Entergy

System Energy Resources, Inc.
1340 Echelon Parkway
Jackson, MS 39213

CNRO-2004-00065

September 30, 2004

U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001
Attention: Document Control Desk

DOCKET: 52-009

SUBJECT: Response to Supplemental Request for Additional Information (RAI) Regarding the Environmental Portion of the Early Site Permit Application by System Energy Resources, Inc. (SERI) for the Grand Gulf ESP Site

REFERENCE:

1. System Energy Resources, Inc. (SERI) letter to USNRC – Early Site Permit Application (CNRO-2003-00054), dated October 16, 2003.
2. USNRC letter to SERI – Supplemental Request for Additional Information (RAI) Regarding the Environmental Portion of the Early Site Permit Application by System Energy Resources, Inc. (SERI) for the Grand Gulf ESP Site (TAC No. MC1379) (CNRI-2004-00017), dated August 26, 2004.
3. SERI letter to USNRC – Response to Request for Additional Environmental Information Related to Early Site Permit Application (Partial Response No. 2) (CNRO-2004-00045), dated July 19, 2004

CONTACT:

Name	George A. Zinke
Mailing Address	1340 Echelon Parkway Jackson, MS 39213
E-Mail Address	gzinke@entergy.com
Phone Number	601-368-5381

In the referenced August 26, 2004, letter (Reference 2) the U.S. Nuclear Regulatory Commission requested additional information to support review of the SERI ESP Application. This letter transmits information as outlined in Attachment 1 to this letter.

Should you have any questions, please contact me.

D069

I declare under penalty of perjury that the foregoing is true and correct. Executed on September 30, 2004.

Sincerely,

George A. Zinke
Project Manager
System Energy Resources Inc.

Attachment: Attachment 1

cc: Mr. R. K. Anand, USNRC/NRR/DRIP/RNRP
Mr. C. Brandt, PNL
Ms. D. Curran, Harmon, Curran, Spielberg, & Eisenberg, L.L.P.
Mr. W. A. Eaton (ECH)
Mr. B. S. Mallett, Administrator, USNRC/RIV
Mr. J. H. Wilson, USNRC/NRR/DRIP/RLEP

Resident Inspectors' Office: GGNS

ATTACHMENT 1

SECTION 3.8, TRANSPORTATION OF RADIOACTIVE MATERIALS

Request:

E 3.8-14 The environmental impacts of the transportation of fuel and radioactive wastes to and from nuclear power facilities were resolved generically for light water reactors in 10 CFR 51.52(a) provided that the specific conditions in the rule are met; if not, *a full description and detailed analysis* is required from the applicant for initial licensing in accordance with 10 CFR 51.52(b) (emphasis added). Once licensed, the NRC may consider requests to operate at conditions above those in the facility's licensing basis; for example, higher burnups, enrichments, or thermal power levels above 33,000 MWd/MTU, 4 percent, and 3800 MW(t), respectively. The rule has not been changed for the initial licensing of nuclear power facilities, and departures from the conditions itemized in the rule that were found to be acceptable for licensed facilities cannot serve as the basis for initial licensing. Unless the applicant uses a plant parameter envelope for considering transportation impacts, each reactor must be considered separately.

Provide a transportation risk assessment for all proposed reactor spent fuel shipments using an accepted methodology such as RADTRAN V. Provide justification that the best available information has been used to generate the RADTRAN input values and that those values are appropriate. Provide a comparison of the results of that assessment with the spent fuel shipment risk estimates contained in NUREG-0170, *Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes.*

Response:

ABWR, AP1000, ACR-700

RADTRAN V spent fuel highway runs were conducted for three LWR technologies: ABWR, AP-1000, and ACR-700, which were determined to be representative of all of the LWR technologies that form the basis of the plant parameter envelope. The analysis assumed that the spent fuel truck shipments were from Maine Yankee Nuclear Plant location to Yucca Mountain. The TRAGIS Routing Engine Version 1.4.15, which uses the 2000 Census data, provided the routing information and the population densities. The input values were taken primarily from the Yucca Mountain Final EIS, in particular the *Transportation Health and Safety Calculation/Analysis Documentation in Support of the Final EIS for the Yucca Mountain Repository.* Specifically the values for the BWR and PWR accident severity and release fractions were taken from Tables 5-24 and 5-25 of the aforementioned reference. For the ACR-700, the PWR input values were used. The results of these runs are provided in the output files that echo the input files as well. These files are provided as separate attachments. The comparisons of the incident free results with NUREG-0170 are shown in the following table.

