October 19, 2001

Mr. Patrick T. Daly
BNFL, Inc.
Big Rock Point Restoration Project
10269 U.S. 31 North
Charlevoix, MI 49720-9436

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING THE MODEL NO. BRP RVP SAR-5339 PACKAGE

Dear Mr. Daly:

This refers to your application dated June 19, 2001, requesting certification of the Model No. BRP RVP SAR-5339 package as a Type B(U)-85 package.

In connection with our review, we need the information identified in the enclosure to this letter. Additional information requested by this letter should be submitted in the form of revised pages. To assist us in scheduling staff review of your response, we request that you provide this information by December 28, 2001. If you are unable to provide a response by that date, our review may be delayed.

If you have any questions regarding this matter, we would be pleased to meet with you and your staff. I may be contacted at (301) 415-8513.

Sincerely,
/RA/
Nancy L. Osgood
Senior Project Manager
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-9300
TAC No. L23336
Enclosure: Request for Additional Information
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DATE	10/11/01		10/11/01		10/15/01		10/16/01		10/17/01	

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NAME	KAGruss		ABBarto		EPEaston		BJDavis			

DATE	10/5/01	10/15/01	10/18/01	10/19/01	

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Enclosure: Request for Additional Information

Request for Additional Information

Model No. BRP RVP SAR-5339 Package
Docket No. 71-9300

By application dated June 19, 2001, BNFL, Inc. requested certification of the Model No. BRP RVP SAR-5339 as a Type B(U)-85 package. This request identifies additional information needed by the U.S. Nuclear Regulatory Commission (NRC) staff in connection with its review of the safety analysis report. The requested information is listed by chapter number and title in the safety analysis report. NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material," was used by the staff in its review of the application. This request describes information needed by the staff for it to complete its review of the safety analysis report and to determine whether the applicant has demonstrated compliance with regulatory requirements.

1.0 GENERAL INFORMATION

1-1 Revise General Notes No. 2 on the drawing for package configurations and dimensions (Figure 2-1, Sheet 5 of 7). The revised note should address the codes and standards and acceptance criteria for materials, fabrication, and methods of nondestructive examination of the reactor vessel overpack shell. Currently, the note only narrowly addresses the performance for welding.

1-2 Clarify that the girth weld joining the containment top plate to the cylindrical shell will be examined by a volumetric and liquid penetrant method (ASME Section III, NB-5231). The girth weld joining the top plate to the shell is a Category C full penetration weld and as such requires volumetric examination of the weld. Explain why this is stated as an exception in Section 8.1.2 of the application.

4.0 CONTAINMENT

4-1 Revise the radiolytic hydrogen generation analysis to provide additional information regarding the $G(H_2)$-value for low density cellular concrete (LDCC), including how the value was determined, and if it is appropriate for the LDCC.

It is not clear that the $G(H_2)$-value of 0.047 H_2 molecules/100 eV, obtained experimentally for samples of grouted radioactive waste, is appropriate for the consideration of hydrogen generation in LDCC. Show that the grout used in experiments described in Ref. 2.3 of Appendix 4-1 of the application, is similar to LDCC with respect to the characteristics affecting radiolysis (e.g., unbound water content). Note that NUREG/CR-6673, “Hydrogen Generation in TRU Waste Transportation Packages” calculates a $G(H_2)$-value of 0.49 H_2 molecules/100 eV for inorganic concreted waste containing predominantly alpha-emitting radionuclides.

The information requested above is needed to ensure that the total combustible gas remains less than 5% of the free gas volume in any confined region of the package, as stated in Section 4.5.2.3 of NUREG-1609, “Standard Review Plan for Transportation Packages for Radioactive Materials.”

5.0 SHIELDING

The following information is needed to show that the package meets the requirements of 10 CFR 71.47 and 71.51.

5-1 Revise Section 5.1.2 and Table 5-2 to address the requirements of 10 CFR 71.51(a)(2). The statements that 10 CFR 71.51(a)(2) is not applicable to the BRP Reactor Vessel Package should be removed.

The provisions of 10 CFR 71.51(a)(2) require that under hypothetical accident conditions that there is no external dose rate exceeding 1 rem/hr at 1 meter from the package surface. Since 10 CFR 71.51(a)(2) applies to Type B packages, the application should address this requirement.

5-2 Justify that the 0.52 gram sample from the core spray nozzle would adequately represent the isotopic mix found within the reactor vessel.

5-3 Provide the analysis described in Appendix 5-2, including ORIGEN2 input files, which generated a radially-averaged flat core flux of 2.49E+14 n/cm^2-s.

5-4 Provide the radial and axial ANISN input files described in Appendix 5-2.

5-5 Provide the ORIGEN2 input files used to calculate the reactor vessel activation described in Appendix 5-2.

5-6 Describe the methodology for using MicroShield to estimate the Co-60 content of the grid bar end pieces.

In Section 5, on page 5-5 of 25, it says that the methodology is discussed in detail in Appendix 1-2. However, it does not appear the methodology is discussed in Appendix 1-2.

5-7 Provide a description of the MegaShield code, including how the code was used, input files, and assumptions used.

The application states that MegaShield was used in conjunction with QAD-CGGP and MicroShield to estimate the Co-60 content for activated components, but no description of MegaShield was included.

8.0 ACCEPTANCE TESTS

8-1 Revise the application to describe how the package will be leak tested prior to shipment. The description should include details regarding how the test is performed and the test sensitivity.

Section 8.1.4 states that the package will be leak tested by a helium sniff test. It is not clear how the helium will be introduced into the package, what areas of the package will be tested for leakage, and how the presence of undiluted helium is assured at the test areas.