Department of Energy

Washington, DC 20585

February 7, 1997

Project No. 697

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington D.C. 20555-0001
ATTN: Mr. James H. Wilson

Subject: Submittal of Additional Information Regarding the Tritium Producing Burnable Absorber Rod (TPBAR) Lead Test Assembly (LTA) Topical Report

Gentlemen:

Reference: 1) Letter, TT Martin (NRC) to SM Sohinki (DOE), Request for Additional Information Regarding Topical Report on the Tritium-Producing Burnable Absorber Rod Lead Test Assembly, January 3, 1997

2) Letter, TT Martin (NRC) to SM Sohinki (DOE), Supplemental Request for Additional Information Regarding Topical Report on the Tritium-Producing Burnable Absorber Rod Lead Test Assembly, January 13, 1997

In response to the request for additional information presented in the referenced letters, enclosed please find ten (10) copies of unclassified, non-proprietary Additional Information regarding the Lead Test Assembly (LTA) Topical Report. The information also provides clarifications and responses to other questions raised during the public meeting held January 22, 1997 between NRC staff and DOE. The enclosed material includes proposed revisions to the LTA topical report as appropriate. Proposed revisions to the Quality Assurance section (Section 7) are not included in this submittal, as discussions between PNNL, Westinghouse and the host utility, Tennessee Valley Authority, are continuing. This information will be provided in a supplemental submittal. As discussed previously, the revised topical report will be submitted March 3, 1997.

100053

1/1
D051

PROJECT - 697

We continue to be available to discuss any further questions on the Topical Report. Please call Max Clausen at 586-8217 or Richard Latorre at 586-4262 to arrange a meeting.

Sincerely,

Max J Clausen for

Stephen M. Sohinki
Director
Office of Commercial Light
Water Reactor Production
Defense Programs

Enclosures:
As stated

ADDITIONAL INFORMATION
TO TOPICAL REPORT PNNL-11419
"REPORT ON THE EVALUATION OF THE TRITIUM PRODUCING
BURNABLE ABSORBER ROD LEAD TEST ASSEMBLY"

FEBRUARY 7, 1997

Project No. 697

CONTENTS

1) Table 2-5 - What Li enrichment was used to calculate the axial power? Many power and burnup-related parameters are well beyond the conditions sustained in prior tests. Viewed in this way, the TPBAR appears to be a large jump beyond prior experience.

Response:

The Tritium-Producing Burnable Absorber Rod (TPBAR) axial power is based upon the ^{6}Li enrichments listed in Table 2-5. *The following paragraph will be added to the topical report, §2.1, to address parameters beyond tested conditions:*

"Experimental irradiation tests of tritium targets were performed in the Advanced Test Reactor (ATR) at the Idaho National Engineering Laboratory. Experience and data from these tests is described in this chapter as it relates to the evaluation of the TPBARs. An objective of the TPBAR LTA irradiation is to obtain sufficient irradiation data under commercial PWR conditions to support full scale production irradiation of TPBARs. The ATR test data, although limited in some respects, is sufficient to support the evaluation of irradiating a small number of LTAs in non-limiting core locations. Experimental irradiation experience, as related to the TPBAR design, is summarized in Tables 2-5 and 2-6. The experimental data bounds expected LTA irradiation conditions with regard to pellet burnup (expressed in GVR[A] which is analogous to fuel MWD/MTU) and thermal neutron fluence. The TPBAR will be exposed to higher fast neutron fluence and time at operating temperature and pressure than was achieved during the ATR experimental tests. The experimental test data together with substantial TPBAR design improvements, large design margins, and conservative analysis provide a high degree of confidence in the performance of the LTAs."

2) Section 3.1.1 - Are WIMS-E and MCNP codes qualified for NRC use? Are auditable calculation files available for review of inputs, assumptions, and other details of the calculations?

Response:

Neither of these two codes are qualified by NRC for use in safety calculations. The neutronic studies performed for this report using WIMS-E and MCNP were scoping calculations only, therefore no auditable calculations are available for review. The analyses of record will be performed by Westinghouse using an NRC-approved methodology (PHOENIX) and documented in auditable calculation files. As stated in Section 3.4, the excellent agreement found in comparisons between PNNL WIMS-E and

---

[A]GVR is the ratio of STP gas volume to pellet volume and is directly proportional to the number of ^{6}Li reactions.

