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

THRU: 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: LWR-4
and Project Manager: 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 O. Meyer, Leader
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Enclosure:
(as stated)

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CPB RDG FILE

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Gd₂O₃-UO₂ and B₄C-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, B₄C-C (graphite) burnable poison rods are used in high temperature gas cooled (HTGR) reactors of General Atomic design, such as Fort St. Vrain, and Gd₂O₃-UO₂ 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 B₄C in graphite matrix (B₄C-C) pellets will be employed in place of eight standard BPRs using Al₂O₃-B₄C pellets. The boronated graphite (B₄C-C) consists of B₄C 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 Al₂O₃-B₄C BRPs shows the dimensions to be identical.

Four fuel assemblies, each containing four Gd₂O₃-UO₂ 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 Gd₂O₃-UO₂ pellet thermal conductivity relative to pure UO₂. The Gd₂O₃-UO₂ 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

B₄C-C BPRs

The principal performance considerations of concern for B₄C-C involve irradiation-induced swelling, gas release, and compatibility with cladding and coolant. Measurements of irradiation-induced swelling of B₄C-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 B₄C during irradiation. The curve used by B&W for their B₄C-C swelling model is based on data from irradiation of 20-30% (by weight) B₄C in graphite (Ref. 2), whereas ~3% (by weight) B₄C in graphite will be used in TMI-2. The lower B₄C 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) B₄C in graphite to predict the swelling of the B₄C-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 2.1% 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 absorption 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 B₄C-C test BPRs.

TABLE 1
BPR Descriptions *

Cladding		Standard Al ₂ O ₃ -B ₄ C BRP	B ₄ C-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

Gd₂O₃-UO₂ Fuel Rods

The principal performance concerns for Gd₂O₃-UO₂ 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 Gd₂O₃-UO₂ pellet thermal conductivity (relative to pure UO₂), 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 UO₂ and Gd₂O₃-UO₂ densification. Recent experimental evidence (Ref. 5) suggests, however, that Gd₂O₃-UO₂ rods will densify more in reactor than pure UO₂. This tendency is further exaggerated, GE reports, by changes in fuel cracking and relocation behavior. Thus, whereas the Gd₂O₃-UO₂ rods will have a lower linear heat generation rate (LHGR) than pure UO₂, 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 Gd₂O₃-UO₂ fuel to be about the same as in UO₂ fuel. Nevertheless, taking into account the lower enrichment of the demonstration Gd₂O₃-UO₂ rods and their location in the core, the hi-power UO₂ 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 apparent 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 innovations 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

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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).