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

DEC 4 1984

Dr. Kamal Araj
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Dear Dr. Araj:

During our telephone conversation on November 26, 1984 and November 27, 1984, you asked the following:

1. Provide a diagram of a Mark II containment.
2. Verify the diagram of the Surry containment, which I sent previously. Focus on the reactor cavity.
3. Provide experimental data for decay heat curves, which I previously sent.
4. Provide diagrams of a core/concrete interaction showing more detail than the diagrams which I previously sent.
5. Provide specifications of the core used in Surry 1 and Peach Bottom 2 so that the data used for the BMI-2104 analyses can be verified.
6. Provide burnup figures for Surry 1 and Peach Bottom 2. The figures should be those used in the analyses and those actually found in a reactor.
7. Define the following terms; degraded core, deterministic approach, event tree analysis, external event, and rule making.

The responses to these items are given below:

Item 1. A diagram of the Mark II containment is enclosed.

Item 2. The diagram of the Surry containment, used in BMI-2104, is erroneous. The diagram is of a high pressure containment but it is not the Surry containment; the Surry containment has a flat mat, not a keyed mat as indicated; the crane is mounted on a crane wall, not the floor as indicated. These differences may influence a source term. E. Warman at Stone and Webster Engineering Corporation will send you the correct diagrams in several weeks.

Item 3. Experimental data for a decay heat curve does not exist as such. The curve is calculated using radionuclide decay series. The decay series are well known. The issue is not one of data but one of determining how many radionuclides to include in the calculations. The curves are sufficient to describe the shutdown of a reactor.

Item 4. I am compiling a set of diagrams of a core/concrete interaction with text.

Item 5. Core specifications that would be useful to verify the BMI-2104 data are difficult to obtain because the core specifications may

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change each time a reactor is refueled.

The Peach Bottom 2 reactor has 764 assemblies with fuel rods in a 8 x 8 lattice. The Surry reactor has 157 assemblies with fuel rods in a 15 x 15 lattice. This is as of November 27, 1984.

Though the designs may change from one refueling to the next, the overall power is about the same. Surface areas and composition may change slightly.

In the BMI-2104 analyses, the following fuel arrangements were used:

	# of Assemblies	Lattice
Peach Bottom	764	7 x 7 and 8 x 8 (mostly)
Surry	157	15 x 15
Zion	193	15 x 15
Grand Gulf	800	8 x 8
Limerick	193	17 x 17

The actual core configuration has little impact on the source term. Given a core configuration, the source terms are sensitive to the power distribution. For the BMI-2104 analyses, the power distribution is modelled as a chopped-cosine curve; actually, a power distribution is much flatter.

Though the core configuration is unimportant for a source term calculation, a consistent analysis is important. A single core configuration was used for each reactor to calculate source terms for the selected accident sequences.

Item 6. The table below shows the burnup figures that you requested.

	Surry 1		Peach Bottom 2
	Core Fraction	Burnup (MWd/MT)	
BMI-2104	1/3 1/3 1/3	33000 22000 11000	Average burnup is 18434 MWd/MT
Actual	31000 MWd/MT *		Predicted for next 18 months 19780 MWd/MT

* NUREG/CR-3602, Fuel Performance Annaual Report for 1982

The BMI-2104 analysis for the Peach Bottom reactor used burnup figures for the seven types of fuel found in the Brown Ferry 1 core. This was done because the information was readily available from the Oak Ridge National Laboratory.

The average burnup can be calculated using the data in table 6.6, p 6-53, Volume II, BMI-2104. For your convenience I have abstracted the data:

Number of Assemblies	Approximate Burnup (MWd/MT)
87	30400
127	23800
140	22900
23	24000
87	16600
68	16900
232	8900

Weighted average = 18434 MWd/MT.

A calculation using the ORIGEN code was done for each type of fuel. The ORIGEN results were then added together

Item 7. The commonly accepted definition of each term that you request are given as follows:

Degraded core - loss of fuel rod geometry, massive zircaloy oxidation, etc.

Deterministic approach - the standard approach to the specification and analysis of accidents in the licensing process; e.g., specification of a design basis accident followed by a prescribed, usually conservative analysis. The term is usually intended to contrast the term "probabilistic" approach used in probabalistic risk assessment (PRA).

Event tree analysis - an accident initiator with an origin external to, and not caused by, faults within the plant; e.g., earthquakes, sabotage, meteorite strikes, etc.

Rulemaking - the process for changing a regulation in the Code of Federal Regulations (CFR). The process usually involves a public hearing.

Sincerely,

Christopher Ryder

Christopher Ryder
Accident Source Term Program Office
Office of Nuclear Regulatory Research