NRC FORM 195
(2-76)

U.S. NUCLEAR REGULATORY COMMISSION

DOCKET NUMBER 50-315

FILE NUMBER

NRC DISTRIBUTION FOR PART 50 DOCKET MATERIAL

TO: Mr Rusche	FROM: Indiana & Michigan Pwr Co New York, NY J Tillinghast	DATE OF DOCUMENT 10-1-76 DATE RECEIVED 10-5-76

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DESCRIPTION

Ltr notarized 10-1-76....re our 9-3-76 ltr....trans the following:

PLANT NAME: D C Cook #1

ENCLOSURE

Addl info concerning ~~EXXON~~ Rpt XN 76-25 & XN-75-39 which furnished info concerning operation at 100% pwr........

ACKNOWLEDGED

DO NOT REMOVE

SAFETY — FOR ACTION/INFORMATION — ENVIRO 10-7-76 ehf

	SAFETY			ENVIRO
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/	BRANCH CHIEF:	Ziemann (5)		BRANCH CHIEF:
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/	LIC. ASST.:	Diggs		LIC. ASST.:

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INDIANA & MICHIGAN POWER COMPANY
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October 1, 1976

Donald C. Cook Nuclear Plant Unit No. 1
Docket No. 50-315
DPR No. 58

RECEIVED
OCT 5 1976
U.S. NUCLEAR REGULATORY COMMISSION
Mail Section

Mr. Benard Rusche, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Rusche:

This letter is in response to Mr. Dennis L. Ziemann's letter of September 3, 1976 which requested additional information concerning Exxon Nuclear Company Reports XN-76-25 and XN-75-39. We submitted these reports in support of our July 20, 1976 request for an amendment to the above captioned operating license to permit operation at 100% of rated power for fuel cycle 2.

Attached you will find a response to each question from Enclosure 1 of Mr. Ziemann's letter, except for 4.0/11/1, as well as responses to questions 1 (portion in brackets only), 6,7,11,12,22,23,26,33,34, and 35 of Enclosure 2. The balance of the questions in Enclosure 2 were identified as being generic in nature with no response necessary for Donald C. Cook Nuclear Plant Unit No. 1. The questions to be answered are as called out in Mr. Ziemann's letter and subsequently modified at a meeting

held between the NRC Staff, Exxon Nuclear Corporation, and ourselves on September 14, 1976.

Very truly yours,

John Tillinghast

John Tillinghast
Vice President

JT:mam
Attachment

Sworn to and subscribed to before me this 1st day of October 1976 in New York County, New York

Kathleen Barry

Notary Public

KATHLEEN BARRY
NOTARY [illegible], State of New York
No. 41-[illegible]92
Qualified in Queens County
Certificate filed in New York County
Commission Expires March 30, 1977

cc: G. Charnoff
R. J. Vollen
R. C. Callen
P. W. Steketee
R. Walsh
R. S. Hunter
R. W. Jurgensen - Bridgman

QUESTIONS AND ANSWERS FROM ENCLOSURE 1
(concerning XN-76-25)

Question 2.0/4

This section indicates that 1 or 2 fingers of control rod E-3 had broken off during rod timing tests; and that an inspection program is being developed to examine control rod E-3 and "possibly additional control rods during the upcoming refueling." Please indicate your plans for inspection of control rods; and, if you do not plan to inspect all rods, provide a basis for this decision.

Answer 2.0/4

The apparent failure of the control rod in core location E-3 is only the fourth such incident of failure known to our NSSS supplier and only the second occurrence to an RCC of the type in Unit No. 1. The RCC assembly, R-43, is located in a Region 1 fuel assembly which is scheduled for discharge during the refueling. Visual inspection will be performed to insure the fuel handling equipment can latch onto and transport the assembly - control rod unit to the spent fuel pit. It is premature, at this point in time, to decide on a definitive inspection program for that control rod since the program itself will depend upon our specific findings from the visual inspection. It is our intent, however, to identify the type of failure and if possible the cause of failure.

With regard to the inspection of other control rods, the failure of R-43 is viewed as an isolated incident not indicative of a generic problem. This isolated event in itself is not considered significant enough to justify a major on-site examination program. There is the need to demonstrate that all of the other control rods are operating properly and are capable of performing their design function. With this in mind we intend to perform rod drop timing tests at the end of cycle 1 operation (damaged RCC R-43 excluded), then perform a drag test on each RCC and lastly perform the rod drop timing tests again as indicated in Section 10, page 70 of our submittal, XN-76-25. If additional RCC assemblies are identified as suspect on the basis of anomalous end of cycle 1 drop tests, a visual examination shall be performed on those assemblies. The scope of any further examination of those assemblies depend upon the findings of the visual inspection.

