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November 21, 1977

Mr. Donald K. Davis, Acting Chief
Operating Reactors - Branch 2
Division of Operating Reactors
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Dresden Station Unit 2
Additional Information Requested
for Reload No. 3
NRC Docket No. 50-237

Reference (a): D. K. Davis letter to R. L. Bolger,
dated November 11, 1977.

Dear Mr. Davis:

Enclosed is the additional information requested by Reference (a). These responses have previously been hand-delivered or telecopied in segments as they became available.

If you have any additional questions concerning this matter, please contact this office.

One (1) signed original and 39 copies of this letter are provided for your use.

Very truly yours,

M.S. Turbak
M. S. Turbak
Nuclear Licensing Administrator
Boiling Water Reactors

Enclosure

QUESTION 1.

Specify the multiplying factors that have been applied to the nominal values of Doppler, void, and scram reactivity coefficients to obtain the values of these input parameters as given in Table 6-1 of NEDO-24034, Rev. 1, Supp. 4.

RESPONSE 1.

The multiplying factors that have been applied to the nominal values of Doppler, void, and scram reactivity coefficients are the following:

Parameter	Multiplier
Doppler	0.95
Void	1.25
Scram Reactivity	0.80

QUESTION 2.

The results provided for the fuel loading error have included only the case of a misoriented fresh 8D250 bundle. Specify the LHGR and MCPR that would result from the worst case misloading of a fresh 8D250 bundle into a 7x7 bundle site.

RESPONSE 2.

The misplacement of a fresh 8D250 bundle into a 7x7 bundle site results in a MCPR of 1.17 and a LHGR of 16.9 kw/ft.

QUESTION 3.

With regard to the control rod drop accident analysis:

(a) Provide the plant specific values of β and P_L and also provide any quantitative information to indicate how conservatism in these parameters might compensate for the cross over in scram reactivity curves beyond 3.75 seconds.

(b) By how much does the value of scram reactivity inserted at 3.75 seconds differ from 0.024 ΔK?

(c) Specify the largest margin by which the bounding scram reactivity curve exceeds the plant specific curve beyond 3.75 seconds.

(d) The description of the Banked Position Withdrawal Sequence in NEDO-21231 indicates that control rod drop accidents involving longer drops and larger worths than specified in NEDO-24034, Rev. 1, Supp. 4, are possible. Explain how it has been determined that the worst case rod drop for Dresden 2 Cycle 6 involves a drop from position 00 to 08 and insertion of 0.0024 ΔK as contrasted with the larger worth rod drops of NEDO-21231.

RESPONSE 3.

(a) The plant specific values for β and P_L are shown below:

	β	P_L
Cold	.0052	1.43
Hot Standby	.0064	1.26

It is the fact that the rod worths (cold are so low ($<$ .0025 ΔK), much less than the delayed neutron fraction (.0052) which must be exceeded for prompt criticality, that more than compensates for the slight cross over (cold only) in the scram reactivity curve.

(b) The value of the scram reactivity at 3.75 seconds is .0024 Δk $\pm$ .0001 Δk.

(c) The largest margin by which the bounding reactivity curve exceeds the plant specific curve (cold) beyond 3.75 seconds is .0043 Δk at 5.33 seconds.

(d) Rod drop worths were calculated for four cases:

Group at (Notches)	Rod Drop (Notches)
0	0 to 4
4	0 to 8
8	0 to 12
12	0 to 48

For the cold condition the group at four with the rod dropped from 0 to 8 notches was the most severe case.

QUESTION 4.

Based on NEDO 20360 it is expected that reload submittals will contain graphs showing instrument responses during rod withdrawal transients, including the effect of failed instruments. Either provide such graphical data, or provide a written description of how the effect of failed instruments was taken into account.

RESPONSE 4.

RBM Channel A (A + C Level detectors) and Channel B (B + D level detectors) instrument responses are determined with 0, 1, and 2 assumed string failures and are used to develop a minimum composite response versus position withdrawn for the error rod.

