AUG 1 . . 74

Docket # STN-50-320

Voss.A. Moore, Jr., Assistant Director for LWR's, Group 2, L

FIRST ROUND QUESTIONS - THREE MILE ISLAND NUCLEAR STATION, UNIT 2

Plant Name: Three Mile Island, Unit 2
Licensing Stage: OL
Docket No: STN 50-320
Responsible Branch and Project Manager: LWR 2-2, B. Washburn
Technical Review Branch Involved: Reactor Systems Branch
Requested Completion Date: August 22, 1974
Description of Review: First Round Questions
Review Status: Awaiting Information

Adequate responses to the enclosed list of questions and comments are required before we can complete our review of the subject application. Commitments with respect to 10 CRF Part 50.46, Appendix K and WASH 1270 are required from the applicant.

These questions are the result of the review by the Reactor Systems Branch of sections 4.4, 5.1, 5.2.2, 5.3, 5.5, 6.3, 15 and 16 of the SAR. We will have further questions with regard to the Technical Specifications (Chapter 16) and testing of the ECCS (Chapter 14, which is incomplete).

Original Signed by
Victor Stello

Victor Stello, Jr., Assistant Director
for Reactor Safety
Directorate of Licensing

Enclosure:
Questions

cc; S. Hanauer, DRTA
F. Schroeder, L
A. Giambusso, L
W. McDonald, L w/o encl.
K. Kniel, L
B. Washburn, L
T. Novak, L
J. Watt, L
~~W. Minners, L~~
R. Kirkwood, L
W. Hodges, L
S. Varga, L

69 173

7904240173

A

OFFICE		L:RSB	L:RSB	L:RSB	L:RS
SURNAME		JWatt:ch	WMinners	TNovak	VStello
DATE		8/15/74	8/15/74	8/16/74	8/ /74

Form AEC-318 (Rev. 9-53) AECM 0240 U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

THREE MILE ISLAND

21.18 (4.3.1.3) Provide a schedule for the submittal of a review of the shutdown system design, plans for any proposed plant changes required to make the consequences of an anticipated transient without scram acceptable and the results of supporting analysis as required by paragraph II-B of appendix A to WASH-1270.

21.19 The thermal-hydraulic design basis requires that the minimum DNBR under operating conditions and transients does not fall below 1.3. At a given value of maximum linear heat generation rate, the radial power ratio affects DNBR more than the axial power ratio. Therefore, merely specifying the maximum linear heat generation rate and the product of axial and radial peaking values does not guarantee that the minimum DNBR limit will not be exceeded. What assurance is there that the radial peaking factor will not exceed the values listed in Section 4.4 and the values used for the safety evaluation of the plant in Section 15.

21.20 (4.4.2.1) Table 4.4-1 lists Rancho Seco as an essentially identical NSSS. Rancho Seco was granted a license limiting its power to 2568 MWt subject to later review of startup reports and initial operating experiences. Also, satisfactory operating experience of the prototype Oconee Unit 1 is required. Summarize this experience and show how it justifies the design thermal rating of Three Mile Island Unit 2.

21.21 (4.4.2.7) Identify reactor internal elements critical to the safe operation and control of the reactor. Tabulate for these elements the limiting design loads along with the most severe up, down and horizontal loads predicted during transient analysis. Identify the events creating the most severe loads.

21.22 (4.4.3.5) Provide the results of the calculation of maximum fuel clad strain for operational transients to end of life.

21.23 (4.4.3.5) In addition to load changes at constant pump combinations, the reactor coolant system must be demonstrated to be free of undamped oscillations or other hydraulic instabilities for all conditions of steady state operation, for all operational transients, for all load following maneuvers and for partial loop operation. Provide analysis, operational experience and experimental results providing this assurance.

21.24 (4.4.4) Discuss experience in observing crud or scale build-up (or absence of) during the life of a plant. Discuss how crud build-up is considered in heat transfer analysis and component design.

21.25 (4.4.5) Describe and discuss instrumentation for vibration and loose parts monitoring in the reactor coolant system.

21.26 (5.2.1.1) Identify the margins in net positive suction head for the operating main coolant pumps when operating with one or two pumps shut down.

21.27 (5.2.2.2) What is the allowable back pressure for the safety valves? What is the basis for this limit? Provide the method used, including experimental verification, in determining that the back pressure limit is not exceeded. If this limit is exceeded, what would be the effect on safety valve relief capacity?

21.28 (5.2.2.3) Show that all the assumptions and initial conditions used in the BAW-10043 analysis are applicable to the Three Mile Island # 2 plant.

21.29 (5.2.2.3) BAW-10043 does not provide the basic plant parameter such as plant geometry and power level. Further, BAW-10043 does not provide the set points for both the primary and the secondary safety valves. Provide this information.

21.30 (5.2.2.3) BAW-10043 does not address the severity of a complete loss of feedwater on the overpressure protection capacity provided. Provide the analysis to substantiate the adequacy of safety valve discharge capacities for a complete loss of feedwater transient.

21.31 (5.2.2.3) In the BAW-10043 analysis, pressurizer spray is assumed not to operate although the high pressurizer pressure signal was used to scram the reactor. Provide an analysis where the spray is assumed to operate and therefore scram is delayed.

