flux history	Table 1
o	Fuel rod prepressurization level (Helium)	125 ± 5 psig
o	Initial pellet density	94.0 ± 1.5% T.D.
o	Final pellet density (after densification)	96.5% T.D.
o	Pellet dish volume	1.0 ± .3%
o	Cladding thickness (minimum 2σ)	.0232 inch
o	Cladding outside diameter	.365 ± .002 inch
o	Cladding inside diameter	.317 ± .0015 inch
o	Cladding material	Zircaloy-4
o	Coolant pressure	2015 psia
o	Axial location of collapse	80% above bottom of core
o	Length of fuel column	91.0 inch
o	Fuel rod plenum volume	.2348 inch^3
o	Initial cladding ovality (2σ)	.00094 inch
o	Helium Absorption	10% initial
o	Cladding Temperature increase in gap due to radiation from pellet ends considered	

Results of the creep collapse calculations are summarized in Figure 1 which shows ovality as a function of time. Ovality at the end-of-life (21000 hours is calculated to be approximately .060 inches. Clad collapse is not predicted for the residency time of the fuel in the core.

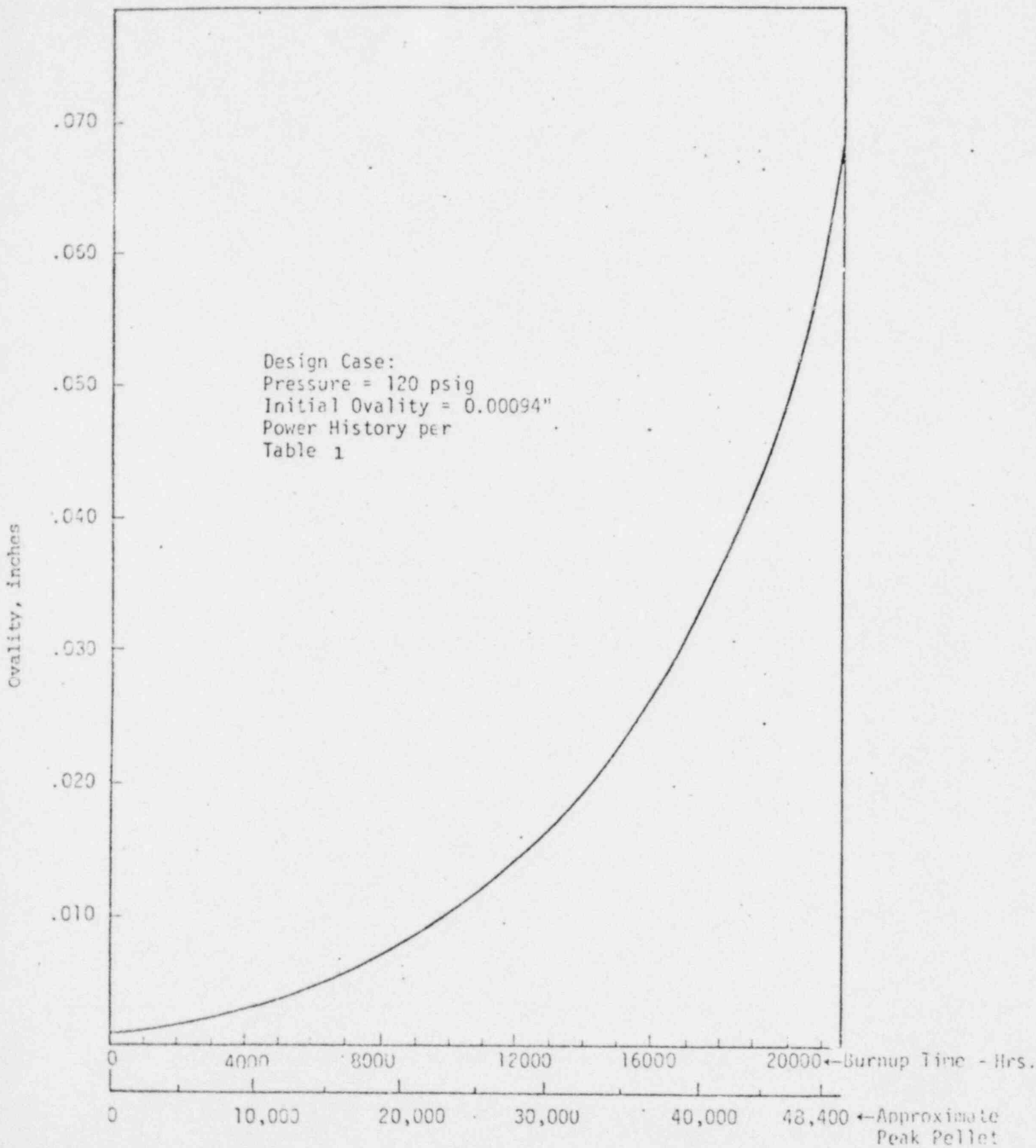

FIGURE 1 FUEL ROD OVALITY VS. BURNUP TIME

TABLE 1

DESIGN BASES VALUES

FOR ROD PRESSURIZATION

TIME PERIOD (Hours)	LINEAR HEAT RATING (kw/ft) Rod	Pellet	FAST FLUX (n/cm^2 - sec > 1 Mev) Rod Peak
0-1000	6.93	8.77	$5.29\ (10)^{13}$
1000-3153	6.93	8.67	$5.29\ (10)^{13}$
3153-6827	6.88	8.34	$5.29\ (10)^{13}$
6827-10500	6.79	7.81	$5.29\ (10)^{13}$
10500-15750	5.73	7.24	$4.51\ (10)^{13}$
15750-21000	5.51	6.48	$4.42\ (10)^{13}$

Question 6

Describe the fuel pellet design (chamfer, dish, etc.) and compare with Core XI pellets and any similar Exxon design for which applicable experience is available.

Response

The response to this question is proprietary to Exxon Nuclear.

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DESCRIPTION:

Ltr re their 7-14-75 tech specs change submittal & our 10-30-75 ltr.......... trans the following:

ACKNOWLEDGED

DO NOT REMOVE

PLANT NAME: Yankee Rowe

ENCLOSURES:

Suppl #3 to Proposed tech specs change #125 ....(40 cys encl rec'd)

(NON-PROPRIETARY VERSION)

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