QUESTION 5.

NEDO-20954, the GE topical report on void and Doppler reactivity feedback, indicates that the most severe transients occur at EOC values of void coefficient and scram reactivity. Since you plan on derating the plant before EOC these parameters may have their most limiting values at some time before EOC.

For the purpose of establishing that the correct operating limit MCPR for 8x8 fuel has been identified, specify how you will assure that the actual values of the void and scram reactivities and transient ΔCPR at the time of derate and through EOC 6 will remain bounded by the values given in NEDO-24034, Rev. 1, Supp. 4. Take into account any uncertainties in predicting rod configuration, void distributions and burnup at the time of derate through EOC 6.

RESPONSE 5.

MCPR limits are established by a number of criteria and may or may not be set by transient analyses. However, calculations are performed at 100% power which yield more conservative results than calculations at derated conditions. The mode of burnup, choice of critical control rod configurations and the resulting void distribution are chosen to maximize transient response.

QUESTION 6.

Indicate your intentions for completion of the requested operational assurance tests as outlined in the appendices A and B.

RESPONSE 6.

Provided separately.

QUESTION 7.

Regarding the discussion of ASME vessel pressure code compliance in NEDO-24034, Rev. 1, Supp. 4:

(a) Explain the statement that the safety/relief valves are assumed to be inactive.

(b) Give the pressure relief capacity of the plant (in percent NBR steam flow of the plant) and also give the pressure relieving capacity assuming the most limiting safety valve fails to open.

(c) Confirm the applicability of the sensitivity study of peak vessel pressure to valve operability described in the December 13, 1975 letter from I. Stuart (GE) to V. Stello (NRC) to Dresden Unit 2.

RESPONSE 7.

(a) The statement that the relief valves are assumed to be inactive means that the four electromatic relief valves are assumed not to open for this event.

(b) For this event, the pressure relieving capacity for the plant is 57.9% of NBR steam flow. If the most limiting safety valve is out of service, the pressure relieving capacity would be 52.5% of NBR steam flow.

(c) A sensitivity study of valve operability is contained in the December 23, 1975 letter from I.F. Stuart, GE, to V. Stello, NRC, "Code Overpressure Protection Analysis - Sensitivity of Peak Vessels to Valve Operability". The sensitivity analysis transmitted by the above referenced letter was performed for a typical high power density BWR. This study is applicable to the Dresden 2 reactor and is supplemental to the specific analysis provided for the reload. A plant specific analysis for Dresden 2 would show results less than that given in the sensitivity study since Dresden 2 is a lower power density plant.

QUESTION 8.

For the ASME Vessel Pressure Code Compliance analysis, the initial operating pressure is assumed to be 1005 psig in the vessel dome. The present Dresden Unit 2 Technical Specifications do not contain a requirement which limits the operating reactor (dome) pressure to that which was assumed in the overpressurization analysis. Moreover, sensitivity studies have not been performed which show the effect of initial operating pressure on the peak transient pressure attained during this limiting overpressure event. Therefore, either: 1) provide a sensitivity study which shows that increasing the initial operating pressure (up to the maximum pressure permitted by the high pressure trip point, i.e., 1060 psig) will have a negligible effect on the peak transient pressure or, 2) propose Technical Specification changes which will assure that the reactor operating pressure will not exceed the initial pressure which was assumed in the Cycle 6 ASME vessel pressure code compliance analysis for Dresden Unit 2.

RESPONSE 8.

A sensitivity study of the affect of vessel dome pressure on the ASME vessel Pressure Code Compliance analysis was presented in the response to NRC questions on Quad Cities 1 Reload 3 (M.S. Turbak letter to D.K. Davis dated April 25, 1977). This analysis is applicable to Dresden 2.

QUESTION 9.