PHOENIX-L establishes confidence that the Westinghouse calculations will provide predictions of the reload core comparable to those in the WIMS-E scoping calculations. Additional calculations required by Westinghouse or the licensee will be performed in a manner that meets Westinghouse QA requirements. *(No changes to the topical report are proposed to address this RAI.)*

3) Section 3.1.1 - Provide an explanation for why the reactivity difference between MCNP and WIM-E of 0.5% appears high.

Response:

The following will be added to the last paragraph of the topical report, §3.1.1:

"The comparison between the two codes indicate differences in reactivity calculations of about 0.5% (5 mK), a ^{6}Li capture rate difference of 2.3%, and a maximum fuel rod fission rate difference of 2.8% in the lattice. The WIMS-E code is a deterministic, transport theory code and MCNP employs a statistical, Monte Carlo solution methodology. In addition, each code uses an independent set of nuclear cross section data libraries (European vs ENDF/B, respectively). In consideration of the substantial differences in calculation methodologies and the use of independent nuclear data libraries, some differences in calculated reactivity are to be expected, and in this case the difference is considered small."

4) Section 3.4 - Will the differences between PHOENIX-L and PHOENIX-P be documented in a topical report?

Response:

The following will be added to the topical report, §3.4, final paragraph:

"The differences between the PHOENIX-L and PHOENIX-P will be documented in a report provided to PNNL and the host utility. Such changes are subject to the reporting criteria imposed on the host utility by 10 CFR 50.46(a)(3). These differences will also be documented internally at Westinghouse in computer software verification and validation documentation. Auditable files will be available that document calculations performed for the core reload safety analysis."

5) Section 4.1.2 - Is the VIPRE code NRC-qualified? Are auditable calculation files available for review of inputs, assumptions, and other details of the calculations?

Response:

The VIPRE code has been reviewed by NRC. A technical evaluation report (TER) was issued in 1994, ITS-NRC-93-1 "Technical Evaluation Report: VIPRE-01 Mod-02 for PWR and BWR Applications, EPRI-NP-2511-CCM-A, Revision 3."

The following will be added to the topical report, §4.1.2, second paragraph:

"The application of the code for analysis of Plants A and B does not use plant-specific heat transfer or CHF correlations. The application uses only the well known Dittus-Boelter single phase heat transfer correlation, and the cylindrical heat conduction model (for heat conduction across the guide thimble wall, between the fuel channel and TPBAR channel). Flow data used in the analysis is provided by the utility/Westinghouse, and is used as a boundary condition. The VIPRE code is not used to evaluate flow distribution or core pressure drop."

In addition, the following will be added to the topical report, §4.1.2, as part of a new final paragraph:

"The analysis performed using the VIPRE code will be verified by hand calculations assisted by MATHCAD software using established thermal-hydraulic calculation methods. The final calculations will be documented in an auditable calculation file. The analysis will be reviewed by Westinghouse during the performance of the core reload safety evaluation."

6) Section 4.1.2 - The last paragraph suggests that the results in Figure 4-1 are preliminary and are more conservative than Westinghouse's normal method. Will a final T-H analysis be submitted or is the decision to be based on the results given? What assumptions are more conservative than normal?

Response:

The calculations described in the original topical report were preliminary in that they were not documented in an auditable manner and the model did not include the guide thimble dashpot region. *The following will be added to the topical report, §4.1.2, as a final paragraph:*

"The analysis described above is preliminary. If final thermal-hydraulic calculations are more limiting for the host reactor, the updated results will be

evaluated as part of the host utility 10 CFR 50.59 evaluation performed for the reload core with LTAs. The analysis performed using the VIPRE code will be verified by hand calculations assisted by MATHCAD software using established thermal-hydraulic calculation methods. The final calculations will be documented in an auditable calculation file. The analysis will be reviewed by Westinghouse during the performance of the core reload safety evaluation."

(If, at the time that Revision 1 to the topical report is finalized, final calculations are complete, the topical report will provide those results and the added paragraph will be revised accordingly.)

For both Condition I and II events, the Westinghouse calculations for BP rods took credit for cladding superheat in evaluating (and preventing) the onset of subcooled nucleate boiling. PNNL calculations do not take credit for cladding superheat, they apply the temperature limit to the bulk coolant.

Also, for Condition I and II events, the Westinghouse BP calculations used nominal inlet temperature, outlet pressure, and minimum flow (based on dimensional tolerances without an additional 5% flow maldistribution penalty). For Condition I, PNNL calcuations used the most limiting conditions as would be applied to LOCA and DNBR evaluations for inlet temperature, pressure, and flow maldistribution (this results in 5% more flow reduction than just for dimensional tolerances). For Condition II, PNNL calculations use similar assumptions as Westinghouse (nominal conditions, with dimensional tolerances accounted for).