Table 1:
LWR RADTRAN V Incident Free Analysis Results as Compared to NUREG-0170

	NUREG-0170 (1) (person-rem/shipment)	ABWR (2) RADTRAN V results (person-rem/shipment)	AP-1000 (2) RADTRAN V results (person-rem/shipment)	ACR-700 (2) RADTRAN V results (person-rem/shipment)	Difference between RADTRAN V results and NUREG-0170
Passengers	0	0	0	0	0
Crew	0.123	0.157	0.157	0.157	0.034
Handlers	0.200	0.188	0.188	0.188	-0.012
Off-Link	0.015	0.012	0.012	0.012	-0.003
On-Link	0.007	0.081	0.081	0.081	0.073
Stops	0.019	0.190	0.190	0.190	0.171
Storage	0.005	0	0	0	-0.005
Totals	**0.369**	**0.628**	**0.628**	**0.628**	**0.259**

(1) *Based on 1530 spent fuel truck shipments for the year 1985*
(2) Spent Fuel shipment from Maine Yankee location to Yucca Mountain

The results are the same for the three reactors since the incident-free dose to a receptor is independent of the isotopic contents of the cask, depending only on the dose rate external to the cask, which was set at the regulatory limits.

The major differences between the NUREG-0170 and RADTRAN results are attributed to the dose incurred during stops ("stop dose"), and to a lesser degree the dose to the crew and the on-link dose. Approximately 42% of the stop dose difference is attributable to the RADTRAN V simulations which included inspections at the beginning and the end of the trip; NUREG-0170 did not include these inspections. The remaining difference can be attributed to the greater distance traveled, hence more refueling stops, and the different methodologies used to calculate the stop doses. This evaluation used 1996 truck stop data (*Investigation of Radtran Stop Model Input Parameters for Truck Stops,* SAND96-0714C) and modeled public doses in two concentric rings: 1 m to 14 m and 30m to 800m. The population in the inner ring used the results of the Stop Model study while the population in the outer ring used route specific 2000 Census population data weighted by a 3% urban, 26% suburban and 71% rural distribution. The NUREG-0170 study modeled just one ring, 10 to 2600 feet, and used three fixed population densities.

Factors contributing to the greater crew dose include the greater distance traveled and more refueling stops.

Factors contributing to the increased on-link population dose are the result of NUREG-0170 assuming a 2500 km shipment distance with a 5% urban, 5% suburban and 90% rural population versus this evaluation using updated 2000 census information showing a 3% urban, 26% suburban and 71% rural population and a 4,733 km shipment distance.

In addition to the incident free results, the RADTRAN V runs also included accident results. A comparison with NUREG-0170 is shown in the table below.

Table 2: Accident Risk Analysis Results

	Table 5-10, NUREG-0170, 1985	ABWR, RADTRAN V	AP-1000, RADTRAN V	ACR-700, RADTRAN V
Person-rem	NA	1.13E-05	2.42E-07	1.21E-07
Latent Cancer Fatalities (1)	0.29	1.78E-02	3.82E-04	1.91E-04

(1) The conversion from the RADTRAN V person-rem results to latent cancer fatalities was completed using the dose conversion factor 6.3E-4 rem/LCF contained in the report BEIR V "Health Effects of Exposure to Low Levels of Ionizing Radiation"

The RADTRAN V results are 16 times (for the ABWR) to over 1500 times (for the ACR-700) lower than the results presented in NUREG-0170.

<u>GT-MHR, PBMR</u>

Transportation risk assessment values for the GT-MHR and PBMR were previously provided in response to RAI 3.8-4 in cover letter Reference 3. For consistency, RADTRAN V was rerun for the GT-MHR and PBMR using the same incident free inputs as the latest LWR runs; a revised response is included below.