Questions 4.0/11/4
and 4.0/23/2

It is not clear that ANSYS or AXIBOW are NRC approved codes. Please cite the document via which these codes were approved.

Answers 4.0/11/4
and 4.0/23/2

XN-76-25 incorrectly indicated that the NRC had approved the ANSYS and AXIBOW design codes. These codes have been used extensively and are considered by Exxon Nuclear to provide acceptable methods for evaluating fuel design.

Question 4.0/19/1

Please clarify and justify the statement that "no significance in reactor spring relaxation is expected" (what is the spring life expectancy and what is the amount of spring relaxation anticipated with time?)

Answer 4.0/19/1

The calculated fast flux in the holddown spring region is 3.3×10^{12} n/cm^2/sec ($>$ 1 Mev). For three cycles of operation, corresponding to 7.6×10^7 secs, the fast neutron fluence is 2.5×10^{20} n/cm^2. This fluence can cause up to 20% relaxation of the springs, although the probable relaxation is less (see Figure 4.8, XN-75-39). There is sufficient margin the spring design to cover more than 20% relaxation.

Question 4.0/26.4

Please cite the reference in which data of Exxon fuel rod length changes with burnup is reported.

Answer 4.0/26.4

ENC has measured fuel rod length changes on fuel from three reactors: Oyster Creek, Big Rock Point, and R. E. Ginna. Approximately 100 rods have been measured, covering burnups from about 6000 MWD/MTM to 19,000 MWD/MTM. The results of these measurements are shown in the attached Figure 1. Maximum fast flux fluence for D. C. Cook is calculated not to exceed 7×10^{21} n/cm^2 ($>$ 1 Mev). Using the R. E. Ginna fuel rod length change as a basis, the fuel rod extension would be less than 8×10^{-3} in/in or 1.15 inches.