7) Section 6.1 - Provide the value of the total off-site exposure and the regulatory limits, rather than stating that the increase due to TPBARs is small.

Response:

New Tables 6-1 and 6-2 (attached) will be added to the topical report in place of existing Tables 6-1, 6-2 and 6-4. In addition, previous references to Tables 6-1, 6-2 and 6-4 in the topical report will be edited to read as follows:

§6.1, sixth paragraph, first two sentences:
"The total off-site dose including a chronic waterborne, annual tritium release of 214 Ci are tabulated in Tables 6-1 and 6-2 ("Normal Operation" condition). The dose to the maximum exposed individual (MEI) during normal operation is increased by a negligible amount over current total doses near the Reference Plant A or Reference Plant B sites."

§6.3.3.1, first paragraph, third sentence:

"Tables 6-1 and 6-2 ("Cladding Breach" condition) present the change in total body dose to the maximum exposed off-site individual due to the release of the inventory of a breached TPBAR to the environment at a constant rate."

§6.4.2, second paragraph, third and fourth sentences:
"Tables 6-1 and 6- 2 ("Steam Generator Tube Rupture" condition) give the off-site total body doses resulting from this event with and without LTAs. The total body dose to an individual at the exclusion area boundary (EAB) would be essentially unchanged at either Reference Plant A or Reference Plant B."

§6.4.3.3, second sentence:
"The release of the entire 369,500 Ci end-of-cycle LTA inventory to the containment would increase the offsite total body doses near Reference Plant A or Reference Plant B by a negligible amount, as shown in Tables 6-1 and 6-2 ("LOCA" condition)."

8) Section 6.1 - Why is the release rate modeled as a constant? As pressure in the TPBAR increases, would the release rate not also increase? If a constant release rate is used as an approximation, it should be a conservative end-of-life number.

Response:

To evaluate this issue, calculations were performed using time-dependent release rates for the TPBARs. Using neutronic data for Plant B, the end-of-cycle tritium inventory increased over that calculated for a constant release rate. The peak tritium concentration also increased, which results in an increase in the offsite dose during a SGTR (not reflected in attached Tables 6-1 or 6-2). These increases were small and will be reflected in the doses that will be presented in the revision to the classified topical report. For the breached rod case, the end-of-cycle tritium inventory decreases compared to the constant release model. The breached rod case assumes all tritium is released to the coolant as it is produced and the production decreases over the cycle due to ^{6}Li depletion.

9) Section 6.2.2 - Does the plant loading and handling procedure account for the actual weight of BPRA? In particular, is there a load limit to prevent damage to the fuel assembly from withdrawing a stuck rod assembly? If so, does the procedure need to be modified to account for the lighter weight of the TPBAR?

Response:

The following will be added to the topical report, §6.2.2, first paragraph:

"The plant handling procedure for the burnable poison rod assembly (BPRA) handling tool contains a precaution stating that if significant resistance is felt during removal that

the assembly is either fully withdrawn or stuck and visual verification of position must be made. The BPRAs are removed by a hand winch operated tool. There are no provisions for a load monitoring device to be attached to this tool. This tool is used to handle BPRAs containing as few as four full length pins to as many as 24 full length pins. The LTA weight is bounded by these conditions and no change to the procedure or handling equipment is warranted."

10) Section 2.2.6.3 - The staff has identified an additional type of accident that should be considered in the TPBAR LTA topical report.

A weld in an iridium capsule failed during irradiation at the Oak Ridge HFIR reactor, allowing water to enter the capsule. As long as the capsule was in the spent fuel pool, the heat removal was adequate to keep the water in the capsule in liquid form. However, when the capsule was withdrawn from the pool and placed in a dry shipping cask, the water boiled and the resulting pressure increase ruptured the capsule.

Although Section 2.2.6.3 mentions a water-logged TPBAR, the increase and rate of change of temperature due to dry shipping have not been analyzed and the resulting maximum pin temperature and pin internal pressure have not been calculated. The heat generation of the TPBAR at 150 hours after irradiation, given in Section 6.2.1.2, is 0.024 kW per LTA, or 3 watts per pin. At this power level, it seems unlikely that the pin temperature would ever exceed the boiling point. However, a simple assertion is not adequate to make the case.