RADTRAN V highway runs were conducted for a GT-MHR and a PBMR spent fuel shipment from Maine Yankee Nuclear Plant to Yucca Mountain. The TRAGIS Routing Engine Version 1.4.15, which uses the 2000 Census data, provided the routing information and the population densities. The analysis was conservative using the 10 CFR 71 regulatory limits of 2 mrem/hr in the cab and 10 mrem/hr at 2 meters from the cask. The input values were taken primarily from the Yucca Mountain Final EIS in particular the *Transportation Health and Safety Calculation/Analysis Documentation in Support of the Final EIS for the Yucca Mountain Repository.* Specifically, the values for the high integrity high-temperature gas-cooled reactor spent nuclear fuel referred to as type 8 were used. A comparison of the incident free results with NUREG-0170 is provided in Table 1 below.

Table 1:
LWR RADTRAN V Incident Free Analysis Results as Compared to NUREG-0170

	NUREG-0170 [1] (person-rem/shipment)	GTMHR [2] RADTRAN V results (person-rem/shipment)	PBMR [2] RADTRAN V results (person-rem/shipment)	Difference between RADTRAN V results and NUREG-0170
Passengers	0	0.000	0.000	0.000
Crew	0.123	0.157	0.157	0.034
Attendants	0.000	0.000	0.000	0.000
Handlers	0.200	0.188	0.188	-0.012
Off-Link	0.015	0.012	0.012	-0.003
On-Link	0.007	0.081	0.081	0.074
Stops	0.019	0.190	0.190	0.171
Storage	0.005	0.000	0.000	-0.005
Totals	**0.369**	**0.628**	**0.628**	**0.259**

[1] Based on 1530 spent fuel truck shipments for the year 1985
[2] Spent Fuel shipment from Maine Yankee location to Yucca Mountain

The major difference is the dose during stops. Much of this difference is attributable to the RADTRAN V simulations which included inspections at the beginning and the end of the trip; NUREG-0170 did not include these inspections. The remaining difference can be attributed to the greater distance traveled, hence more refueling stops, and the different methodologies used to calculate the stop doses. This evaluation used 1996 truck stop data (*Investigation of Radtran Stop Model Input Parameters for Truck Stops,* SAND96-0714C) and modeled public doses in two concentric rings: 1 m to 14 m and 30m to 800m. The population in the inner ring used the results of the Stop Model study while the population in the outer ring used route specific 2000 Census population data weighted by a 3% urban, 26% suburban and 71% rural distribution. The NUREG-0170 study modeled just one ring, 10 to 2600 feet, and used three fixed population densities.

Factors contributing to the increased on-link population dose are NUREG-0170 assumed a 2500 km shipment distance with a 5% urban, 5% suburban and 90% rural population. This evaluation used updated 2000 census information showing a 3% urban, 26% suburban and 71% rural population and a 4,733 km shipment distance.

In addition to the incident free results, the RADTRAN V runs also included accident results. Due to the preliminary nature of the gas-cooled reactor fuel design, it is premature to provide a meaningful comparison with NUREG-0170. The RADTRAN V runs were made with the gas-cooled fuel values in the Yucca Mountain FEIS. Specifically, the values for the high integrity high-temperature gas-cooled reactor spent nuclear fuel referred to a type 8 were used. As such, these runs provide a reasonable estimate of what the GT-MHR and PBMR results might look like. It is important to remember that the gas-cooled reactor spent fuel shipments are no different from other spent fuel shipments in that all shipments are required to meet NRC and DOT regulations. These regulations address design and performance standards for the casks and specify radiological performance criteria for both normal transport and severe accident conditions. Compliance with these regulations is mandatory and ensures that shipments will be conducted in a manner that results in minimal environmental impacts.

GENERAL

The incident-free results presented above are the same for the five reactor types since the dose to a receptor is independent of the isotopic contents of the cask, depending only on the dose rate external to the cask, which was set at the regulatory limits. As discussed above, the environmental impacts for both incident-free and accident scenarios are dependent on the cask design and performance standards and dose limits established in NRC and DOT regulations. Compliance with these regulations ensures the shipments will be conducted in a manner with minimal environmental impacts. The Early Site Permit application did not select any specific technology(s) but used a Plant Parameter Envelope type approach to postulate bounding and/or representative environmental impacts. The "plant parameter envelope" equivalent values important to the incident-free analysis would be the regulatory dose limits used in the analysis.