Nebraska Public Power District

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March 22, 1979

Director, Nuclear Reactor Regulation
Attention: Mr. Thomas A. Ippolito, Chief
Operating Reactors Branch No. 3
Division of Operating Reactors
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Reload 4, Cycle 5 Reload Licensing Submittal - Request for Additional Information
Cooper Nuclear Station
NRC Docket No. 50-298, DPR-46

Dear Mr. Ippolito:

Your letter of March 13, 1979, transmitted a request for additional information relating to Nebraska Public Power District's January 31, 1979 Reload License Application. Enclosed please find the District's response to these four questions. These questions were informally received from the Staff February 23, 1979.

As stated in our application, the current refueling schedule still allows for plant startup on April 27, 1979; therefore, approval of the reload submittal is respectfully requested prior to that date.

Should you require additional information, please do not hesitate to contact me.

Sincerely yours,

Jay M. Pilant
Director of Licensing and
Quality Assurance

JMP/jw:srs22/3
Enclosure

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STATE OF NEBRASKA)
)
PLATTE COUNTY)

Jay M. Pilant, being first duly sworn, deposes and says that he is an authorized representative of the Nebraska Public Power District, a public corporation and political subdivision of the State of Nebraska; that he is duly authorized to submit this information on behalf of Nebraska Public Power District; and that the statements in said application are true to the best of his knowledge and belief.

Jay M. Pilant

Jay M. Pilant

Subscribed in my presence and sworn to before me this 22nd day of March, 1979.

Marilyn R. Hohndorf

NOTARY PUBLIC

My Commission expires October 14, 1980.

GENERAL NOTARY - State of Nebraska
MARILYN R. HOHNDORF
My Comm. Exp. Oct. 14, 1980

REQUEST FOR ADDITIONAL INFORMATION
COOPER NUCLEAR STATION UNIT 1, RELOAD 4

Question 1

The staff stated in Section 6.2.2 of its safety evaluation of the Generic Reload Fuel Application (which you have referenced in your reload application) that "Additional Data should be submitted by GE to the staff for review, to justify the conservatism of the GEXL correlation for the second and subsequent cycles of operation of the retrofit 8x8 bundles, when local peaking factors may increase sufficiently to cause non-conservative CPR calculations." Your reload submittal has not addressed this issue. Accordingly, we request you provide either directly or through reference, adequate information which speaks to this concern. Your response should include:

The extent to which individual heater rods are instrumented in steady-state critical power tests for the retrofit fuel design.

For each test bundle provide measured and predicted results in tabular form for the various test conditions.

Provide trend plots (measured critical power/predicted critical power vs. h_{IN}, G, P, critical power, test bundle).

Maximum R-Factor for each test bundle (new and old R-Factor definitions).

Thermocouple locations (rod-by-rod and axially).

Spacer-grid locations.

Provide power and heat flux for all plots of transient CPR cases.

Response

General Electric Company will respond to this request on a generic basis. This response will be contained in a letter, R. E. Engel (GE) to Robert L. Tedesco and Darrel G. Eisenhut (NRC), "Additional Information, 8x8R Fuel GETAB R-Factors", to be issued 3-31-79.

Question 2

The staff stated in its evaluation of the Generic Reload Fuel Application that it is acceptable to reanalyze only a relative few limiting transient events as part of reload safety analysis. The basis for the selected events stems in part from previous (e.g. FSAR) analysis results (consequences). Furthermore, the relative consequences of all of the anticipated transient events considered (in the FSAR) is based on specific equipment performance characteristics and reactor protection system characteristics. It is not known whether the proposed reduction in the low pressure main steam line isolation valve setpoint (from 850 psig to 825 psig) will cause the currently non-limiting pressure regulator failure event to become a limiting event. We require that you provide sufficient information that shows the pressure regulator failure event remains non-limiting, even with the proposed setpoint change. An acceptable response would be to provide the transient ΔCPR (for each fuel type) for a pressure regulator failure transient analysis which models the proposed technical specification change.

Response

The proposed reduction in the main steam line low pressure isolation setpoint from ≥850 psig to ≥825 psig will not increase the severity of the pressure regulator failure opening event. This transient is initiated by a sudden reduction in pressure demand signal. This signal causes the turbine control and bypass valves to begin opening resulting in a rapid drop in core pressure. This pressure decrease results in increased void formation. The rapid insertion of negative reactivity due to the increased void content results in a rapid power decrease. The rapid pressure/power decay is terminated by the closure of the main steam isolation valves. Lowering the setpoint will actually result in a larger power decay before the transient is terminated, the resulting peak heat flux will be lower and the CPR response of this transient will actually be improved (reference SAR Figure XIV-5-8).

Question 3

Provide the ΔCPR result for a fuel loading error consisting of mislocating an 8x8R bundle in a 7x7 cell.

Response

The NRC has accepted the reload submittal format contained in NEDO-24011A, Rev. 0 which provides for reporting only the most limiting bundle loading error. In the event the NRC imposes new criteria which will make the mislocated 8x8R bundle limiting, the District will provide the results of the analysis of this event.