21.32 (5.2.2.3) Show that the pressurizer does not go solid for any overpressure transients. Otherwise, provide the bases for the water discharge rates through the safety valves.

21.33 (5.5.7) The capability of the RHR system to perform its shutdown cooling function assuming the most restrictive single active failure in the RHR system has not been demonstrated. The RHR system is not single failure proof and, therefore, violates the intent of AEC General Design Criterion 34. An example of a single failure which could render the RHR system inoperable is a failure-to-open of one of the isolation valves in the RHR line leading from its associated recirculation loop. Such a single failure could place the reactor in the position of not being able to achieve a cold shutdown condition within a reasonable period of time. It may be possible that some "bootstrap" type of operation outside of the RHR system could be effective in achieving a degree of shutdown capability (such as with the ECCS), however, it is the intent of GDC 34 that the system normally utilized to place the plant in a cold shutdown condition (the RHR system) be single failure proof.

Also, since the RHR system is a low pressure system for which overpressure protection is required, any design modifications should not reduce the level of protection against overpressurization.

The RHR system should be modified so as to be immune to single active failures before final Regulatory staff approval.

21.34 (6.3.1.1) The AEC "Interim Acceptance Criteria" has been superseded as stated in the Federal Register, Vol. 39, No. 3-Friday, January 4, 1974. It is required for Three Mile Island 2 that analysis and evaluation of ECCS cooling performance following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of 10CFR 50.46 using an Evaluation Model in conformance with Appendix K. A commitment is required from the applicant identifying when the Safety Analysis Report will be revised and resubmitted so that the review may proceed. This comment applies to Chapters 6 and 15.

21.35 (6.3.2) Refer to Figure 6.3-1, the ECCS P&ID. Starting at either ECC vessel injection nozzle, trace back along the piping toward the low pressure system, through two check valves, the reactor building boundary, and a normally open motor operated valve. At this last valve there is a transition from high pressure to low pressure piping. Our concern is that no means are provided to detect leakage from the reactor coolant system back through the first (relative to the RCS) check valve, or from the core flooding tank (CFT) back through the second check valve. In the latter case, a decreasing CFT level may be sufficient indication of leakage for the operator. However, in the former case, undetected leakage from the RSC could pressurize the line between the two check valves for an undetermined period of time. Subsequent failure of the second check valve or the CFT check valve would result in a LOCA (outside containment in the first case, inside in the second) with diminished ECCS capability. Thus, the failure of one check valve could lead to a LOCA and a degraded ECCS. A second concern is that no pressure relief devices are shown on the Figure.

A change in design or monitoring should be made so that full credit can be taken for both check valves as protection against back leakage from the RCS. Such a change could take the form of a pressure indicator between the check valves, use of high pressure piping throughout, additional valves, addition of safety valves, different valve administrative alignment, or a combination of these.

21.36 (6.3.2) The ECCS design shown in Figure 6.3.1 does not meet the requirements of GDC 35. A failure of one injection line resulting in a LOCA coupled with a single failure in the other injection train would incapacitate the ECCS. The proposed design for such B&W plants as North Anna 3/4, Bellefonte, Greenwood, and WPPSS are examples of acceptable designs with respect to low pressure injection. Provide a description of the re-designed ECC system which fully complies with GDC 35. Include a discussion of the design basis and an evaluation of the operation of the system.

21.37 (10.1) Provide an enlarged (legible) Figure 10.1-2.

21.38 (15.1.2) Provide a curve showing the effect of reactivity insertion rate on minimum DNBR. For this insertion rate which gives the minimum DNBR provide a curve of DNBR vs. time. Assume that the transient starts at 102% rated power, 2132 psia and 559°F inlet temperature.

21.39 (15.1.2) What values of radial peaking factors and enthalpy rise factors ($F_{\Delta H}$) were used in the analyses presented in Chapter 15?

21.40 (15.1.5) For the loss of Coolant Flow Analysis, provide curves of DNBR vs time and hot spot heat flux vs. time for the four pump shutdown.

21.41. (15.1.5) What is the DNBR for steady-state operation at the conditions assumed for the start of the flow coast-down transient (102% rated power, 2135 psia and 559°F inlet temperature)?

21.42 (15.1.5) Provide evidence that the uncertainty in inlet temperature is only 2°F. Reference: Table 15.1.5-1.

21.43 (15.1.8) Determine if one motor driven emergency feed pump (470 GPM) is sufficient to bring the plant to a safe shutdown condition. The event postulated is as follows: A rupture occurs in the high energy steam supply line to the emergency feedwater pump turbine. This is coupled with the active failure of one motor driven emergency feed pump.

Provide analysis and discussion of this postulated event.

21.44 (15.1.8) Provide an analysis for a feedwater line rupture. In the analysis justify the method used to calculate break flow, the sizes and locations of breaks. Show that the single failures considered in the analysis are the most limiting ones. Further, if the pressurizer goes solid as a result of this accident, provide bases for water discharge rates through safety valves.

21.45 (16.2.2) The set points for the various overpressure protection devices should be stated under "Specifications."

69 178