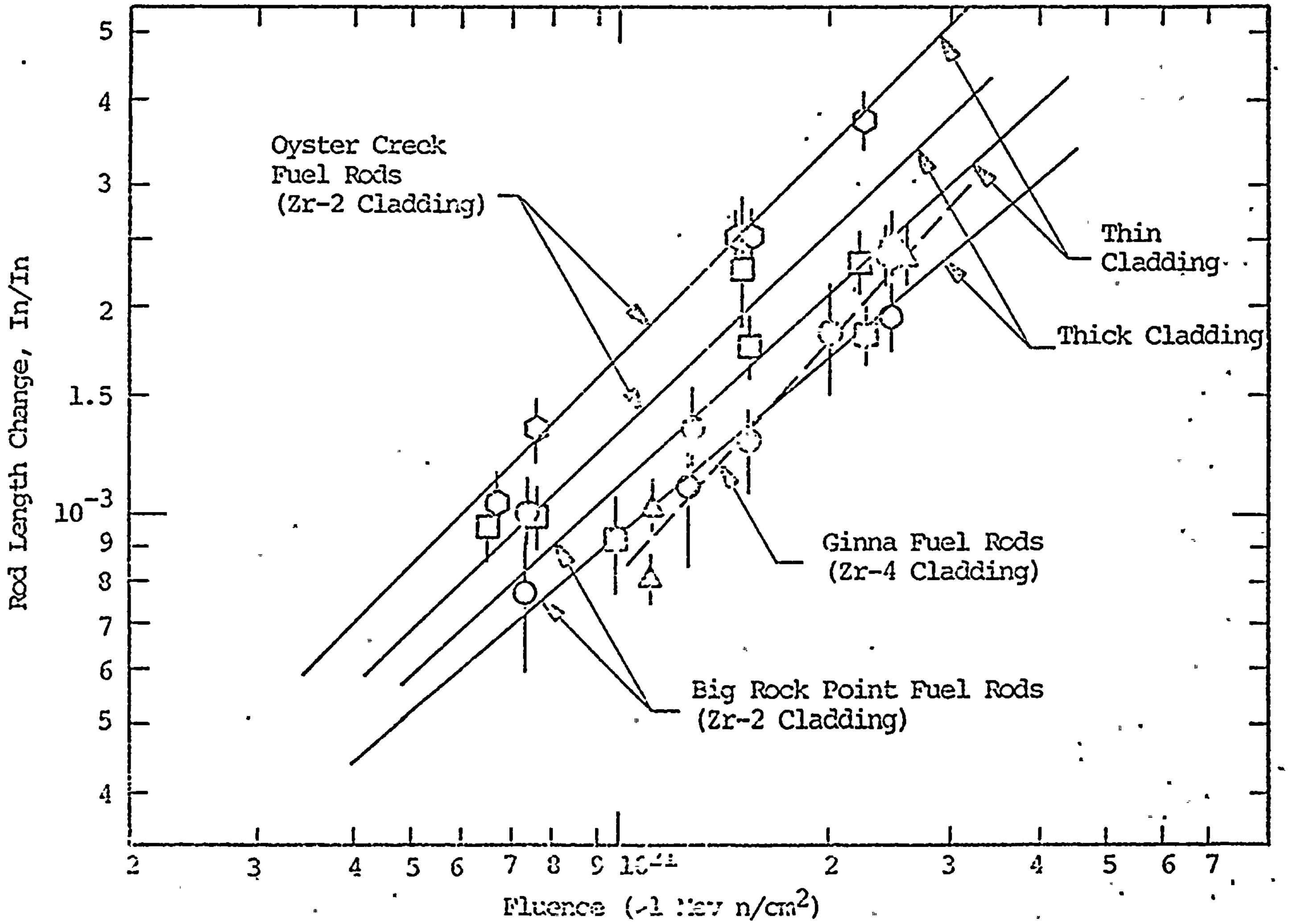

FIGURE 1. ENC Fuel Rod Length Change Data

Question 5.0/55/5.2

It is stated that "That Neutronics design methods utilized to calculate the data presented herein are consistent with those described in Reference 5.1...". However, we note that Reference 5.1 has not been submitted. Completion of our review requires receipt of that Reference.

Answer 5.0/55.5.2

Draft copies of Revision 1 to XN-75-27 were given to the staff on September 15. The report will be submitted formally by October 1, 1976.

Question 10/70 A.

Provide the acceptance criterion for each test and discuss how the measured parameter(s) relates to the values in the accident analysis.

Answer 10/70 A.

1. Rod Position Indication System

The acceptance criterion is based on the technical specifications which require that all control rod position indicator channels are within 12 steps of their bank demand position.

2. Rod Drive Mechanism Timing

The acceptance criterion is that the control rod mechanisms function so as to move the rods in/out when the motion is called for by the operator.

3. Rod Drop Timing

This test is acceptable if each control rod drops from the fully withdrawn position to the dash pot entry in $\leq$1.8 seconds.

4. Initial Criticality

The acceptance criteria are:

a) criticality shall be achieved within 75 ppm of the analytically predicted value,

b) the neutron flux level at which nuclear heating (Doppler feedback) effects become measurable is established so that all zero power tests shall be conducted below that level,

c) the reactivity measured using the reactivity computer shall be within 4% of the reactivity obtained from reactor period measurements,

5. Critical Boron Concentrations

The acceptance criterion is that the difference between measured and predicted boron concentrations shall be within 75 ppm,

6. Control Rod Bank Worths

This test is acceptable if the integral worth of the control banks is within $\pm$ 10%, or $\pm$ 100 pcm if measured worth is less than 1000 pcm,

7. Isothermal Moderator Temperature Coefficient

The test is acceptable if the isothermal temperature coefficients are determined to be negative,

8. Incore Moveable Detector Maps

The acceptance criteria is such that the relative assembly powers are within $\pm$ 10% for $P_i \geqslant 0.9$ and $\pm$ 15% for $P_i \leqslant 0.9$. The values of F_Q, $F_{\Delta H}$, and quadrant power tilts are to comply with the technical specifications,

9. Excore - Incore Detector Calibration

The excore and incore detectors are to be calibrated to each other.

10. Power Coefficient

The power coefficient shall be determined.

The startup physics tests will be performed to verify that the core can be operated safely and to make an initial evaluation of the expected performance as predicted by design calculations. Accident analysis configurations in general are not used, thus many measurement results do not relate directly to the values of the parameters used in the accident analyses. However, the measured values of the parameters should relate to the values used in accident analyses such that they fall in the range of the values used. Further, a direct comparison of the measured and safety limit values can be misleading. In many cases the most conservative combination of reactivity coefficients is used in a transient analysis, even though the extreme coefficients used do not occur at the same conditions of lifetime, power level, temperature, and boron concentration assumed in the analysis.

