Three Mile Island Unit No. 2

Review of the Proposed Modifications of TMI Unit 2

Steam Generator Instrumentation Program

Safety Evaluation Report

Mechanical Engineering Branch

Division of Systems Safety

Introduction

By letter dated Dec. 22, Metropolitan Edison Company has submitted a proposed Steam Generator Instrumentation Program. The purpose of the measurement program to be conducted on the "B" OTSG at TMI-2 is to provide tube response, flow data and plant process data. These data are to be used to evaluate the effect of normal and transient plant operation on the response of tubes in the OTSG and relate the response of the tubes to the most plausible steam flow excitation mechanism. It is proposed to acquire the following data during the test program.

- Response measurements of on-lane tubes and off-lane tubes at the upper two spans and between the ninth and tenth support plates during normal operating and transient conditions.

- Measurements of steam pressure and flow during transients such as turbine stop valve testing for correlation with tube response.

  The measurement programs at Oconee Nuclear Station Units #1 and 2 were similar to the proposed program at TMI-2.

Discussion

A modification of TMI Unit 2 OTSG "B" is necessary to accommodate the instrumentation. The detailed stress calculations in the reports (References 1 through 7) were performed by the applicant to demonstrate that the structural integrity of individual components as well as the total OTSG in its final configuration will be equivalent to that described in the FSAR.

The stress report for the instrumentation program (ref. 2) contains the sizing calculations for the primary handhole, the modified 5 in. secondary handhole and the primary manway to support the steam generator instrumentation program. Also included in this report are calculations which show that a material change from SA-540, B-23 Class 5 to SA-540 B-23 Class 4 for handhole and manway bolts will have no significant effect on the design stress levels. The stress levels in the primary handhole, secondary handhole and the primary manway (including bolts) have been shown to be within ASME Code Section III allowable limits.

The OTSG enclosure analysis (ref. 4) verifies the structural adequacy of the enclosure cans for routing accelerometers cables through the OTSG head at the inspection and manway openings. Stress resulting from flow loads, and thermal stresses are considered and a fatigue analysis has been performed. The enclosure cans have been found to be structurally adequate. The allowable bolt torques for the studs through the cans have been found to be greater than those required for both initial and subsequent torque-ups.

An analysis to verify the structural adequacy of the internal instrumenation routing has been performed. The components which carry the main structural loads are investigated, bushing, lockings, caps, and other components not highly stressed are not addressed. The routing has been found adequate for one fuel cycle (2 yrs.) subject to the following conditions:

1. The pipes should not be fitted tightly into the elbows.

2. The bolt torque should not exceed

    a. Yoke: 125 ft-lbf

    b. Elbow: 15 ft-lbf

    c. Wall bracket: 15 ft-lbf (full weld)

       11 ft-lbf (3/4 in weld)

3. The pipe cover should have 1 1/4 inches of welds between supports.

Evaluation

The NRC staff has reviewed the information submitted by the licensee to show compliance with the equipment specification, and our evaluation is as follows:

1. Information provided by the applicant provides sufficient assurance that neither the structural integrity nor the operational characteristics of any safety related equipment is ~~e~~affected by the proposed modifications.

2. Additional structures will not fail as a result of these modifications and the consequences of an accident will not be increased by this mechanism.

3. The instrumentation programs at Oconee 1 and 2 are very similar to the one proposed at TMI-2. The operating experience at Oconee plants provides additional assurance that there is no reduction in the safety margins as defined in the TMI-2 FSAR and technical specifications.

Based on the above evaluation the staff finds that the proposed modifications do not pose any danger to the health and safety of the public and are acceptable.

H. Silver

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JAN 17 1978

MEMORANDUM FOR: D. B. Vassallo, Assistant Director for Light Water Reactors, Division of Project Management

FROM: R. L. Tedesco, Assistant Director for Plant Systems, Division of Systems Safety

SUBJECT: SUPPLEMENT NO. 3 TO THE THREE MILE ISLAND NUCLEAR STATION, UNIT 2, SER

In our SER, we concluded that the design for each diesel generator air starting system was acceptable based on our position that the system contain two air storage tanks with a total capacity of ten starts. In Amendment 60, the applicant reduced the capacity of the two air storage tanks to a total of five starts. This design is unacceptable as stated in the Supplement. Attached is our revised evaluation.

