DEC 10 1973

Docket No. 50-249

Commonwealth Edison Company
ATTN: Mr. J. S. Abel
Nuclear Licensing Administrator -
Boiling Water Reactors
Post Office Box 767
Chicago, Illinois 60690

Gentlemen:

Your letter and reload report of September 14, 1973, requested approval of a reload core containing a number of 8 x 8 fuel assemblies. We have reviewed the report and require the information listed in Attachments A and B to continue our evaluation. Attachment A refers to Section 6 of your reload submittal. Attachment B refers to Sections 3, 4, and 5 of NEDO-20103. In addition, the information requested in Attachment C relating to GESSAR questions 3.69 and 4.13 through 4.46 which were transmitted to GE on November 1, 1973, is also required. The response by GE to us to these GESSAR questions is scheduled for December 17, 1973. We also note that you have a commitment to complete an analysis of potential densification in the 8 x 8 fuel by December 1, 1973, and have stated that modifications, if any, to the refueling plans would be submitted by November 1, 1973.

We request that the information in Attachments A and B be submitted by December 17, 1973, with one signed original and thirty-nine additional copies.

Sincerely,

Original signed by
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Directorate of Licensing

Enclosures:
1. Attachment A
2. Attachment B
3. GESSAR Questions of 11/1/73

cc: See next page

R2 App

cc w/enclosures:
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ATTACHMENT A

Dresden 3 Nuclear Power Station

Second Reload License Submittal

Section 6 Safety Analysis

6.1 In reference to figures 6-4 and 6-5 discuss the variation of the scram reactivity function with burnup throughout the cycle. Do these figures represent the worst case during the cycle? Is the proposed technical specification scram time of your letter of August 1, 1973, used? Explain the differences figures curves 6-4 and 6-5 and figures 6 and 7 of your letter of September 14, 1973, concerning maximum allowable in-sequence control rod worth.

6.2 For the steam line break inside containment, provide curves of power flow, quality, MCHFR and pressure differentials in the highest powered 8x8 fuel assembly as a function of time.

6.3 Provide curves of flow, quality at the channel inlet and outlet, convective heat transfer coefficient and the clad temperature of several representative rods in the highest powered 7x7 fuel assembly as functions of time following the design basis loss-of-coolant accident.

6.4 Provide curves similar to Figure C-19 of NEDO-10329 of flow and MCHFR in the hottest 8x8 fuel assembly versus time for both the case of 1) no flow redistribution occurs and 2) flow redistribution is considered.

6.5 Provide curves similar to Figure C-21 of NEDO-10329 which show the sensitivity of the MCHFR in an 8x8 assembly following the design basis loss-of-coolant accident to variations in initial flow.

6.6 Provide the local rod-to-rod peaking factors for each rod, the axial and radial peaking factors and the peak LHGR in the highest powered 8x8 fuel in a) the ECCS analysis and b) at 5,000, 15,000 and 30,000 MWd/t exposure.

6.7 Although the lower stresses in a rod in an 8x8 assembly during a LOCA may result in fewer perforations than in a rod in a 7x7 assembly, the extent of swelling and blockage may be greater. Provide an estimate of the degree of blockage in an 8x8 assembly following the design basis LOCA.

6.8 Demonstrate that the water rod (rod #37) will not be damaged or distorted due to depressurization during the worst liquid or steam-line break LOCA.

6.9 Provide an analysis of the long-term cooling capability of the LPCI system which is applicable to a core with 8x8 fuel assemblies. Provide curves of swollen level and clad temperature versus bundle power for the 8x8 assemblies.

6.10 Provide the range of parameters (e.g., inlet flow and enthalpy, bundle power, peaking factors and flow coastdown and pressure decay rates) in the full scale 8x8 bundle transient or steady-state CHF test that are to be used to "verify the applicability of the existing model " (Section 6.1.2.3). Provide the schedule for submittal of the data and evaluation of these tests.

6.11 Provide a schedule for the submittal of promised information (Section 6.2.3.2) which evaluates the thermal-hydraulic stability of the 8x8 assemblies and the core with a mixture of 8x8 and 7x7 assemblies.