Question 4

It is the staff's position that adequate start-up physics testing be performed following each plant refueling in order to assure that the core conforms to the design, i.e. that the actual (measured) reload core configuration is consistant with the analyzed reload core configuration. The staff currently has a study underway for the purpose of generically establishing requirements for minimum BWR start-up physics test programs. Although this effort is not yet complete, we have concluded at this juncture that, in order to be acceptable, BWR start-up test programs, must include each of the following (or acceptable equivalents):

A. A visual inspection of the core including a photographic or videotape record.

B. A check of core power symmetry-by checking for mismatches between symmetric detectors.

C. Withdrawal and insertion of each control rod to check for criticality and mobility.

D. A comparison of predicted and measured critical insequence rod pattern for nonvoided conditions.

In view of the importance the staff places on the above four BWR start-up physics program elements, we request that you provide a commitment to include them (or acceptable equivalents) in the Cooper Station Unit 1, Reload 4 start-up program.

Additionally, in order that we may adequately assess the characteristics of the Cooper Station Unit 1 Cycle 5 start-up test program, we request that you provide the following information:

A description of the core loading verification (inspection) procedures to be followed for the core refueling including the number of independent checks to be made of a) the actual core loading, b) the intended core loading and c) the consistency between the two.

A description of each start-up physics test (including those indicated above).

The acceptance criteria and basis for each test (including those indicated above) which provides assurance that the actual core conforms to the design.

The actions to be taken for each test (including those indicated above) whenever the acceptance criteria are not satisfied.

Response

Nebraska Public Power District will perform the four start-up physics tests described in Attachment II in the Cooper Nuclear Station Reload 4 start-up program.

ATTACHMENT II

COOPER NUCLEAR STATION

RELOAD 4, CYCLE 5

START-UP PHYSICS TEST PROGRAM

CORE LOADING VERIFICATION

I. PURPOSE

The core loading verification is performed after the core is fully loaded to assure that the core is constituted as per the design loading plan.

II. DESCRIPTION

The core loading verification consists of the following three distinct phases:

A. A scan of the core with a television camera is made by operations personnel located on the fuel handling bridge. The fuel assembly serial number and orientation is observed and communicated to the Control Room via phone. The Control Room verifies the correctness of the information (or asks for a reread). The person located in the Control Room has a computer generated map of fuel assembly serial numbers versus location. The personnel on the fuel handling bridge must correctly identify the serial number and orientation before moving to the next fuel assembly. The person located in the Control Room will only verify correctness (or ask for a reread); he does not communicate the serial number to the personnel on the fuel handling bridge.

B. Concurrently with the above verification a video tape is made and the appropriate blocks of a blank core map are filled in with the fuel assembly serial numbers and orientations by a representative of the Reactor Engineering Staff. The Reactor Engineering representative is viewing the verification on a monitor located remotely from that used by the personnel on the fuel handling bridge and is also in continuous communication with both the fuel handling bridge and the Control Room. He must agree with the fuel assembly serial number and orientation as identified by the personnel on the fuel handling bridge or he calls for a reread of any fuel assembly in question. He also marks the beginning of each row scanned with voice identification on the video tape for subsequent review. The core map produced during the video taping is then checked against a computer generated core map of the design loading for correctness.

C. After the whole core has been scanned and taped, the video tapes are reviewed by a three man team (different personnel than participated in the above two phases) as a final check. Team member one is responsible for operating the tape recorder and observing fuel assembly serial numbers; member one also serves as a quality control check for member two's activities. Team member two views the tape and calls out the fuel assembly serial numbers and orientation in sequence. Team member three writes down the called fuel assembly serial numbers in the appropriate block of a blank core map. Member three also indicates on the core map that the orientation is correct after member two verifies that the spring clip location is acceptable (orientation as it should be). After each row is reviewed and the fuel assembly serial numbers entered on the map, member three reads back the recorded fuel assembly serial numbers and member two verifies these numbers against a computer generated core map of the design loading.

III. ACCEPTANCE CRITERIA

The core must be loaded according to the design loading pattern (or a fuel vendor approved variation thereof to accommodate discharged leaking fuel assemblies).

IV. ACTIONS TO BE TAKEN IF CRITERIA NOT MET

The core will be rearranged to conform to the design loading pattern and the core loading verification will be redone for the affected areas of the core.

CONTROL ROD CRITICALITY AND MOBILITY CHECK

I. PURPOSE

This test is performed to assure that all fuel assemblies are properly loaded and that all control rods are operable.

II. DESCRIPTION

This test will be performed after the core loading verification has been completed; this will provide assurance that an inadvertent criticality will not occur due to fuel assemblies being mislocated (it is very unlikely that a fuel loading error could result in a situation where criticality could be achieved with a single control rod being withdrawn). Performance of this test will assure that there are no fuel assemblies loaded so as to affect the movement of a control rod. If movement of a control rod is affected, it could be caused by such things as a rotated fuel assembly, a fuel assembly not being properly seated, etc. Each control rod in the core will be withdrawn and inserted to assure that it can be moved with normal drive pressure. Also, the nuclear instrumentation will be monitored during the movement of each control rod to verify subcriticality.