Provide an analysis of the increase and rate of change of temperature in a water-logged TPBAR pin due to dry shipping and calculate the resulting maximum pin temperature and pin internal pressure. The analyses should consider the dry conditions during dry shipping that result in a low heat removal rate in spite of the enhanced heat transfer provided by the water in the interior of the pin. The analysis should also consider the effect on tube failure pressure (normally 3000 psi) because of the assumed failed weld or clad.

Response:

The following will be added to the topical report, §2.2.6.3:

"Water-logged TPBAR in Dry Cask Storage

"If a water-logged TPBAR is placed in dry cask storage, there is a potential for an increase in TPBAR temperature and pressure due to internal heat generation of the TPBAR. The concern is that the water in the TPBAR could boil and cause overpressurization. The TPBAR generates about 3 watts of heat 150 hours after

shutdown. An analysis of TPBARs stored in a dry cask shows that the maximum temperature increase of a TPBAR due to internal heat generation is less than 3°F and boiling in the TPBAR will not occur."

11) At the January 22, 1997 meeting with DOE, the staff requested that the topical discuss the potential for, and consequences of, misloading the LTAs into the core to address Standard Review Plan Section 15.4.7.

Response:

The following will be added to the topical report, as a new §6.3.4:

"TPBAR absorber pellet ^{6}Li loading errors are precluded by use of a single loading value for all pellets and administrative controls during fabrication of the TPBAR and TPBAR components. The LTAs are symmetrical, so there is no improper orientation. LTA loading errors are prevented by the same administrative procedures implemented by the host facility to prevent fuel assembly and burnable poison misloading. To preclude core misloading events, host utility procedures include confirmation of the final core configuration by video taped documention of the component identification numbers. Loading of an LTA in the wrong fuel assembly is bounded by host facility analysis for 'Inadvertent Loading of a Fuel Assembly into an Improper Position'.

"In the unlikely event that an LTA loading error occurs, the resulting power distribution effects will either be readily detected by the incore moveable detector system or the perturbation in core power distribution will be sufficiently small to be acceptable within the specified fuel design limits. An analysis will be completed to demonstrate that the LTA would not exceed TPBAR design limits even if it were to be loaded in the limiting fuel assembly in the core."

12) At the January 22, 1997 meeting with DOE, the staff requested that the topical discuss how the presence of the TPBARs in the core may affect the analysis of Anticipated Transients Without Scram (ATWS) events.

Response:

The following will be added to the topical report, as a new §6.3.5:

"The approach to the treatment of ATWS events for currently licensed PWRs is established in 10 CFR 50.62 in the form of prescriptive design requirements. The prescriptive design requirements of 10 CFR 50.62 are not explicitly applicable to the design of the TPBARs. During the consideration of ATWS events that formed the basis

for 10 CFR 50.62, reactor coolant system (RCS) maximum pressures attained during the course of an ATWS and the potential for RCS failure was the principal concern. TPBARs could only affect maximum RCS pressures attained during an ATWS by affecting reactivity assumptions in the ATWS analysis. The cross-section of the ^{6}Li absorber is not particularly sensitive to temperature variations, and has no resonance behavior that could adversely affect reactivity assumptions during an ATWS. Accordingly, the TPBAR mimics the neutronics behavior of conventional burnable absorber rods during a postulated ATWS event and has no impact on existing ATWS neutronics analysis. Therefore, compliance with the requirements of 10 CFR 50.62 will continue to be sufficient to address postulated ATWS events."

TABLE 6-1
Summary of Off-Site Radiological Consequences for the TPBARs in Plant A

Condition	Receptor	Current Total Body Dose[1] (rem)	Total Body Dose with LTA (rem)	Increase w/ LTA	Reference Value / Source
Normal Operation	MEI[2]	0.000839	0.000840	0.074%	0.003 rem / 10CFR50, App. I
Cladding Breach	MEI	0.000839	0.000872	3.93%	0.003 rem / 10CFR50, App.I
Steam Generator Tube Rupture[2]	EAB[3]	0.0717	0.0717	3.4e-3%	2.5 rem / SRP 15.6.3
	LPZ[4] (8 h)	0.0307	0.0307	4.7e-3%	
Fuel Handling Accident (Containment)	EAB	0.0142	0.0142	none	
	LPZ (2 h)	0.0058	0.0058	none	
Fuel Handling Accident (Auxiliary Bldg)	EAB	2.4000	2.4000	none	
	LPZ (2 h)	0.9500	0.9500	none	
LOCA	EAB	3.1060	3.1060	4.0e-4%	25 rem / 10CFR100
	LPZ (30 d)	2.8120	2.8125	0.019%	

NOTES:

[1] The total body dose was calculated using the following equation:

$$\text{Total Body Dose} = \gamma \text{ whole body dose} + w_T*\text{Thyroid dose} + w_\beta*\beta \text{ dose}$$

where:

w_T = Thyroid weighting factor from 10CFR20.1003 (= 0.03)
w_β = Beta skin dose weighting factor (= 0.01) from ICRP Publication 60 [Ref. 6-7]

[2] Maximum Exposed Individual.