6.12 Provide a schedule for the submittal of the evaluation of abnormal transients, such as core coolant flow decrease, for which "confirmatory calculations are currently in progress " (Section 6.2.3.2), 6.13 Provide specific comparisons of the 8x8 and 7x7 assemblies which demonstrate that the "conservative fuel type" as stated in Section 6.1.3.1 of the D-3 submittal has been used in the analysis of each abnormal operational transient and the nuclear system pressure increase transients in particular.

ATTACHMENT B

8x8 BWR RELOAD FUEL NEDO-20103

Section 3.0 Mechanical Design

3.1 Provide an assembly drawing of the fuel assembly and, if necessary for completeness and clarity, detail drawings of components. The drawing should be similar to Figure 4.2-2 in GESSAR, but the following additional information should be provided:

(a) Dimensional tolerances

(b) Fuel pellet dimensions, including pellet length, edge chamfer and end dishing.

(c) Filler gas pressure and composition, including water vapor and other impurity content.

(d) A description of the getter, including the volume, weight, surface area and alloy constituents.

(e) Water rod dimensions, including diameter, wall thickness and number, location and size of vent holes.

3.2 For tables 3-3, 3-4, 3-5 and 3-6 of NEDO-20103, provide the dimensions of the edge chamfer and end dish of the pellets in each type of rod. For tables 3-3 and 3-5 provide the mean density of the pellets in each type of rod. For Table 3-4, provide the fuel rod diameter, clad thickness, pellet-to-clad gap, active fuel length, plenum volume, and maximum linear heat generation rate for each type of rod.

3.3 Provide the basis for the statement that the flow-induced fuel rod "vibrational amplitude" does not exceed 0.002 inch (Section 3.2.9). Describe the tests and analyses of flow-induced vibration in an 8x8 assembly which have been performed.

3.4 Describe the tests of 8x8 fuel assemblies mentioned in Section 3.2.10 which "have been conducted both out of reactor as well as in reactor." Provide the results of "All tests and post-irradiation examinations" which have indicated that fretting corrosion does not occur.

3.5 Describe the post-irradiation surveillance program planned for the 8x8 reload assembly. Describe the proposed tests and inspections, the number of rods and assemblies involved, and the exposure and time-in-reactor of the assemblies.

Section 4 Thermal-Hydraulic Characteristics

4.1 Demonstrate that the referenced CHF correlation is applicable to an 8x8 assembly. Compare the available data from full scale tests of 8x8 bundles to the predictions based on this referenced correlation. From the test data which most closely approximates the conditions in the hot assembly at normal full power operation, estimate the bundle power which will produce the onset of transition from nucleate boiling. Compare the test parameters with expected operating conditions (e.g., bundle power and flow, inlet enthalpy, axial and local power peaking factors).

4.2 Describe the methods used to calculate the steady state flow distribution in terms of mass velocity between 8x8 and 7x7 assemblies. Estimate the error in calculating the flow in an 8x8 assembly relative to a 7x7 assembly. Provide the basis for this estimate.

4.3 Explain the difference between the tests "performed in single-phase water to calibrate the orifice and tie plate and in both single-and two-phase flow to arrive at best-fit design values for spacer and upper tie plate pressure drop" (Section 4.1.1.2 of NEDO 20103) and the full scale 8x8 tests "performed to determine the local loss coefficients for upper and lower tie plates and fuel rod spacers." Compare the results of these tests to each other and to the pressure drop at various flows and powers calculated using the standard design method.

4.4 For the hot 8x8 and 7x7 assemblies, provide the flow rate, bundle power, axial and local peaking and exit void fraction at normal full power operation.

4.5 Provide the design correlations, including all constants, used to calculate the friction factor, two-phase friction multiplier and two-phase local multiplier described in Sections 4.1.1.1 and 4.1.1.2 of NEDO-20103.

Section 5 Nuclear Characteristics

5.1 Provide comparisons of calculated parameters (e.g., relative power and reactivity coefficients) with experimental observations (critical facilities and reactor irradiation) for the cores referred to in Section 5.2 of NEDO 20103 which contain:

(1) mixtures of 6x6 and 7x7 bundles,

(2) mixtures of 8x8 and 9x9 bundles, and

(3) only 7x7 bundles.

5.2 Define the terms "controlled" and "uncontrolled" used in Figure 5.2 of NEDO-20103. Provide comparisons of multiplication factors as functions of void fraction similar to Figure 5.2 for the 50% controlled case.