The Technical Specification ratio of the design total peaking factor to the maximum total peaking factor establishes both rod blocks and high flux setpoints. The maximum total peaking factor is calculated during the cycle by the process computer and compared to the design total peaking factor. Explain how changes in the relative numbers of 7x7 and 8x8 assemblies are accommodated in consistently normalizing the design total peaking factor and the maximum total peaking factor.

RESPONSE 9.

The change in the relative numbers of 7x7 and 8x8 assemblies are accommodated by calculating the peaking factors separately, determining the worst case rod block or scram setpoint between the two, and using this as the operating limit.

QUESTION 10.

Qualitatively explain why the incorporation of a different pressure rule assumption from the previous analyses results in a greater reflooding time increase for QC1, 2/D2, 3 compared to other BWR/3s (as stated on the bottom of Page 606) of LOCA submittal.

RESPONSE 10.

The lead plant, Quad Cities 1/2, Dresden 2/3, has a larger vessel-volume-to-break-size ratio than other BWR/3s. The result is that the lead plant has a slower depressurization rate for similar sized breaks in other plants and with the old pressure rule would have held up at a relatively high pressure. The new pressure rule assumption generally has a greater impact on LOCA transients with a slower depressurization rate (as discussed in the next response), hence it is likely to result in a greater increase in reflooding time for the lead plant than for other BWR/3s.

QUESTION 11.

Provide a qualitative explanation of the hot Node uncovery time versus break size (Figure 8 of LOCA submittal). The explanation should include a description of how the interrelation of the phenomena involved (new pressure rule, bypass area, etc.) combine for various break sizes to cause the calculated results. A thorough explanation on the lead plant (QC12/ D23) may obviate the need for extensive questions on each "non-lead" plant, some of which have limiting breaks smaller than the DBA. It is hoped that the requested explanation will allow acceptance of such results on those other plants (e.g. Monticello) without extensive explanation and/or further calculations.

RESPONSE 11.

The hot Node uncovered time versus break size (Figure 8 of LOCA submittal) has peaks at breaks smaller than the DBA for the following reasons:

1) The depressurization rate, in the new model as modified from the previous model, generally has a greater impact on the smaller breaks than on the larger breaks as the new method results in longer periods of steam generation due to flashing. The increased steam generation calculated then affects the amount of core spray flow to the lower plenum as determined by the counter current flow limiting characteristics of the core or the bypass regions.

2) At some break size smaller than the DBA and generally for all breaks smaller than that, the REFLOOD code uses the small break model (SBM) instead of the large break model (LBM). (The differences in the two models are discussed below.) As there are some differences in the two models, there appears to be an apparent discontinuity in the break spectrum analysis of these breaks. The break size at which this switch from the LBM to SBM occurs is determined by a combination of interrelated factors such as "effective break size (i.e., ratio of vessel volume to break size), depressurization rate, and time at which ECCS flow into the pressure vessel is initiated. As a result of the switch to different models at different break sizes for the various plants and different sensitivities to various parameters for each plants, slight variations in the shape of the break spectrum curve should be expected.

For the lead plant, the break at which the SBM is used occurs around 60% of the DBA and then shifts back to the LBM. A second apparent discontinuity occurs at about 40% of the DBA where again a switch is made to the SBM. The switch in models is determined by a combination of factors as discussed above. Though the breaks at about 40% and 60% of the DBA have longer uncovered times for the hot mode than the DBA, the DBA was determined to be the most limiting break as discussed in Reference 1.

Difference between REFLOOD Small and Large Break Models

The REFLOOD code automatically uses the small break model for any transient for which there is a water level in the active core region, when the calculation switches from the SAFE code to the REFLOOD code.