[3] Exclusion Area Boundary for 2 h = Fenceline Receptor

[4] Low Population Zone = MEI Receptor

TABLE 6-2
Summary of Off-Site Radiological Consequences for the TPBARs in Plant B

Condition	Receptor	Current Total Body Dose[1] (rem)	Total Body Dose with LTA (rem)	Increase w/ LTA	Reference Value / Source
Normal Operation	MEI[2]	0.0007	0.0007002	0.029%	0.003 rem / 10CFR50, App. I
Cladding Breach	MEI	0.0007	0.000713	1.86%	0.003 rem / 10CFR50, App.I
Steam Generator Tube Rupture	EAB[3]	0.8625	0.8625	2.2e-4%	2.5 rem / SRP 15.6.3
	LPZ[4] (8 h)	0.2004	0.2004	2.2e-4%	
Fuel Handling Accident (Containment)	EAB	2.0075	2.0075	none	
	LPZ (2 h)	0.4663	0.4663	none	
Fuel Handling Accident (Auxiliary Bldg)	EAB	0.7815	0.7815	none	
	LPZ (2 h)	0.1815	0.1815	none	
LOCA	EAB	3.2874	3.2874	4.0e-4%	25 rem / 10CFR100
	LPZ (30 d)	2.0002	2.0007	0.027%	

NOTES:

[1] The total body dose was calculated using the following equation:

$$\text{Total Body Dose} = \gamma \text{ whole body dose} + w_T * \text{Thyroid dose} + w_\beta * \beta \text{ dose}$$

where:

w_T = Thyroid weighting factor from 10CFR20.1003 (= 0.03)

w_β = Beta skin dose weighting factor (= 0.01) from ICRP Publication 60 [Ref. 6-7]

[2] Maximum Exposed Individual.

[3] Exclusion Area Boundary for 2 h = Fenceline Receptor

[4] Low Population Zone = MEI Receptor

1) Section 1 - The design life for the assembly is 550 EFPD. What were the fluence levels used to establish the design life? Also, does the design take into account possible power uprate or extended cycles?

Response:

The neutron fluence values used in the design of TPBARs are identified in Table 2-5 of the Topical Report. The TPBAR was not specifically designed to take into account possible power uprates. The 550 EFPD represents an extended operating cycle relative to the planned cycle lengths for both plants. Neither Plant A nor Plant B plan to implement a power uprate for the period of LTA irradiation. *The following sentence will be added to the topical report, §2.1, fourth paragraph, after the fifth sentence:*

"The neutron fluence levels associated with the 550 EFPD design criteria are identified in Table 2-5"

2) Section 1.1 - "The TPBAR design has been developed to demonstrate... tritium leakage consistent with a TPBAR design goal of <6.7 Ci per rod per year...." This appears to imply that for full operation, the reference plant will be leaking thousands of curies of tritium per year of operation.

Response:

The design goal for TPBAR tritium leakage to the RCS for a full production core is less than 20,000 curies per year. Assuming approximately 3000 TPBARs in a production core, this equates to the <6.7 curies per rod per year used for evaluation of the LTAs. (The acceptability of the design level of leakage for the production core will be evaluated as part of the license application to support production.) Note that, although the design goal is *less than* 6.7 Ci per rod per year, dose calculations in Chapter 6 of PNNL-11419 (and in response to Reactor Systems' requests for additional information, item 7) are conservatively based on 6.7 Ci per rod per year. *(No changes to topical report are proposed to address this RAI.)*

3) Sections 2.2.1 & 5.3.1 - Why was 316 SS with 20% CW selected as the material for cladding and end plugs? Were higher strength, corrosion resistant alloys (e.g., Alloy 690, Inconel 718, Zircaloy-4) considered?

Response:

The following will be added to the topical report, §5.3.1 as a new beginning paragraph:

"The LTA TPBAR is an evolutionary target design and 316 SS has been the material historically used for tritium target cladding. The processes for permeation barrier application were developed using 316 SS cladding. Sufficient experimental and performance data for barrier-coated tubing using other materials is not available. 316 SS with 20% CW was specified to establish adequate strength while staying within the experience base established for 300-series SS."

