Project No. 716

February 25, 2003

Laurence L. Parme
Manager: GT-MHR Safety and Licensing
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P.O. Box 85608
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SUBJECT: NRC STAFF QUESTIONS RELATED TO THE GENERAL ATOMICS PRESENTATIONS AT THE JANUARY 28 -29, 2003, GAS TURBINE-MODULAR HELIUM REACTOR PREAPPLICATION REVIEW MEETING (PROJECT 716)

Dear Mr. Parme:

On January 28 and 29, 2003, representatives from General Atomics (GA) met with the NRC staff to discuss preapplication activities related to Gas Turbine-Modular Helium Reactor (GT-MHR) source term and fuel quality specifications. The objective of the meeting was to familiarize NRC staff with the GT-MHR approach to source term, and fuel quality and performance basis. The meeting was conducted to support the staff's GT-MHR preapplication review activities related to these technical topics. During the meeting, GA representatives made a series of presentations related to these topical areas and responded to staff questions. The staff will base its GT-MHR preapplication review on formal correspondence and documents (e.g., reports, white papers, responses to requests for information) provided by GA. In this regard, the staff asked a number of technical questions during the meeting. Per your request these questions are enclosed. If you would like GA's meeting responses to be included in the basis for the staff's review, it will be necessary that GA formally document and submit its responses to the enclosed staff questions.

Additionally, significant discussions centered on GA's approach to the treatment of parameter uncertainties. Methods reviews are not currently included in the GT-MHR preapplication review scope, however, NUREG/CR-5249 is available in ADAMS, ML030380473.

The reporting and/or record keeping requirements contained in this letter affect fewer than ten respondents; therefore, OMB clearance is not required under P.L. 96-511.

Please contact me (301-415-7499) or Stuart Rubin (301-415-7480) if you have any questions.

Sincerely,

/RA/

Farouk Eltawila, Director
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research

Enclosure: As stated

cc w/encl.: Standard Service List Addresses

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Letter dated: 2/25/03

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NRC Staff Questions and Comments
from the
NRC/GA Meeting on the GT-MHR Pre-Application Review
January 28-29, 2003

NRC Questions

Slide 1-5:

How is the effectiveness of the GT-MHR Helium Purification System (HPS) considered in the mechanistic source term analysis. Are there any source term related DDNs for the HPS. Does GA consider the effectiveness of the HPS outside the scope of the staff's preapplication review for the source term logic, etc.?

Slide 1-6:

Is indicated that, for standard particles, there are negligible particle failures during accidents. This view appears to be different from recent analyses of irradiation induced TRISO particle failures. A contemporary view and analysis of TRISO particle test failures suggests that the statistical variation of particle properties (e.g., thickness, tensile strength, modulus) can result in a non-significant rate of operations-related failures. This is in addition to the failure rate from defective particles. Explain the basis for assuming negligible particle failures for standard particles.

What is the basis for assuming that as manufactured coating defects (e.g., missing buffer layers) will be a primary cause of particle failures? It appears that such manufacturing defects may be a vestige of historical US manufacturing experience. What is the basis for assuming that an improved manufacturing process that GA intends to develop will continue to exhibit the significant numbers of the types of particle defects of the sort assumed for the fission product release models?

Slide 1-11:

Is the reactor building volume a design-analysis consideration with respect to limiting oxygen ingress

Slide 1-13

Do the events shown on this page serve as the basis for framing the scope of the fission product release mechanisms and hence the scope of the DDNs? For example, are oxidation events due to a double ended break in the cross connection vessel or reactivity addition events due to control rod ejection accidents part of the licensing basis? How will sensitivity studies to assess the potential consequences of such an event be carried out if fission product transport property event data is not developed?

Enclosure

Slide 2-3

What is the basis for knowing the extent of condensible radionuclides? Is it calculated? Can it be verified by measurement during operation?

Slide 2-4:

To what extent are “non-dominant” radionuclides accounted for in the offsite dose calculation?

Slide 2-7:

Are changes in the fuel kernel microstructure due to burnup explicitly accounted for in the fission product release rate model and data needs? For example, does the diffusion coefficient separately account for changes in kernel gas bubbles, grain size or is an effective kernel diffusion coefficient versus burnup utilized?