The two most significant differences between the small and large break models are:

a) Use of the Vaporization Correlation: The vaporization of spray water in the core during the period when core sprays are operating is calculated using a bounding correlation. The correlation, as discussed in Reference 2, requires the PCT at time of spray initiation. The LBM correctly uses a constant value whereas the SBM conservatively uses a continuously increasing value. This difference generally results in a more conservative calculation of the reflooding time using the small break model.

b) Level and Vaporization Following Bottom Reflooding: The LBM uses an empirically based void fraction of 0.50 for calculating the level and the vaporization below the level. The SBM uses the conservative fuel rod heatup model with a reflooding heat transfer coefficient to calculate the level and the vaporization below the level. This difference generally results in a more conservative calculation of the reflooding time using the SBM.

Reference 1: "LOCA Accident Analysis Report for Dresden 2,3 and Quad Cities 1, 2 Nuclear Power Stations (Lead Plant)," NEDO-24046, August 1977.

Reference 2: General Electric Company "Analytical Model for Loss-of Coolant Analysis in Accordance with 10CFR50 Appendix K" NEDO-20566, Vol. II.

QUESTION 6 (Appendix A)

The following test descriptions address the six specific areas discussed in Appendix A, Item 1. In addition, the usual program for reload startup testing is planned as described in response to previous NRC reload questions (e.g. D3R4 and Q1R3).

Additional tests scheduled for D2R3 BOC6 include:

Core Loading Verification

Moderator Temperature Coefficient

SRM and IRM Performance Checks

ARPM and LPRM Calibrations

Core Performance Evaluations
(MCPR, LHGR, MAPLHGR, TPF)

Core Flow Calibration and Recirc. System Baseline Data Acquisition.

TEST A: Control Rod Drive Tests and Scram Time

1) Scram Timing Test Description:

a) After each refueling outage and prior to power operation with reactor pressure above 800 psig, all control rods shall be subject to scram-time tests from the fully withdrawn position. The scram times shall be measured without reliance to the control rod drive pumps.

b) At 16 week intervals, 50% of the control rod drives shall be tested as in 4.3.C.I so that every 32 weeks all of the control rods shall have been tested. Whenever 50% of the control rod drives have been scram tested, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

Criteria:

a) The average scram insertion time, based on the de-energization of the scram pilot valve solenoid as time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

% Inserted Time Fully Withdrawn	Average Scram Insertion Times (sec)
5	0.375
20	0.900
50	2.00
90	3.50

The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.12
90	3.80

b) The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

2) CRD Overtravel and Timing Test:

Description: The prupose of this is to check for overtravel and to check the actual time it takes to withdraw and insert the drives.

Criteria: The rod is considered not to overtravel if when given a continuous withdraw signal at position 48, the indication does not drift, and the digits 4 and 8 remain lighted in the RPIS display.

The rod timing must be calibrated at the beginning of the cycle such that the time to travel between 0 and 48 (or 48 and 0) is 48 seconds ± 10%.

TEST B: Verification of Shutdown Margin

Description: The purpose of this test is to demonstrate that the reactor will be subcritical to its most reactive condition during the ensuing cycle with the strongest operable control rod in the full withdrawn position with all other operable control rods fully inserted. This may be demonstrated by withdrawing via a special control rod and one or two more nearby control rods to predetermined positions, and/or by performing an insequence method S.D.M. Control rod worth information obtained from General Electric can then be directly used to determine how much reactivity was inserted by the withdrawal of the second, or second and third, control rods.

Criteria: A shutdown margin of R + .25% ΔK must be demonstrated with the strongest rod fully withdrawn. R is assumed to include any increase possible in core reactivity during the cycle from the time of shutdown margin calculation. For Dresden 2 Cycle 6, R is .02% ΔK to account for inverted control blade poison tubes.

If shutdown margin cannot be demonstrated, the reactor will be shutdown, the NRC will be notified and the core loading will be altered as necessary.

TEST C: Critical Eigenvalve Comparison

Description: This test is to compare the actual cold critical control rod pattern with the predicted critical rod pattern obtained from General Electric.

Criteria: The actual cold critical rod pattern should be within 1% ΔK of the predicted control rod pattern. If the difference is greater than $\pm$1% ΔK, Core Management Engineers will also be promptly contacted to explain the anomaly.

