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Mr. George E. Lear, Chief
Operating Reactors - Branch 3
Division of Operating Reactors
U.S. Nuclear Regulatory Commission
Washington, DC 20555

REGULATORY DOCKET FILE COPY

Subject: Dresden Station Unit 3
Request for Additional Information
Concerning Reload 5 Cycle 6
NRC Docket No. 50-249

Reference (a): R. B. Bevan telecopy to M. S. Turbak
on February 24, 1978

Dear Mr. Lear:

Enclosed is Commonwealth Edison's response to the NRC Staff questions concerning Dresden Unit 3 Reload 5 Cycle 6 transmitted by Reference (a).

Please contact this office if you have any questions concerning this matter.

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

Very truly yours,

R. J. Naughton

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

Enclosure

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DRESDEN 3/RELOAD 5 LICENSING RESPONSES

1. The Doppler, void, and scram reactivity coefficients given in Table 6-1 of NEDO-24074 contain the following multiplying factors:

Doppler	0.95
Void	1.25
Scram reactivity	0.80

2. The results of the fuel loading error reported in NEDO-24074 are the result of misloading a fresh 8 x 8 reload 5 (8D262) bundle into an exposed 8 x 8 fuel bundle location. The misloading of a fresh 8 x 8 reload 5 bundle (8D262) into a 7 x 7 location results in a LHGR of 18.95 kw/ft and a ΔCPR of 0.15. For the worst case misoriented bundle, the resulting LHGR is 15.87 kw/ft and the ΔCPR is 0.17.

3. a) The pressure relief capacity of the plant for the ASME Pressure Code Compliance case is 58.3% of nuclear boiler rated steam flow. The relief capacity for this case with the limiting safety valve inoperable is 52.5% nuclear boiler rated steam flow.

   b) The applicability of the sensitivity study of peak vessel pressure to valve operability, described in the December 23, 1975, letter from I. Stuart (GE) to V. Stello (NRC), was confirmed for Dresden 3 in the G.A. Abrell (CECo.) letter to D. L. Ziemann (NRC), dated October 19, 1976, for Reload 4, and is reconfirmed for Reload 5.

   c) The sensitivity study on the effect of initial operating pressure on peak transient vessel pressure described in the April 25, 1977, letter from M. Turbak (CECo.) to D. Davis (NRC) is applicable to Dresden 3. The referenced analysis was performed for Quad Cities Station reactors and since the pressure relieving capabilities, setpoints, etc. are the same for Dresden, the analysis results are applicable to Dresden 3.

4. A summary of the nuclear physics startup tests performed for Dresden Unit 3 Reload 5 Cycle 6 as described in the November 21, 1977, letter from M. S. Turbak (CECo.) to D. K. Davis (NRC) will be submitted to the NRC 45 days following completion of the startup test program.

5. a) Procedures are implemented at the site to assure that the plant process computer constants are checked for compliance with the as-loaded core. The process computer input data is not transmitted to CECo. until after the defective fuel bundles are identified, the core is reloaded and the core verification is complete. Therefore, the process computer update received by CECo. reflects the as-loaded core and will have been modified by the vendor to reflect any changes in core loading. In the unlikely event that a core loading change occurs subsequent to receiving the computer update from the vendor, an On-Site Review of necessary modifications will be performed.

b) If the as-loaded core differs from the referenced core upon which the licensing analysis was performed, the fuel vendor will be informed. The licensing analysis will then be reviewed by the vendor to determine if additional calculations are necessary to assure that the safety analysis will be applicable to the as-loaded core. With the aid of the fuel vendor, the fuel assembly (assemblies) replacing the damaged assembly (assemblies) are chosen to match as closely as possible the nuclear characteristics of the damaged bundle(s). If the nuclear characteristics of the damaged and replacement bundles are in close agreement, it is unlikely that new licensing calculations would be necessary. If the agreement is not acceptable, some parameters or transients would require reanalysis (shutdown margin, rod drop accident, or rod withdrawal error for example) for the modified core. If needed, CECo. will obtain (and On-Site review prior to startup) additional calculations from the vendor to ensure that the licensing analysis is applicable to the as-loaded core. Differences which result in changes to the Technical Specifications, License, or unreviewed safety questions will be reported as required by the Technical Specifications.

c) If the as-loaded core differs from the referenced core upon which the shutdown margin calculation was based, the vendor will determine if additional rod worth calculations are required. If needed, CECo. will obtain the additional calculations and rod worth information from the vendor prior to startup.