R Tedesco

Robert L. Tedesco, Assistant Director
for Plant Systems
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Enclosure

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9.5.6.2 Other Deisel Generator Auxiliary Systems

In the SER, the design of the air starting system for each diesel generator was found acceptable based on a proposed system containing two air storage tanks with a total capacity for ten starts. In Amendment 60, the applicant reduced the air storage capacity from ten to five starts. We find this change unacceptable. As stated in Paragraph II of Standard Review Plan Section 9.5.6, it is our position that an acceptable design contains two storage tanks with a total capacity for ten starts.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

S. Varga
H. Silver
Harley

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MEMORANDUM FOR: D.B. Vassallo, Assistant Director for Light Water Reactors, DPM

THRU: RC P.S. Check, Chief, Core Performance Branch, DSS

FROM: R.O. Meyer, Leader, Reactor Fuels Section, CPB, DSS

SUBJECT: SER SUPPLEMENT CONCERNING BURNABLE POISON MATERIALS

Plant Name:	Three Mile Island, Unit 2
Docket Number:	05000320
Milestone Number:	24-24
Licensing Stage:	OL
Responsible Branch and Project Manager:	LWR-4 H. Silver
Systems Safety Branch Involved:	Core Performance Branch
Description of Review:	SER Supplement Input
Requested Completion Date:	December 16, 1977
Review Status:	Complete

The Reactor Fuels Section of the Core Performance Branch has prepared the attached supplement to the TMI-2 SER. The supplement describes our review of the Gd_2O_3-UO_2 demonstration fuel rods and B_4C-C burnable poison rods to be irradiated during cycle 1. We conclude that the proposed irradiation of a few test rods poses no safety concern for TMI-2. We thus approve the irradiation tests subject to the condition that surveillance and post-irradiation examinations be performed and reported to the NRC. We note in passing that B&W would be required to submit a more comprehensive safety analysis for these materials before including widescale uses in future applications.

Ralph Meyer
Ralph O. Meyer, Leader
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Division of Systems Safety

Enclosure:
(as stated)

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Gd_2O_3-UO_2 and B_4C-Graphite Burnable Poison Rods
Materials, Mechanical and Thermal Design Evaluation

Description

Two new burnable poison materials are proposed for limited use in test rods to be irradiated in TMI-2, Cycle 1. These burnable poisons, while new to B&W pressurized water reactors, have seen extensive use in other thermal reactor designs. Specifically, B_4C-C (graphite) burnable poison rods are used in high temperature gas cooled (HTGR) reactors of General Atomic design, such as Fort St. Vrain, and Gd_2O_3-UO_2 fuel rods have been used in BWRs for several years and have also seen limited use in PWRs as well.

Eight Zircaloy-clad burnable poison rods (BPRs) using B_4C in graphite matrix (B_4C-C) pellets will be employed in place of eight standard BPRs using Al_2O_3-B_4C pellets. The boronated graphite (B_4C-C) consists of B_4C particles dispersed uniformly in a pelletized graphite matrix. Approximately 97% of the pellet is matrix material. Nominal dimensions of the BRPs are provided in Table I. Comparison with Al_2O_3-B_4C BRPs shows the dimensions to be identical.

Four fuel assemblies, each containing four Gd_2O_3-UO_2 fuel rods (total of 16) are also to be irradiated in TMI-2, Cycle 1. Enrichment in the gadolinia-bearing rods was reduced from 1.98 to 1.80% (by weight) U-235, to compensate for a reduction in Gd_2O_3-UO_2 pellet thermal conductivity relative to pure UO_2. The Gd_2O_3-UO_2 fuel pellet dimensions, total column length, stack weight, cladding material and dimensions, grids and end fittings are identical to those in non-gadolinia bearing MK-B4 fuel assemblies.