Question 10/70 B.

Describe in detail the control rod bank reactivity worth tests, the maximum deviations from calculated values expected, the criteria used for determining these maximum deviations and the procedures to be followed if these deviations are exceeded.

Answer 10/70 B.

Three distinct rod worth tests are planned for the Donald C. Cook Nuclear Plant Unit 1, Cycle 2 startup physics test program. They will consist of rod worth measurements for control banks D and C under HZP conditions. The tests are:

a) D - bank worth

b) C - bank worth with D fully inserted

c) D + C worth in the overlap mode

The reactivity worths of the control rod banks are obtained by monitoring reactivity changes while maintaining the reactor approximately critical through boron/control rod bank exchanges. Following the establishment of a constant dilution/ boration rate, the controlling control bank is periodically inserted/ withdrawn to compensate for the changing reactor coolant system boron concentration. The reactivity change resulting from each control bank movement is recorded by a reactivity computer. The differential control rod reactivity worth is defined as the change in reactivity per unit change in bank position about an average bank position, and the integral worth is the summation of the individual reactivity steps between fully inserted and fully withdrawn positions.

The results of the control rod worth measurements are compared to the values obtained from design calculations. The maximum difference expected between the measured rod worths and design predictions is $\pm 10\%$.

The acceptance criteria used for comparison of measured results with design predictions are based on the experience that has been acquired as a result of the startup physics testing programs of the PWR's which are now commercially operating. However, these acceptance criteria are viewed as

guides to possible measurement or design errors and failure to meet these criteria does not by itself constitute a safety problem. If observed differences between measured and predicted values are beyond the acceptance range, one or more of the following actions will be taken:

a) re-examining the measurement data for possible computational errors or unaccounted experimental perturbations

b) verifying that the measurement conditions are consistent with the assumptions in the design prediction

c) repeating the measurement or performing additional tests if the discrepancy remains unresolved

d) evaluating the impact of the value measured with respect to the appropriate accident analyses sensitive to the measured parameter

e) repeating the appropriate accident analyses using values consistent with the measured values of the parameter in question

f) revising the plant operating procedures in a conservative manner to permit operation while a review of the safety implications is being made.

Question 10/70 C.

Minimum test requirements for certain tests are given below.

Control Rod Group Worth

At least the first and second regulating banks. If either of those measured values differs from the predicted by more than 10%, then measure the first shutdown bank. If the sum is off more than 10%, then measure the worth of the N-1 configuration. (All rods inserted into the core except the rod predicted to have the highest worth).

Moderator Temperature Coefficient

At least two configurations:

The all-rods-out, and first-bank in.

Power Distribution

At least one configuration at zero power:

All-rods-out; at least 1 map at power before 75% power is reached.

Power Coefficient

At least 1 measurement at a power level over 65% of rated power.

Answer 10/70 C.

We intend to incorporate into our startup program the requirements in 10/70 C.

Question 10/70 D.

State your schedule for submitting to the NRC a brief summary report of physics startup test results. This report should include both measured and predicted values. If the difference between the measured and predicted value exceeded the acceptance criterion, the report should discuss the actions that were taken and justify the adequacy of these actions.

Answer 10/7 D.

Following each of the startup physics tests that will be performed at the Donald C. Cook Nuclear Plant Unit 1 power station, a formal report will be issued. This report will describe in detail the physics tests performed as part of the startup program, and it also will contain the specific results for each of the tests performed. The startup physics test report will be issued to the NRC approximately 90 days following the completion of the testing program associated with the reload.

The procedure is to compare the measured value of the parameter(s) of interest associated with each test to their respective predicted values in this report. If the difference between the measured and predicted values exceed the acceptance criterion for the test, the report shall discuss the actions that were taken and justify the adequacy of these actions.

Please note the following changes to our original submittal of XN-76-25, section 10.0, page 70:

1. Change item 1 to read:

Rod Position Indication System

This test is performed prior to criticality in the hot shutdown condition to verify the system performs satisfactorily.

2. Change item 6 to read:

Control Rod Bank Worths

To measure the differential bank worths at hot zero power in both the overlap and non-overlap conditions.