In addition, the topical report, §5.3.1.2, will be revised to read as follows to further address the concerns regarding stress corrosion cracking of 300-series SS:

Stress corrosion cracking (SCC) in 300-Series stainless steel requires: 1) thermal sensitization or irradiation-induced susceptibility; 2) high stresses (near or above the yield stress); and 3) an aggressive environment (e.g., reactive species such as O, Cl, and/or F in an aqueous medium at concentrations much higher than typical levels in PWR coolant; typical levels are <5 ppb O, <50 ppb Cl, and <50 ppb F). Each of these factors are described below.

Material Susceptibility

Thermal treatments in the temperature range of 800°F (425°C) to 1500°F (815°C) have the potential to sensitize 304 or 316 SS. The process of applying an aluminide coating to the internal surface of TPBAR cladding requires temperatures within this range. Therefore, TPBAR cladding may be thermally sensitized.

Irradiation of 300 Series stainless steel to neutron fluence greater than 1 to 2×10^{21} n/cm^2 (E>1MeV) may decrease Cr concentrations at grain boundaries by promoting Cr diffusion away from the grain boundary region, thereby increasing the susceptibility to SCC [Ref. 5-4]. The peak neutron fluence expected for the TPBAR cladding is approximately 5×10^{21} n/cm^2 (E >1 MeV). Therefore, TPBAR cladding may be subject to some reduction in grain boundary Cr during irradiation, resulting in irradiation-induced susceptibility.

Stress

The TPBAR design inner pressure limit of 3000 psia is approximately 750 psi (5.2 MPa) above the reactor system external pressure of 2250 psia (15.5 MPa). Therefore, the tensile hoop stresses from internal pressurization remain very low throughout the period of irradiation. Irradiation-induced swelling of the pellets and irradiation growth of the getters are insufficient to stress the cladding. The absorber pellets remain

intact during irradiation. Absorber pellets generally do not fracture, but if a pellet fractured, localized stresses in the cladding due to interaction of the cladding with pellet fragments would be prevented by the intervention of the getter, which confines the absorber pellet material. The thin walls of the absorber pellets, nominally 0.040 in., will prevent mechanical damage from interaction between pellets and cladding, such as could result from fracturing of solid absorber pellets or fuel. The stress in the TPBAR cladding should be insufficient to cause or propagate SCC.

Environment

The external operating environment for the TPBAR during irradiation will be standard PWR coolant maintained within Technical Specification chemistry limits for oxygen and reactive compounds. Austenitic stainless steel is not considered susceptible to SCC in this environment [Ref. 5-7 and 5-8]. Issues such as boiling, crevices, highly borated solutions, and stagnant flow that can result in more severe environmental conditions are not present in the TPBAR operating environment.

During reactor shutdown and cooldown the TPBAR cladding is not susceptible to SCC because the temperature is below 200°F (90°C) and the stresses are low. Experience with SS-clad spent nuclear fuel at temperatures below 200°F (90°C) has shown that for storage periods up to 25 years, there is no measurable degradation of stainless steel cladding in pool water typical of PWR spent fuel pools [Ref. 5-9].

Type 304 stainless steel has been used extensively and successfully as cladding for BPRA rods, an application very similar to TPBAR cladding. Barrier-coated 316 stainless steel clad target rods were irradiated in the Advanced Test Reactor (ATR). Three rods were irradiated in static water capsule tests and eight rods were irradiated in a flowing water loop test (see Chapter 2, Table 2-5 regarding these tests). Coolant conditions for the loop test were approximately representative of commercial PWR conditions. No failure of the test rods occurred during the experimental irradiations. There have been no indications of failure during water storage of the test rods over the last five years. The WC-1 capsule test rod was irradiated for 281 EFPD with 222 EFPD accumulated at test goal temperatures and pressures. Extensive visual and metallographic examinations performed for the WC-1 test rod showed that the cladding and end-plugs (including the weld heat affected zone) were in excellent condition and showed no signs of cracking or other deterioration. Puncture tests have recently been performed on four of the irradiated test rods (three LOOP-1 rods and the B-1

capsule test rod) in order to analyze gases in the rods. All four rods were found to be pressurized after five years of storage.