TEST D: Power Distribution Comparison at a Given Control Rod Pattern and Power Level (Reactivity Anomaly).

Description: During the startup test program the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every equivalent full power month.

Criteria: The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% ΔK. If this limit is exceeded, the reactor will be shutdown until the cause has been determined and corrective actions have been taken. In accordance with Specification 6.6, the NRC shall be notified of this reportable occurrence within 24 hours.

TEST E: TIP Reproducibility Test and Core Power Symmetry Tests
+
TEST F: Have Been Combined

Description: The purposes of these tests are to:

a) Confirm the reproducibility of the TIP system readings; and

b) To determine core power symmetry.

Core power distributions will be calculated during the power ascension program using complete sets of axial power traces obtained from the Traversing In-Core Probe (TIP) System. At the intermediate and higher power levels, TIP data will be obtained to determine the overall TIP uncertainty.

TIP data will be obtained with the reactor operating with a symmetric rod pattern and at steady state conditions. The total TIP uncertainty for the test will be calculated by averaging the components of axial TIP uncertainty, which are made up of random noise and geometric components.

Four TIP traces of the same channel on each machine should be obtained at a steady state power level > 50% RTP to calculate TIP reproducability uncertainties.

Criteria:

1) The total TIP uncertainty (including random noise and geometric uncertainties obtained by averaging the uncertainties for all data sets) must be less than 12%.

NOTE: A minimum of two and up to six data sets may be used to meet the above criteria. If the 12% total TIP uncertainty criteria cannot be met by the six sets of data, testing may continue provided the MCPR limit is adjusted to reflect the TIP uncertainty and this change is reported to the NRC.

Additional data sets may be obtained in order to improve the TIP data base, and the MCPR limit adjusted accordingly. If the 12% total TIP uncertainty becomes satisfied, the MCPR limit can be returned to its original value.

2) TIP reproducibility should be within ± 10% relative error and a maximum absolute deviation should be no greater than 8% of full scale (y axis).

ANSWER: 2

A summary report of all physics tests performed as BOC startup tests are generally available at Dresden approximately 90 days following the completion of the startup test program.

Since no startup test program is required by the

Technical Specification this cycle, as per 6.6.A.1 (below), the previously mentioned startup tests and completion date seem entirely adequate.

" Startup Report

A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant."

The core loading information requested (Question 2), including various core maps, bank designations, loading sequences, and shutdown margin verification will be included in the Startup Test Report.

ANSWER: 3

The Startup Test Report will summarize all nuclear physics related calibrations performed for BOC6. These may include, but will not be limited to Core Flow Calibration, TIP Machine Calibration, and Initial and Initial LPRM Calibration, etc.

ANSWER: 4

No specific tests and inspections, as mentioned in Question 4, are planned since the relead fuel is identical to fuel loaded in two previous cycles (8x8, 2.50 WT%), and no test assemblies or rods have been added to the core.

JAS:rap

Additional Information Concerning
Question 6 Test D

D. Core Power Distribution Comparison at a given Rod Pattern and Power Level.

Description:

The following core performance parameters should be checked at several power levels for reasonableness of data:

1. Minimum Critical Power Ratio
2. Maximum Average Planar L.H.G.R.
3. Total Peaking Factor
4. Maximum L.H.G.R.

These parameters shall be checked for compliance with their respective Tech. Spec. Limits.

In addition, at a constant control rod pattern, the following should be verified using an offline simulator code, (such as N.F.S.'s TRIBIG), or by using some other independent method such as T.B.A.R. (which is based on offline simulator runs.):

1. Radial Power Factors (Bundle Powers)
2. A.P.L.H.G.R.
3. L.H.G.R.

Criteria:

The above limits should be compared in several core locations which are representative of different fuel and cell types in the core. Any significant discrepancies should be investigated and resolved.