3. Change the number of the third item from 2. to 3.

4. In XN-76-25 p. 41a, drawing dimension for height of lower tie plate should be 2.720.

QUESTIONS AND ANSWERS FROM ENCLOSURE 2
(concerning XN-75-39)

Question 1.

The introduction of the report should clarify how the report will be used with regard to reload licensing applications. For example all Westinghouse plants may not have the same operational procedures and limits so that the maximum power distributions reached during transients may differ from plant to plant. (Does the Exxon fuel design limits bound the worst 15 x 15 plant operational limits?) The introduction and the design basis sections should include the bounding conditions considered and provide the justification for the values used.

Answer 1.

The design limits for 15 x 15 PWR fuel assemblies are as follows:

<table>
<tr><th></th><th>3 Loop Plant</th><th>4 Loop Plant</th></tr>
<tr><td>Nominal Design Pressure, psia</td><td colspan="2">2250</td></tr>
<tr><td>Maximum Design Pressure, psia</td><td colspan="2">2500</td></tr>
<tr><td>Flow Velocity (average), ft./sec</td><td>15.0*</td><td>15.5*</td></tr>
<tr><td>Peak LHGR, kw/ft.</td><td colspan="2">13.5</td></tr>
<tr><td>Batch Average Burnup, MWD/MTM</td><td colspan="2">33,500</td></tr>
<tr><td>Peak Pellet Burnup, MWD/MTM</td><td colspan="2">47,500</td></tr>
<tr><td>Maximum Cladding Temperature, °F</td><td colspan="2">850°F ID
675°F OD</td></tr>
<tr><td>Maximum Coolant Temperature, °F</td><td colspan="2"></td></tr>
<tr><td>Inlet</td><td colspan="2">550</td></tr>
<tr><td>Outlet</td><td colspan="2">642</td></tr>
<tr><td>Operational Transients</td><td colspan="2">Per Table 4.5 in XN-75-39</td></tr>
<tr><td>Maximum Full Power Hours</td><td colspan="2">22,000</td></tr>
</table>

*Requires different holddown spring design for 3 and 4 loop plants.

Question 6.

The thermal conductivity of Zr-4 is based on data published by Scott (reference 1, XN-75-39). There is a discrepancy in the equation for thermal conductivity given in your report compared to the equation given in the report referenced. Please clarify.

Answer 6.

The equation given in XN-75-39 for the thermal conductivity of Zircaloy is based on the Nickel-free Zircaloy-2 data of Scott and not on the Scott equation for Zircaloy-4. The use of the Nickel-free Zircaloy-2 data gives conservatively low values for Zircaloy-4 thermal conducitivty.

Question 7.

Why is the equation for fuel densification for fuel densities between 85 to 91.9% T.D. for $t < 10$ hours inconsistent with the Exxon model submitted for staff approval (reference letter: R. Nilson to J. F. O'Leary, December 13, 1973).

Answer 7.

There is a typographical error in XN-75-39 for densities of 87 to 91.9% TD. The correct term for $t < 10$ hours is:

$$\Delta\rho/\rho_{max} = [0.0148\ t]$$

Question 11.

Your design limits appear to be incomplete. Design limits for buckling (instability), fatigue, etc., as well as stress and strain, should be provided. In addition, there should be a specific value assigned for the design limits for the events catagorized in question 8.

Answer 11.

Design limit requirements are scattered through the report. For steady state conditions the mechanical design limits and stress or load intensities for each component is summarized in the attached Table (Answer 11).

Table (Answer 11)

Component	Stress Intensity or Load	Design Basis	Limit Value	Safety Factor
Fuel Rod				
Primary Stresses:				
BOL - Cold	-13,200 psi	1/3 UTS	26,670	2.04
BOL - Hot	- 9,600 psi	1/3 UTS	16,670	1.72
Secondary Stresses:				
BOL - Cold	-13,950 psi	1/2 UTS	42,500	3.05
BOL - Hot	-10,120 psi	1/2 UTS	25,000	2.47
Total Stresses:				
BOL - Cold	-37,560 psi	UTS	85,000	2.26
BOL - Hot	-34,360 psi	UTS	50,000	1.46
EOL - Hot	-40,740 psi	UTS	50,000	1.23
Instantaneous Collapse				
BOL - Cold	2485 psi	(1)	5002 psi	2.01
EOL - Hot	2485 psi	(1)	2527 psi	1.02
Fatigue Usage Factor	0.27	-	0.67	2.68