Although the TPBAR material may become sensitized or susceptible, SCC is not anticipated because:

- The TPBAR has very low tension stress.
- PWR reactor coolant chemistry control program ensures a non-aggressive environment [Ref. 5-9]. There are no special environmental conditions that will promote SCC (e.g., stagnant coolant, crevices, etc.).
- The TPBAR will only be resident in the reactor for one cycle of plant operation.
- No SCC problems have been identified for stainless steel clad BPRA rods extensively used in PWRs.
- No failures have occurred in 316 stainless steel clad ATR test rods.

4) Section 2.2.4 - What is the basis for selecting 302 SS for the plenum spring material?

Response:

302 SS was selected for the TPBAR plenum spring since this material has been successfully used in burnable poison rod and fuel rod plenum spring applications in Westinghouse reactors. An analysis of the TPBAR plenum spring demonstrates that spring stresses are well within allowable stresses for this material during worst-case combinations of preload, manufacturing tolerances, pressure conditions, and component thermal and radiation growth. *The topical report, §2.2.4 will be revised by editing the second paragraph, third sentence, to read as follows:*

"Based on a conservative safety margin and satisfactory commercial reactor experience with the material in this application, the spring is expected to provide the bearing load required for shipping and handling."

5) Section 2.4 - Provide additional details regarding the nondestructive testing. What techniques are used for each component? What are the applicable standards?

Response:

The topical report, §5.4, will be revised to refer to a new Table 5-2 and Table 5-2 will be provided, as follows:

TABLE 5-2
Nondestructive Testing Techniques and Applicable Standards
for Acceptance of TPBAR and TPBAR Components

TPBAR Component	Method	Applicable Standard	Characteristic
Nondestructive Tests Performed by PNNL[(1)]			
Coated Cladding	Eddy Current		Coating thickness, uniformity of thickness along the tube, and inter-metallic phase
Final TPBAR Assembly	Radiography	NE F3-10 [(2)]	Welds, component placement
Final TPBAR Assembly	Helium Leak Test	ASME B&PV, Section V, Article 10	Rod cladding integrity (leak tightness)
Nondestructive Tests Performed by Subvendors			
316 SS Bar Stock	Ultrasonic	ASTM E213-93	Defects
316 SS Bar Stock	Liquid Penetrant	ASTM E165-95	End defects
Cladding tubing	Ultrasonic	ASTM E213-93	Wall thickness, flaws
Plated getters (tubes and disks	X-ray spectrometry	ASTM B568-91	plating thickness and coverage

NOTES:
[(1)] All personnel performing acceptance inspections will be certified to ASNT-TC-1A.
[(2)] DOE specification developed for fast breeder reactor program.

1) Section 8.3 - As indicated in the staff's letter dated November 1, 1996, the staff agrees that additional individual access authorization by NRC is not needed. However, the staff is still considering the issue of DOE conducting and granting security facility approvals at NRC licensee sites.

Response:

See Section 8.3 of the attached PNNL-11419, Rev. 1, Draft B.

8 SAFEGUARDS AND SECURITY

To protect DOE's investment, the host utility will account for each tritium-producing burnable absorber rod (TPBAR) and TPBAR lead test assembly (LTA) from receipt until offsite shipment. The TPBARs and some related documentation necessary to support the preparation of the host utility safety evaluation will be classified Confidential Restricted Data.[1] As classified material and documents, they require safeguards measures to prevent diversion and/or unauthorized access to or disclosure of classified information.

8.1 Materials Accountability

TPBARS will be handled and protected the same as nuclear fuel. Each TPBAR will have a unique number engraved or etched on the top endplug. The TPBAR LTAs will be identified by a unique serial number on the hold-down assembly such as is currently used on fuel inserts. These serial numbers will be used in shipping documentation and irradiation records so that they can be tracked and accounted for.

A carrier who meets Department of Transportation requirements for shipment of nuclear fuel will be used for the transportation of the TPBARs to and from the site. The accountability and control of the TPBARs will be conducted consistent with the Facility's administrative guidelines for handling of new and spent fuel and fuel inserts, and the special nuclear material accountability plan. Irradiation records will be provided to DOE to allow estimates of ^{6}Li conversion and ^{3}H production for accounting purposes, so that the utility need account for only the physical LTAs.

8.2 Physical Security of Classified Hardware

The TPBARs are visually unclassified; however, TPBAR internal characteristics are classified. Therefore, the TPBARs are considered classified material. No inspections other than visual will be allowed by personnel who have not been granted access to classified information.