Design Evaluation

B_4C-C BPRs

The principal performance considerations of concern for B_4C-C involve irradiation-induced swelling, gas release, and compatibility with cladding and coolant. Measurements of irradiation-induced swelling of B_4C-graphite at General Atomic (Ref. 1) indicated that the dimensional change in boronated graphite is controlled by fast neutron damage to the graphitic matrix and B-10 fission product damage in the graphite binder matrix; not by swelling of B_4C during irradiation. The curve used by B&W for their B_4C-C swelling model is based on data from irradiation of 20-30% (by weight) B_4C in graphite (Ref. 2), whereas ~3% (by weight) B_4C in graphite will be used in TMI-2. The lower B_4C content results in a somewhat lower expected swelling contribution from B-10 fission fragment damage. Hence, the use of the swelling curve for 20-30% (by weight) B_4C in graphite to predict the swelling of the B_4C-C BPRs, is, in this sense, conservative. Like the pellets irradiated in references 1 and 2, the B&W pellets for the demonstration BPRs are fabricated by extrusion. This produces an anisotropy in the crystallographic orientation of the graphite matrix. The degree of anisotropy is important because it affects the swelling behavior. Purchase specifications on the B_4C-C pellets limit the anisotropy to values below those used to develop the swelling curve in the General Atomic tests. B&W calculations yield ~1% cladding strain (resulting from B_4C-C swelling) at end of cycle-1. This is well within their 3% design limit, which is based on thermal-hydraulic calculations of the amount of coolant flow reduction required to cause slug-flow boiling.

Like the Al_2O_3-B_4C BPR, the B_4C-C BPR accumulates helium and lithium daughter products as a result of the absorbtion of thermal neutrons via the B-10 (n, α) Li-7 reaction. In its model for gas release, B&W assumes that all the helium produced is released from the B_4C-C compact to the available plenum volume. B&W's calculation of the resultant BPR pressure shows that the internal rod pressure is always below system pressure, even for the most limiting moderate frequency transient. This satisfies a design criterion for burnable poison rod pressure by preventing a condition under which ballooning of the cladding could occur.

In terms of pellet/cladding chemical compatibility, the main concerns involve the potential for carburization and hydriding of the Zircaloy, since formation of zirconium carbide or hydride embrittles the cladding and increases the likelihood of cladding breach. From thermal-hydraulic design calculations, using the most conservative values for irradiated thermal conductivity, thermal expansivity, and pellet and cladding tolerances, the maximum pellet surface temperature and cladding inner surface temperatures were 814 and 668°F, respectively. Since the minimum surface temperature for carbide formation is approximately 1450°F (Ref. 3), there is ample margin to preclude carbide formation.

With respect to hydriding, B&W uses the same fabrication procedure, including the same drying process, as that used in B_4C-alumina rods. In this procedure, the rods are dried after loading, just prior to sealing, in the belief that removal of trace amounts of moisture reduces the available source of hydrogen required for hydriding to occur. Hydriding and resultant cladding perforation has been identified as having led to a power anomaly (Ref. 4), which occurred when the B_4C in some B_4C-Al_2O_3 burnable poison rods was leached out by the primary coolant. By a similar process, if the B_4C-C BPR Zircaloy cladding were perforated, either due to hydriding or some other process, the B_4C would be rapidly lost through a two-step chemical reaction involving (1) the reaction of B_4C with H_2O to form B_2O_3, followed by (2) formation of boric acid by contact of the B_2O_3 with additional water. However, even if this were to occur so that all the B-10 were to be dissipated, the major result would be an increased power peaking in the surrounding rods in the fuel assembly. Analyses have been performed which show that , neither at BOL when the poison worth is greatest nor later in life when the poison worth is lower but assembly power is greater, is there inadequate margin between the accident conditions and the peaking limits.