Component	Stress Intensity or Load	Design Basis	Limit Value	Safety Factor
Spacer				
Side Plate Loading - Cold	Yield at 2560 lbs	No yielding 2 x Bundle Weight (wet)	2500 lbs	1.04
Spacer Intersection Strip Weld Strength	126,000 psi (2)	(3)	55,000 psi	2.29
Guide Tube to Spacer Weld Strength	> 300 lbs/tab	(4)	150 lbs/tab	2.00
Crush Strength - During Shipping	1890 lbs	(5)	1740 lbs	1.08
Tie Plates:				
Primary Load	> 8000 lbs (no yielding detected in tests)	2.5 x Bundle Weight (dry)	3500 lbs	2.28
Tie Plate Locking Hardware:				
Primary Load - Sleeve	> 1200 lbs	(6)	600 lbs	2.00
Primary Load - Locking Rings	> 245 lbs	(7)	175 lbs	1.40
Guide Tubes:				
Primary Stresses	330 psi	1/3 UTS	11,670 psi	35.4
Secondary Stresses	9250 psi	1/3 UTS	17,500 psi	1.89
Buckling	230 lbs	Euler Column	465 lbs	2.02

Component	Stress Intensity or Load	Design Basis	Limit Value	Safety Factor
Springs:				
Spacer - Cold	101,000 psi	Y.S.	150,000 psi	1.49
Plenum - Cold	95,370 psi	Shear Stress	100,000 psi	1.05
Holddown - Cold	169,000 psi	Y.S.	180,000 psi	1.07

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(1) Design basis is determined using Timoshenko and Gere equation for tube buckling (see "Theory of Elastic Stability", 1961).

(2) The 126,000 psi is the measured failure stress.

(3) Design basis is that weld to have strength equal to or greater than base metal.

(4) Minimum load is ~ 2.5 times the guide tube load of 230 lbs assumed to be concentrated evenly on the four tabs of one spacer.

(5) Design Basis is established that a spacer shall withstand a lateral load equal to 6 times the bundle weight distributed evenly over 5 spacers (the upper and lower spacer do not carry much load during shipping).

(6) Minimum allowed load is ~ 2.5 times calculated guide tube load of 230 lbs.

(7) Minimum allowed load based on 2.5 times bundle weight evenly distributed over the 20 locking devices.

Question 12.

Discuss local primary membrane stress with examples. Examples for general primary membrane stress, namely stress in the cladding due to pressure differential across the wall thickness, were provided as well as formulas showing how one would calculate such a quantity. The same type of discussion is needed to define and assess local primary membrane stress.

Answer 12.

A local primary membrane stress is defined as the average stress across any solid section. It considers discontinuities but not stress concentrations and is produced by mechanical loads. This limit is applicable to any local loading such as stresses due to fuel rod spacer support forces. Although these stresses were calculated and reported, i.e., Table 4.3, XN-75-39, the methodology used in the analyses is not described in the generic report.

Question 22.

Describe in more detail the extent of the surveillance program. For example, identify the number of assemblies to be examined and the number of fuel rods to be inspected to evaluate the fuel rod parameters identified in Section 2.5.

Answer 22.

ENC surveillance programs are based on need for confirmation of new designs. In the past, these programs have involved typically two to four assemblies wherein rods are removed for individual inspection. To date ENC has obtained performance data from at least 100 fuel rods. From these inspections, the trend in fuel performance has been established to the extent that future surveillance programs will be setup on a limited basis merely to ascertain that these trends are being maintained. Should some change in fuel processing or design occur, then a more extensive program would be instituted to evaluate the change.

For H. B. Robinson the fuel surveillance program utilizes two characterized fuel assemblies. Post-irradiation inspection at H. B. Robinson will involve the typical visual inspection plus dimensional measurements on bundle length and spacer grid envelope. Both measurements are to determine irradiation growth of zircaloy. The spacing between fuel rods and fuel rods and guides tubes will be measured on four assemblies to ascertain row bow characteristics.

Question 23.

Fuel assembly lift off is not addressed in these reports. Justify why this item was not discussed. If you choose to address this in response to our question, we are specifically interested in the margin to lift-off as a function of percent of rated flow.

Answer 23.