Slide 2-8

Why isn’t burnup included among the “controlling parameters?” Why aren’t pressure induced failures considered?

Slide 2-9

If a filter was provided to mitigate initial blowdown and longer term releases, what would be the effect on reactor building radionuclide releases?

Slide 2-21

There are potentially many different design, manufacture, operational and accident-related causes that contribute to exposed kernels. What is the basis for the position that missing buffer layers from manufacture will be the dominant cause of exposed kernels, considering that fuel has yet to be manufactured and tested and examined for causes particle failures?

Slide 2-15

Fuel performance is evaluated against Part 100 (50.34) and the EPA PAG. However, GA refers to the PAG as a “utility/user requirement.” Does GA expect that the PAG dose level would also become part of the facility licensing basis?

Slide 2-30

NUREG-1338, Preapplication Safety Evaluation Report for the MHTGR and selected supporting technical evaluation reports and letter reports included statements on DDN-related needs with respect to the MHTGR source term analysis. Were all of these comments addressed in the DDN assessment for the GT-MHR source term? Explain.

Slide 2-31

The NRC is in the process of completing a Phenomena Identification and Ranking Table (PIRT) for fuel integrity and fuel fission product transport. The NRC PIRT addresses the factors in the areas of design, manufacture, operations and accidents effecting fuel integrity and fuel fission product transport. For each of these areas the PIRT identifies the specific individual factors associated with the kernel, each particle layer, the fuel matrix and nuclear graphite effecting fuel integrity and fuel fission product transport. Was a PIRT conducted for the GT-MHR fission product transport analysis? If not, explain the basis used for identifying the important factors in the GT-MHR fission product transport and for identifying the related DDNs.

Slide 2-31

The logic/approach appears to take credit for each and every significant barrier/hold-up/delay associated with the transport of fission products between the fuel kernel and the site boundary. Explain how the principle of “defense in depth” for fission product barriers relates to the proposed logic/approach to defining fuel requirements.

Slide 3-5

Are the codes (e.g., SORS) and methods that will be used for the GT-MHR fuel performance (integrity) and fission product transport analysis the same (with improvements) as those that were developed for the MHTGR?

Slide 3-6

Will the fertile particle be UO_2 or UCO? Explain?

Slide 3-9

Are the mechanistic models for calculating fuel performance (integrity) and fuel fission product transport based on basic engineering principles (e.g., stress/strain analysis) or empirical correlations or both? Explain.

Slide 3-10

Is it expected that the fuel development program will succeed in fuel quality and performance objectives being achieved predominantly or completely through the specification of product (e.g., material property) requirements, or is it expected that significant number of fuel process requirements will need to be included to achieve these objectives? Explain.

Slide 3-13

Why is kernel microstructure (e.g., grain size, pore structure, grain orientation, phase homogeneity not included as an important property?

Slide 3-17-19

Why are missing IPyC, SiC, OPyC not considered important properties while a missing buffer layer is considered an important property?

Slide 3-12

German experience with different matrix materials shows that the source of the matrix materials can alter the diffusion coefficient by an order of magnitude. Why isn't the source of the raw materials for making the compact considered important properties?

Slide 3-38:

Will GA be integrating the SPC (statistical process control) program with the outgoing fuel acceptance testing program or will the programs be completely separate? If so, in what way?

Slide 4-4:

What data source will be used to assess the effects of postulated reactivity events (rapid and large energy depositions) and air ingress events (extended oxidation) beyond the design basis?

Slide 4-12:

Which properties (e.g., conductivity, strength, elastic modulus, creep coefficients) will rely on existing properties data and which properties will involve DDNs? What is the criteria for using existing data for the fuel design and fuel performance analysis models? Sensitivity studies? PIRT? Variations seen in the historical properties data bases?

Slide 4-19:

Describe /define what is meant by the statement that "design methods used to predict fuel failure during normal operation are accurate to a factor of 4."

Slide 4-22:

Will the irradiation capsule monitoring instrumentation be able to detect individual particle failures as they occur? Explain.