5.3 Provide the assumptions and bases used to calculate the maximum local peaking as a function of exposure as shown in Figure 5.7 of NEDO-20103, (e.g., was an infinite lattice of one bundle type assumed; what value of void fraction, what control rod program and gadolinia distribution were assumed?) Explain why the variation in maximum local peak with exposure is different for the two lattices.

5.4 What is the initial value of maximum local peaking and the variation of local peaking with burnup in the initially unexposed 8 x 8 assemblies placed in the array of exposed 7 x 7 assemblies? Compare this maximum local peaking with that which would occur with initially unexposed 7 x 7 assemblies loaded in the exposed 7 x 7 array.

5.5 What is the maximum expected exposure of a six inch axial segment of 1) the 7x7 bundles which remain in the core, 2) the 8x8 bundles which are to be loaded in the core and 3) fresh 7x7 bundles which could be loaded in lieu of 8x8 bundles. If greater than 22 GWd/t, provide the maximum local peaking factor for each type of bundle out to its maximum exposure.

5.6 Provide the expected operating power level as a function of exposure of 1) the 8x8 bundles and, 2) the 7x7 bundles which could have been used in lieu of 8x8 bundles.

5.7 If fuel shuffling is to be done, describe the procedures to be used. What calculations are done to a) determine local and gross peaking factors, b) verify shutdown margin and, c) determine weighting factors used in calculating behavior following accidents involving significant spatial effects, such as a rod drop accident

ATTACHMENT C

QUESTIONS RELATING TO GESSAR

3.69 Provide detailed information concerning the effects of a steam line break on reactor internals. Include the effects on fuel assemblies and the possible interference to control rod insertions.

4.0 Reactor

4.13 With respect to Section 4.2.1.1.2.4 of GESSAR, justify the use of maximum shear stress theory and related stress intensity limits with respect to Zirconium fuel cladding. Clarify whether the fuel cladding is considered a ductile material after exposure to reactor core operating conditions or is classed as a non-ductile material, with design, analysis and qualification procedures based on brittle material behavior.

Section III of the ASME Code specifically excludes "tubes or other forms of sheathing used only for cladding nuclear fuel" and is not considered an applicable guide in this area.

4.14 Provide fuel handing and shipping design loads and their relation to stress and strain limits to establish that design limits have not been violated during handling or shipping.

4.15 The design values for pressure, temperature, neutron flux and their distribution through fuel cladding, channel and assembly during normal operation should be given. The pressures should include those which are internal and external to the fuel rod and pressure gradients in the axial and radial direction for the fuel assembly and channel. Temperature distributions should be also given for the top and bottom tie plate. For transients and accidents the above values should also be given as a function of time. If not available as a function of time, demonstrate that they do not exceed the stated design values.

4.16 For the cyclic loadings in Section 4.2.1.1.2.8 specify the time duration of each cyclic condition. Explain how the cyclic loading would be reflected in typical technical specifications.

4.17 Provide the numerical values used for the zircaloy cladding's yield strength and ultimate tensile strength mentioned in conjunction with the stress intensity limits specified in Section 4.2.1.1.2.4. In addition, state the cladding thermo-mechanical history and associated temperature and fast neutron flux (or fluence) for which the stress limits apply.

4.18 List fuel rod deflection and cladding strain limits and provide justification for their adequacy. For example, 0.060 inches is given for the rod to rod clearance and a 1% cladding strain limit is given.

4.19 What is the strain limit in the axial or radial direction? Since creep rupture strain depends on stress level, show how the stress effect is incorporated in the strain limit. Total strain is not defined in the strain calculations. If the strain limit was based on total strain, discuss how it is calculated.

4.20 Is there a natural frequency limitation on the fuel assembly? If so, is it relatable to primary system frequency?

4.21 Is there any stiffness limitation on the spacer grid assembly and individual grid spring?

4.22 What is the deflection limitation on the channel?

4.23 Give safety factors applied in the fatigue design, creep rupture, fatigure creep interaction, and instability (buckling) analyses.

4.24 With respect to fuel rod and assembly behavior, discuss the design limits for the accidents such as LOCA and seismic design.