6. Data constants to be loaded into the process computer for the upcoming cycle are compared to the information loaded into the process computer for the previous cycle. Any differences in the data constants are noted and evaluated to assure that the data constants are correct for the upcoming cycle. The data is processed through internal vendor programs that perform limit, consistency, and value checks. To further analyze the consistency of the data, the vendor will compare estimated full power BOC off-line simulator cases with process computer cases for gross data deficiencies.

   After the process computer information is verified as described above, the data is transmitted to the CECo. general office where an independent verification of selected data constants is performed. The process computer update is checked to reflect the installation of new LPRMS, installation of new control rods, installation of new fuel and shuffle of old fuel, so that it complies with the as-loaded core. The process computer update is also checked for the accuracy of bundle flow areas, MCPR correlation constants, and power distribution correlation constants. Many other constants are checked as well.

The station nuclear engineers verify that the various thermal limits, such as MCPR and LHGR, are correctly calculated during startup by performing hand calculations using the process computer equations and constants or by comparison with off-line 3D simulator results (see response 9). The calculations used to verify the accuracy of the process computer are available at the site for review.

After the process computer data is loaded, a computer memory dump is obtained. This data is transmitted to the fuel vendor for their re-evaluation and concurrence to assure that the process computer data has been loaded correctly.

7. The General Electric Thermal Analysis Basis-GETAB (Ref. 1) was approved by the NRC in 1974 (Ref. 2). The use of the single set of uncertainties in determining the safety limit MCPR for all reload cores was a part of this approval.

   It is General Electric's opinion that this issue was resolved with the issuance of the GETAB SER (Ref. 2). A requirement for plant specific uncertainty criteria represents a change in the licensing basis. If a change in the licensing basis is to be made, it should be in accordance with the procedure specified in the Code of Federal Regulations. General Electric has advised the NRC of this opinion and they will formally document it within the next few months.

   If the NRC does not concur with this position, CECo. will apply the recommended TIP uncertainty penalty as an interim measure until the issue is resolved. Specifically, if the standard deviation of the total TIP uncertainty plus LPRM calculational uncertainties exceeds 8.7% for the upcoming Dresden 3 Cycle 6 the plant operating MCPR limit will be adjusted to account for this additional uncertainty. CECo. will use a 0.01 MCPR increase per 2% increase above 8.7%. If an increase in the operating limit is required, the Startup Test Report will indicate the administrative action taken.

   It should also be noted that the combined geometric and noise TIP uncertainty for a reload core is 7% and not the 6.5% stated by the NRC. Combining all other uncertainties yields a total of 8.7%.

REFERENCES: (1) General Electric Thermal Anaylsis Basis Data, Correlation and Design Application, NEDE-10958, November 1973.

(2) Review and Evaluation of GETAB for BWR's, Technical Review, Directorate of Licensing, USAEC, September 1974.

8. Using estimation techniques, a number of candidates are identified for the strongest rod at beginning of cycle. Using a 3D Simulator Code, the shutdown margin (SDM) is determined with each of these rods fully withdrawn. Those calculations are repeated for middle and end of cycle exposures. The method used for the calculation of the incremental worths is the 3D Simulator Code. Calibration of the incremental worths is accomplished by benchmarking the code against previous cycle criticals.

The margin is evaluated at the site by applying the supplied incremental worth data to the actual rod configuration. Temperature and period corrections are also included.

There are several methods available for demonstrating adequate shutdown margin. The most commonly used methods include:

1. Local subcritical tests involve withdrawal of a few control rods to a predetermined configuration which is calculated to demonstrate at least .25% ΔK + R (typically .50 to 1.5% ΔK is demonstrated). Subcritical tests may utilize either
   a. face-adjacent control rods (~ 2) or
   b. diagonally-adjacent control rods (~ 2 or 3).

2. Local critical tests involve withdrawal of a few rods until criticality is attained with a manageable period. Positive and negative period measurements are also included to check calculated notch worths. Local criticals may utilize either
   a. face-adjacent control rods (~ 2) or
   b. diagonally-adjacent control rods (~ 2 to 5).

3. Dispersed critical tests use in-sequence withdrawal of many control rods (~ 30 to 55).

An obvious advantage of the first type of test (local subcriticals) is the lower probability of an unanticipated criticality or short period. Although this method does not quantify the actual maximum shutdown margin available, it does provide assurance that the Tech. Spec. SDM requirement has been met prior to startup.

Determining the magnitude of the total shutdown margin is possible by utilizing either the second type (local criticals) or the third type (dispersed criticals) of testing. As a result of the high rod worths involved in the local critical methods, special procedural precautions are necessary including the "pumping technique" of cycling the strongest rod back in prior to withdrawal of each additional notch of the second (or third) rod.

Although local criticals are no longer performed routinely, they may be warranted in order to obtain data which is sensitive to a specific area of the core (such as benchmarking new fuel types or quantifying stuck rod margin).