Slide 4-23:

What are the units for fission gas release?

Slide: 4-26

The higher particle powers, temperatures, fast fluxes over the much shorter times associated with an accelerated irradiation may bias against potential particle degradation and failure mechanisms (e.g., time-dependent fission product reactions with the SiC) that might otherwise be revealed in a real time irradiation. Will the test conditions involve some "real-time" irradiations as well as "accelerated" irradiations?

Slide 5-4

Can the GA codes and methods descriptions be made available to the NRC to assist the NRC staff in obtaining additional insights that would be useful to the NRC staff in developing its infrastructure of independent NRC HTGR fuel performance and fission product transport analysis tools (Not for review). Are they publicly available?

Do the codes provide best estimate or conservative predictions. If the codes are best estimate, how do you account for uncertainties and design margins. Have the best estimate codes been validated?

Slide 5-16:

Does GA plan to use any of the existing data? What is the effect of UCO versus UO_2 on the applicability of the existing data base. How does GA plan to deduce from a mixed (UO_2 and UCO) data base, fission gas release specifically for UCO fuel?

Slide 5-17:

For HFR B1, did R/B data get collected at different burnups to account for the effects of changed kernel micro-structure at increased burnup?

Slide 5-19

Will the SiC layer diffusion properties in the GT-MHR fuel particle be adjusted by manufacturing processes so as to lower the Ag diffusivity or will the SiC layer diffusion properties be a byproduct of the design and manufacturing efforts to optimize the in-service SiC layer strength and performance integrity?

Is irradiation testing conducted under thermal gradient conditions for C.07.03.03 FP Effective Diffusivities in Particle Coatings? Explain.

Slide 5-22

German experience with different matrix feed sources showed that the diffusivity for metals can change by an order of magnitude depending on the petroleum pitch. Will the matrix diffusivity data be revisited and revised if and when new feed sources are needed.?

Will GA use the (e.g., Japanese) H451 fine grain graphite, feed sources and attendant diffusion, sorption data base? Explain.

Slide 5-25

With respect to the need for RN deposition characteristics, is that an occupational dose issue?

Slide 5-27

Can control rod motion within the prismatic graphite blocks be a source of dust in the GT-MHR design?

Slide 5-27

Will re-entrainment DDNs include helium fluid velocity conditions representative of a double ended break in the horizontal cross connect vessel? If not, what re-entrainment data base(s) will be used for conducting sensitivity studies of the effects of larger GT-MHR breaks on offsite dose?

Slide 5-32

Will the SURVEY and SORS models be revised to reflect the validation failure data specific to the GT-MHR design?

Slide 5-34

How will you know the failure fraction?

Slide 5-34

Is there any intent or plan to utilize a fission product transport data from currently operating HTGRs to benchmark or validate GA's fission product transport models, codes and methods?

A factor 4 or 10 has been targeted for the accuracies of all models, codes. How will these factors be used in the integrated methods? How will these accuracies be propagated to get to a 95% confidence statement?

<u>NRC Comments</u>

The examples shown during the presentations utilize 10 CFR 100 and EPA protective action guidelines (PAGs). The PAGs and 10 CFR 100 have changed since the examples were developed. For example, per 10 CFR50.34 GA would need to show that 25 rem total effective dose equivalent (TEDE), as defined in 10 CFR 20.1003 at the exclusion area boundary (EAB) and the low population zone (LPZ) and 5 rem TEDE for the control room. The EPA PAG is now 1 rem TEDE.

We understand that GA intends to use a 95% confidence level based on manufacturing tolerances, core design parameters, accident behavior and other factors. While it may require only a few hundred samples to know the 50% level with confidence many more samples are required at the 95% confidence level even for well behaved distributions. As part of an actual application the staff would expect that sample size and distributions will be addressed and justified in the analysis and application of the data.

We referred to the code scaling, applicability and uncertainty (CASU) evaluation methodology. A description of the CASU evaluation methodology is contained in NUREG/CR-5249, "Quantifying Reactor Safety Margins, Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large Break, Loss-of-Coolant Accident," dated December 1989.