The fuel assembly liftoff force is a function of plant design, i.e., 4 loop plants have a higher coolant flow velocity than 3 loop plants and, therefore, require a different holddown spring design. As a result, we have addressed lift-off in the documentation provided for a particular plant. The holddown spring analysis for D. C. Cook is described in XN-76-25, Section 4.1.3, and is generally applicable for Westinghouse 4 loop plants. A summary of comparable analysis for H. B. Robinson #2 was transmitted with CP&L's letter to Benard C. Rusche, Docket No. 50-261, dated October 17, 1975, and is generally applicable for 3 loop plants.

Question 26.

Show the test results to support your spacer-grid analysis results in Section 4.8.2.

Answer 26.

Structural tests performed on grid spacers have involved the following:

a) Side locking strength tests simulating loads during refueling

b) Static crush tests

c) Intersection weld strength tests

d) Spacer to guide tube weld tests

e) Dimple stiffness test

f) Lateral impact tests

The results of these tests are discussed in the following:

1) The side loading strength and crush tests results are discussed in Section 5.1.4 of XN-75-39.

2) Spacer intersection weld strength were evaluated through tensile tests on 10 intersection sample welds. The average load at failure was 138 lbs with a sigma value of 9 lbs. On a three sigma basis, the minimum load expected would be 110 lbs. For annealed Zircaloy-4 with a minimum tensile strength of 655,000 psi, the load carrying capability at minimum cross-sectional area at an intersection is 48 lbs. Consequently, the weld design provides a safety factor greater than 2.

3) Spacer to guide tube weld strength have been evaluated through both weld qualification and in-progress weld sample tests. The spot weld strengths have ranged from 300 to 400 pounds, consequently, the minimum load at 150 pounds per tab is routinely met with a safety factor of 2.0 or greater.

4) Dimple stiffness tests were performed by compressively loading dimples on production spacers. The results show that dimple stiffness is 1760 lbs/in.

5) Impact tests are discussed as part of the seismic-LOCA analysis documented to be issued.

<u>Question 33</u>.

What specific information will be derived from the fuel surveillance programs to provide the necessary confirmatory data to better predict PCI?

<u>Answer 33</u>.

PCI can cause lengthening of fuel rods via a ratchetting mechanism and can also enlarge the diameter of a fuel rod. ENC monitors both fuel rod diameter and length to determine performance. Pellet stack length measurements and gamma scanning which are part of ENC's surveillance program, can provide additional information pertinent to PCI.

<u>Question 34</u>.

Provide justification, based on short-term fretting-corrosion tests, for assuming that no significant fretting corrosion or mechanical wear will occur over the lifetime of the fuel.

Answer 34.

ENC has run fretting-corrosion tests on 7 prototype PWR assemblies and 5 prototype BWR assemblies. Although these tests cover wide variations in design, the design philosophy and design details for fuel rod support are substantially the same in most cases. Test periods ranged from 500 to 1500 hours at reactor conditions of temperature, pressure and flow. Fuel rod wear depths at spacer contact points has typically ranged from 0.1 to 0.5 mils although wear of up to ~ 1.5 mils in depth has been occasionally observed.

Examination indicates that the wear is due primarily to fuel rod loading and unloading and not fuel rod motion during the test. There has been little or no difference between observed wear for 500 hour, 1000 hour and 1500 hour tests. No active fretting corrosion has ever been observed on any test rod in spite of the fact that spacer springs have been relaxed up to 100% in several test assemblies. We believe that if conditions are conducive to fretting corrosion it will show up in a few hundred hours.

Reload fuel representing 8 of the assembly designs tested have been loaded into a reactor. Five have been in a reactor 3 or more cycles. Detailed examination including profilometry of a number of rods in 3 reactors has not revealed wear significantly different than that observed in fretting corrosion tests. No active fretting corrosion has been observed.

Question 35.

The onsite inspection of new fuel consists of the determination of fuel rod axial position, rod-to-rod spacing and cleanliness. Is this adequate to insure mechanical integrity of the fuel rod assemblies and components after shipment.

Answer 35.

In addition to these measurements, one assembly per truck load is characterized by measuring the spacing or distance of the outer fuel rods to the spacer envelope. The purpose is to have a traceable dimensional check in case of an unpredicted occurrence. ENC has performed extensive shipping tests to assure that the packaging and containers maintain the proper mechanical integrity of the fuel assemblies.