The TPBARs will be brought to the site by a carrier approved by DOE who meets Department of Transportation requirements for shipment of nuclear fuel. Once inside the protected area, movement of the TPBAR LTAs will be monitored by personnel with DOE clearances. While the

TPBAR LTAs are stored in the new fuel storage racks or in the fuel pool, a suitable level of physical protection will be provided by normal plant security measures meeting the requirements of 10 CFR 73, supplemented with escort by DOE-cleared personnel, when appropriate and necessary according to the plant security plan (see discussion below). While the TPBAR LTAs are in the reactor with the reactor head bolted, they will be considered secure and no escort by DOE cleared personnel will be required.

8.3 Control of Classified Documents and Hardware

Under the requirements for a reactor operating license, the site must have a physical security plan meeting the requirements of 10 CFR Part 73 for physical protection of vital equipment and control of safeguards information. These requirements include a physical security organization, physical barriers, access controls, detection aids, communications, procedures for testing and maintenance of security equipment, and a safeguards contingency plan. Based on the facility meeting the requirements of 10 CFR Part 73, the site will only have to develop and implement procedures (under DOE assistance and guidance) for the handling of classified hardware. [2] Personnel will be granted access to classified information and hardware by DOE under 5 CFR 732 and 10 CFR 710 to meet Executive Orders 10450, 12958 [Ref. 8-1], and 12968 [Ref. 8-2]. These controls will meet or exceed those that would be required to meet 10 CFR Part 25 and 10 CFR Part 95 and shall be deemed to satisfy 10 CFR 50.37 requirements for licensee access to restricted data. In accordance with the memorandum of understanding between DOE and the NRC [Ref. 8-3], DOE has reached an agreement with the NRC that NRC licensees seeking access authorization from the DOE, based on their participation in the CLWR tritium program, do not require additional access authorization [Ref. 8-4].[3] The NRC has acknowledged this agreement and confirmed that conformance with DOE requirements relative to access authorization satisfy corresponding NRC requirements contained in 10 CFR Parts 25 and 95 as well as the facility operating license provisions contained in 10 CFR 50.37 relative to licensee access to restricted data [Ref. 8-5].[4] The DOE will perform the required background investigations appropriate to the level of access authorization being sought for the personnel and no additional NRC action is required for personnel access authorizations relative to the CLWR LTA program. Security facility approval will be coordinated between DOE and NRC to meet applicable regulatory requirements.[5] The DOE will perform the reviews of the facility to ensure that DOE classified hardware (e.g., TPBAR LTAs; host utilities will not receive or store classified documents) to be handled or stored at the facility is appropriately safeguarded.[6] The

granting of access authorization and the coordination of facility approval with NRC[7] for a DOE program in an NRC licensed facility is consistent with the direction provided in Executive Order 12968 and the NRC "Proposed Rule on Access to and Protection of Classified Information" (61FR40555).

8.5 REFERENCES

8-1. Executive Order 12958, "Classified National Security Information", dated April 17, 1995.

8-2. Executive Order 12968, "Access to Classified Information", dated August 4, 1995.

8-3. Memorandum of Understanding Between the Department of Energy and the United States Nuclear Regulatory Commission Under The Provisions of The National Industrial Security Program, dated September 19, 1996.

8-4. Letter S.M. Sohinki to J.H. Wilson, October 4, 1996 "DOE Clearances for NRC Licensees Supporting Tritium Program; Project No. 697"

8-5. Letter T.T. Martin to S.M. Sohinki, November 1, 1996 "DOE Clearances for NRC Licensees Supporting Tritium Program".

CHANGE DESCRIPTIONS:

[1] Editorial, clarification; revised wording to indicate that access to classified material may be necessary to support safety evaluation preparation rather than for safety committee review since the host utility does not plan to have classified documentation on site and not all safety review committee personnel will be authorized for access to classified documents.

[2] Clarification; Removed reference to site procedures for handling of classified documents since no classified documents will be sent to or maintained at the host plant site.

[3] Response to Security RAI; removed text related to security facility approval.

[4] Response to Security RAI; removed text related to security facility approval.

[5] Response to Security RAI, clarification; clarified that no further NRC action is required for access authorizations relative to the CLWR LTA program and that security facility approval would be coordinated between DOE and NRC.

[6] Editorial, clarification; Changed "assure" to "ensure" and clarified that DOE reviews will be done, irrespective of any NRC reviews for security facility approval, to ensure safe storage and handling of classified hardware.

[7] Response to Security RAI; noted that coordination between DOE and NRC for facility approval is consistent with the executive order and proposed rule.