The third method minimizes the rod worths even further by utilizing in-sequence, dispersed withdrawal. This method, however, requires considerably more pre-calculated rod worth data.

Since no new fuel types or unusual loading strategies are utilized for Dresden 3 Reload 5, the local critical tests will not be performed at BOC. Diagonally adjacent subcritical tests will be performed, however, for demonstration of Tech. Spec. compliance prior to startup. During the D3 R5 initial startup the dispersed method may also be used to further quantify the magnitude of the SDM available.

The applicable Dresden Station procedure is DTS 8134 "Units 2/3 Shutdown Margin Demonstration." It should be noted that this procedure is currently being revised to reflect earlier NRC commitments but will be reviewed and approved prior to Dresden 3 Cycle 6 testing.

Normally, the calculated shutdown margin is significantly greater than the required shutdown margin. Since the worth of the rod is insensitive to the degree of code bias (critical offset) there has previously been no requirement to evaluate the actual versus predicted rod pattern to determine the adequacy of the shutdown margin demonstration. Table 1 summarizes the shutdown margin demonstrations performed on recent startups of the large Dresden and Quad-Cities units. It should be noted that this Table is the same as that supplied for the Q2R3C4 question with the inclusion of temperature corrections. The shutdown margin demonstrations were well in excess of the required margin.

Should a condition arise during the shutdown margin demonstration when the demonstrated margin is evaluated to be less than the Technical Specification limit or when the adequacy of the shutdown margin demonstration is in question, the unit will be shutdown and the condition evaluated. Corrective action for an unacceptable margin will most likely include shuffling the core. A Licensee Event Report and additional documentation for the new core loading would be submitted to the NRC for their approval prior to startup.

9. A comparison of power distributions calculated by the process computer and an off-line simulator can be used to identify any power distribution differences in the D3 C6 core. Spatial differences can be identified by comparing the radial power factors for the planar dimension and by comparing the maximum average planar LHGR (MAPLHGR) and LHGR for the axial direction.

The data obtained from the two methods will be compared, at a minimum, for the two most limiting process computer APLHGR bundles of each reload (these are usually centrally located bundles). In addition, the radial power factors and LHGRs will be compared. The peak process computer APLHGR will be compared to the off-line simulator APLHGR at the same elevation for the bundles. A separate axial location within these bundles will be compared also. Since the reactor will be quarter core symmetric during this comparison, it is sufficient to obtain the data from only one quadrant in the core. Discrepancies beyond the expected range will be investigated and resolved.

10. An estimated critical rod pattern using in-sequence control rod withdrawals will be determined. Selected group and individual rod worths are also determined as part of the analysis. The core averaged moderator temperature coefficient is used to estimate any change in the estimated critical rod pattern caused by a difference between the analyzed and actual core temperature conditions. The actual critical control rod pattern is then evaluated against the estimated critical control rod pattern with period and temperature corrections included.

If the actual rod pattern indicates that the core is greater than 1% ΔK more reactive than predicted, the discrepancy will be immediately investigated by the organization responsible for fuel management and an On-Site Review will be conducted.

If the actual rod pattern indicates that the core is less reactive than predicted, the site nuclear engineers and core management engineers will evaluate the need for additional investigation based on the magnitude of the discrepancy and engineering judgement.

TABLE 1

Shutdown Margin Summary

UNIT	CYCLE	SUBCRITICAL POSITION DEMONSTRATED			SUBCRITICAL POSITION NEEDED TO DEMONSTRATE MARGIN (0.25% Δk + R)	CALCULATED VALUE OF (0.25% Δk + R)	MARGIN ACTUALLY DEMONSTRATED		MARGIN DEMONSTRATED in excess of (0.25% Δk + R)
		STRONGEST ROD	ADJACENT ROD	POSITION	POSITION	(Δk)	Non temperature corrected	Temperature corrected	(Δ K)
D2	C6	D-8		48	48	.0027	.0120	.0092	.0065
			E-7	24	6				
D3	C5	F-13		48	48	.0046	.0146	.0141	.0095
			G-12	48	**				
Q1*	C4	H-8		48	48	.0058	.0088	.0080	.0022
			H-9	8	8				
Q2*	C3	H-5		48	48	.0030	.0270	.0266	.0236
			H-6	48	***				

* After demonstrating at least .25% Δk + R, the Quad Cities units were made critical by further rod withdrawal in order to quantify the max. SDM available at BOC. The Dresden units were not withdrawn past the points indicated.

** Total worth of G-12 was calculated to be 1.46% Δk by the vendor.

*** The total worth of H-6 was calculated to be 2.70% Δk by the vendor. Partial rod worths were supplied for the 3rd rod.