III. ACCEPTANCE CRITERIA

Each control rod must be exercised (fully withdrawn and inserted) one at a time to verify mobility under normal drive pressure (normal drive pressure is plant dependent); subcriticality will also be verified.

IV. ACTIONS TO BE TAKEN IF CRITERIA NOT MET

For those control rods that will not move under normal drive pressure, appropriate repairs or adjustments will be made so that the normal drive pressure criterion can be met. If criticality was to be achieved by the withdrawal of a single control rod, the control rod would be inserted and all further startup activities would cease; the fuel loading verification would be redone and no further rod movements would occur until the situation was satisfactorily resolved.

INITIAL CRITICALITY PREDICTION

I. PURPOSE

This test is performed to compare the predicted and measured insequence critical rod pattern for nonvoided conditions.

II. DESCRIPTION

The initial critical control rod pattern in the cold Xenon free condition will be predicted at the beginning of each fuel cycle using data supplied by the fuel vendor. Control rod worths will be determined from rod worth curves provided for the shutdown margin demonstration. The reactor will be brought critical utilizing the control rod sequence determined by the fuel vendor such that the withdrawal of the analytically strongest control rod will occur before criticality is attained. A moderator temperature correction (supplied by the fuel vendor) will be applied if criticality occurs at greater than 68° F. The actual and predicted critical rod patterns will be compared.

III. ACCEPTANCE CRITERIA

Acceptance criteria of +0.5% Δk/k and -1.5% will be applied. The +0.5% Δk/k criterion is taken from the section of the Technical Specifications describing reportable occurrences (Reactivity Anomalies). The -1.5% Δk/k criterion is arbitrarily set so as to be consistent with the 2% Δk/k window allowed for the steady state power reactive anomaly check (±1% Δk/k).

IV. ACTIONS TO BE TAKEN IF CRITERIA NOT MET

If criticality occurs 0.5% Δk/k before the predicted critical position, actions will be taken to determine the cause and the event will be reported as per Technical Specification requirements. If criticality occurs more than 1.5% Δk/k after the predicted critical position, actions will be taken to determine the cause of the event. In both cases, this action would probably consist of requesting the fuel vendor to recheck the rod worth curves provided and if necessary, have the fuel vendor renormalize his core simulator calculations so that more accurate rod worth curves can be provided. Startup would continue and all thermal limits would be closely monitored to assure compliance with all Technical Specification Limiting Conditions of Operation.

TIP SYMMETRY CHECK

I. PURPOSE

This test is performed to measure the total TIP symmetry and to compare the readings of individual symmetric pairs of TIP's.

II. DESCRIPTION

Typically, during a refueling outage, fuel is both removed and shuffled to accommodate the core loading plan for the subsequent cycle. Also, local power range monitors (LPRM's) are replaced as necessary due to their having reached end of life conditions. This test determines the magnitude of the TIP system and neutron flux asymmetrics after the core has been reconstituted according to the loading plan. The reactor will be above 75% thermal power and operating with an octant symmetric rod pattern for the performance of this test. Additionally, the reactor core should be at steady-state conditions essentially free of Xenon transients and expected to remain at steady-state conditions for the duration of the testing. Data will be taken utilizing the TIP system hardware and statistically analyzed using methods provided by the fuel vendor.

III. ACCEPTANCE CRITERIA

A criterion of 12% will be applied to the total TIP reading uncertainty; this 12% criterion translates into a criterion of 10.55% for total TIP instrument uncertainty (random noise and geometric components). These values were determined utilizing General Electric developed methods for determining the total process computer uncertainty (NEDO-20340). A criterion of 23% will be applied to the individual symmetric pairs; this was determined by multiplying 6.5% by $\sqrt{2}$ by 2.5 standard deviations. The 6.5% value is the total TIP instrument uncertainty that corresponds to a total TIP reading uncertainty of 8.7%; 8.7% total TIP reading uncertainty is that value used in reload licensing submittals.

IV. ACTIONS TO BE TAKEN IF CRITERIA NOT MET

Startup activities would continue and all thermal limits would be closely monitored to assure compliance with all Technical Specification Limiting Conditions of Operation. The responsible test engineer would examine the test data and analysis method carefully for errors. If no errors are found, the TIP system hardware would be checked with emphasis initially being placed on the checkout of TIP axial calibration and alignment. If faults are found, they would be corrected and the tests repeated. Another thing that could be checked if no faults are discovered would be the fuel channel location history for those channels located around the affected TIP's; channel bowing due to non-uniform neutron flux levels can affect the water gap dimensions around the TIP detector. This would in turn affect the power calculated for one member of a symmetric TIP pair; analysis by General Electric has demonstrated that the probability is high that thermal limits will be affected conservatively by such TIP asymmetrics. If no hardware faults or calculational errors are found, operation would continue and all thermal limits would be closely monitored.