4.25 Provide tables of material properties of both fuel rods and pellets as functions of temperature and irradiation. The properties should include modulus of elasticity, Poisson's ratio, thermal expansion coefficient, yield stress, ultimate stress, uniform ultimate strain, creep constants and creep equations.

4.26 Provide a more detailed description of the top guide and its connection with upper tie plate.

4.27 Describe how the channel is connected to the fuel assembly. What is the interference dimension between the reusable fuel channel and the fuel assembly and what are the expected interference loads?

4.28 Provide the dimensions and spring constant of the fuel stack holddown spring.

4.29 Provide the specific materials used in the spacer grid design.

4.30 Describe the grid lock on the water rod, and provide a drawing.

4.31 The analytical calculation of fuel-clad mechanical interaction that is used to satisfy the design bases should be given. A detailed complete description is needed, including a general description, assumptions, mathematical equations, sequence of application of equations or a flow chart, sample calculation (bench mark) and a comparison with test results.

4.32 In addition to the fuel-clad mechanical interaction analysis, describe the calculations used to justify:

1. Plenum volume and pressure calculation including plenum buckling stability and creepdown.

2. Local stress of cladding at fuel pellet interfaces, and relation to observed cladding brittle failures.

3. Cladding transient power stress-strain analysis.

4. Axial ratcheting of fuel-cladding.

4.33 Discuss how the different loading categories are combined to satisfy the design limit for each component of the fuel assembly.

4.34 Evaluate the behavior of the spacer grid in high temperature and oscillating fluid environments. Show that during fuel rod axial expansion and contraction no binding is expected due to bending of fuel rod and asymmetry of support in an 8 x 8 design. Evaluate the stresses in the spacer grid lock. Show that the spacer is adequate to support lateral loading during LOCA and Seismic lateral loadings.

4.35 Discuss the behavior of the bottom fuel rod support during cyclic thermal axial expansion and contraction of the fuel rods.

4.36 Evaluate the axial and radial thermal cycling problem of a fuel channel. Discuss the preoperation and normal operating loads on the channel, including stresses due to control rod interference and thermal differential expansion stress between stainless steel tie plate and Zr-4 channel.

4.37 Determine the thermal stress in the fuel rod caused by lateral thermal differential expansion between the tie plate and the spacer grid.

4.38 Evaluate the effects of fuel rod bowing together with spacer grid response including time dependent behavior due to creep.

4.39 Explain in detail how the fuel rod flow induced vibration is not a problem in an 8 x 8 design. Evaluate vibration of the shroud and other internals and their effects on fuel rod vibration.

4.40 Discuss the operating experience you have with the 8 x 8 fuel design, including how much experience you have with 95 TD UO_2, 9 mil gaps, water rods, no dish pellets, and annealed Zr-2 cladding of 34 mil wall thickness.

4.41 Describe in detail the experience, testing, and analyses, that provide an understanding of the failure mechanism of the fuel channel and measures taken to eliminate such failures.

4.42 Provide the percentage gadolinia per pellet, the number of gadolinia-poisoned rods per assembly, number of assemblies containing poisoned rods, total number of assemblies, axial distribution of the gadolinia in the rods, location of gadolinia-poisoned rods, and operating limits on the poisoned rods, i.e., linear powers at various stage and powers calculated to result in incipient melting.

4.43 Discuss the operating experience of BWR's with gadolinia-poisoned fuel, including reactor names, loading dates, fuel burnups and maximum powers.

4.44 Discuss the results of all post-irradiation examinations of irradiated gadolinia-poisoned rods to date including visual examinations, neutron radiography, dimensional measurements, fission gas analysis, ceramographic examinations, radial grind isotopic sampling, and microprobe analysis.

4.45 Thermophysical properties for the specific $GdO_{1.5}$-UO_2 solid solution composition to be used should be presented and compared to the pure UO_2 properties. Properties of interest include melting behavior, thermal expansion, specific heat, vapor pressure and creep behavior.

4.46 Quality assurance and inspection procedures used in the manufacture and loading of gadolinia-poisoned rods should be presented. Specifically, methods used to verify the gadolinia concentration in the UO_2-Gd_2O_3 blend, the sintered pellet UO_2-Gd_2O_3 solid-solution homogeneity, the gadolinia-urania pellet identification, and the gadolinia-urania fuel rod identification should be presented.