In summation, the major performance considerations for B_4C-C, involving irradiation-induced swelling, gas release, and chemical compatibility, have been addressed. We thus believe there is reasonable assurance that cladding integrity will be maintained throughout cycle 1 operation. However, even if all the B-10 in the BPRs were rapidly removed via primary coolant ingress through cladding perforations, the neutronic effects would be unimportant because the number of BPRs (8) is small and they are well-spaced in one BPR spider. Therefore, we conclude that no significant safety concern exists regarding the proposed irradiation of the eight B_4C-C test BPRs.

TABLE 1

BPR Descriptions *

Cladding		Standard Al_2O_3-B_4C BRP	B_4C-C BRP
Zircaloy-4	OD	0.430 in.	0.430 in.
	ID	0.362	0.362 in.
	Min. Wall Thickness	0.0325 in.	0.0325 in.
Pellets	OD	0.340 in.	0.340 in.
	Stack Length	126 in.	126 in.

*nominal dimensions are given

Gd_2O_3-UO_2 Fuel Rods

The principal performance concerns for Gd_2O_3-UO_2 involve the effects of the gadolinia additions on material's properties such as thermal conductivity and irradiation-induced densification. To compensate for a decrease in Gd_2O_3-UO_2 pellet thermal conductivity (relative to pure UO_2), the gadolinia-bearing rod enrichment was reduced from 1.98 to 1.80% (by weight) U-235. With this lower enrichment, B&W has stated that the maximum gadolinia pin power remains less than 81% of the peak pin power in the core at all times during the cycle.

B&W has also stated that there is no discernable difference between UO_2 and Gd_2O_3-UO_2 densification. Recent experimental evidence (Ref. 5) suggests, however, that Gd_2O_3-UO_2 rods will densify more in reactor than pure UO_2. This tendency is further exaggerated, GE reports, by changes in fuel cracking and relocation behavior. Thus, whereas the Gd_2O_3-UO_2 rods will have a lower linear heat generation rate (LHGR) than pure UO_2, the enhanced densification of the gadolinia-bearing rods, coupled with their reduced thermal conductivity, will have the effect of increasing pellet temperatures. Therefore, despite the substantial reduction in rod power, offsetting effects cause stored energy in Gd_2O_3-UO_2 fuel to be about the same as in UO_2 fuel. Nevertheless, taking into account the lower enrichment of the demonstration Gd_2O_3-UO_2 rods and their location in the core, the hi-power UO_2 rods are probably more limiting.

In summation, because they are relatively few in number and are not located in peak power assemblies, we conclude that no significant safety concern exists regarding the proposed irradiation of Gd_2O_3-UO_2 demonstration fuel rods and that there is reasonable assurance that cladding integrity will be maintained throughout cycle 1 operation.

Summary

There is no appraent safety concern regarding the proposed irradiation of B_4C-C and Gd_2O_3-UO_2 test rods. Furthermore, we encourage irradiation testing of fuel system design inovations prior to their introduction into widescale use. However, to reap the potential benefits to future plant safety analyses, observations of the performance of the test rods must be made. None has been proposed to the NRC. The irradiation of the test rods in TMI-2 is thus approved on the condition that (a) surveillance and post-irradiation examinations (PIE) be performed, including destructive PIE if rod perforations occur, and (b) results are reported to the NRC.

REFERENCES

1. O.M. Stansfield, "Irradiation Induced Dimensional Change in HTGR Control Materials," Gulf General Atomic Report GA-12035, April 28, 1972.

2. O.M. Stansfield, "Neutron Irradiation Effects in Boronated Graphite, Hafnated Graphite, B_4C, and HfC - Summary Report on the BG-1 and BG-2 Experiments," Gulf General Atomic Report GA-10648, June 13, 1971.

3. B. Lustman and F. Kerze, Jr., The Metallurgy of Zirconium, McGraw-Hill, 1955, p. 449.

4. "Report to Congress on Abnormal Occurrences," NRC Report, NUREG-0090-5, March, 1977.

5. G.A. Potts, "General Electric Densification Program Status," General Electric Company Proprietary Report NEDE-21282-P, Rev. 1, April 1977. (Non-Proprietary Version, NEDO-21282).