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TECHNICAL EVALUATION REPORT

EVALUATION OF THE DOE STANDARD MHTGR CONTAINMENT DESIGN ALTERNATIVES

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I. Introduction

The purpose of this study is to summarize containment-related RES-sponsored work done prior to FY 1993, and to reply to specific NRC requests for assessments of the DOE Containment Study (DOE-HTGR-88311). Our evaluations of the containment issues were formulated as a result of a series of broad-based information meetings with DOE, NRC, and their consultants and contractors, and numerous discussions and correspondence. The specific NRC questions about HTGR-88311 were originally formulated by NRR/PDAR (J. N. Donohew).

II. Summary of Containment-Related Previous Work (FIN A9477)

1. NRC Policy and Positions on Containment Building Design for Advanced Reactors - General Issues

NRC is reviewing four advanced reactor concepts (ARCs): the MHTGR, ALMR, PIUS, and CANDU-3. Of the four, only the MHTGR and ALMR designers require relief from a strict adherence to the LWR-based GDC-16 requirements for an essentially leak-tight containment building (CB). The current design of the ALMR containment, while designated "leak tight" (1%/day), is not conventional in that it is not a "building," but rather is a vessel in close proximity to the reactor vessel. Its volume is therefore relatively small, and the tight coupling and common penetrations between the two increase the probability of simultaneous failures.

Most of the ARCs base their economics (in varying degrees) on "passive safety" design features, and in two cases, the CB issue could have a major impact on passive safety features. The natural-circulation air-cooled reactor cavity cooling system (RCCS) of the MHTGR, and the similar RVACS of the ALMR, for example, could probably not be used in conjunction with a conventional leak-tight building as the containment.

The NRC would be expected to allow some latitude here, in keeping with the NRC's Advanced Reactor Policy Statement. It is not clear, however, how rigid the NRC would or should be in defending a position allowing waivers to GDC-16 in the face of concerns that very-low-probability accidents might lead to doses in excess of 10 CFR 100, or even in excess of the lower-dose EPA protective action guidelines (PAGs) for sheltering limits at the exclusion area boundary (EAB).

The timing of releases is recognized as being very important. In several ARC designs, it is claimed that IF any major very-low-probability accidents were to occur, releases would not appear at the EAB for several days, thus allowing adequate time for ad hoc mitigation action and/or evacuation. The crucial point is that ARC designers must first convince NRC that the probability of a significant EAB dose is extremely small if prompt, and not very much more likely even if the dose is delayed.

There have been several solutions proposed for resolving the problem of specifying adequate containment (or confinement) structure capabilities needed to meet anticipated operational occurrences (AOO), design basis accidents (DBA), and beyond DBA (BDBA) events' 10 CFR 100 dose guidelines. Some appear to be "reasonable," but present some very difficult practical problems for licensing and verification. For example, one recommended solution is to require an analytical "demonstration" of the capability for BDBA events to meet 10 CFR 100 dose guidelines at the EAB, assuming sub-specification fuel. The problem with this approach is that there are the compounded difficulties of dealing with both the extremely low probabilities ($<10^{-6}$/yr) for failures of passive decay-heat removal systems such as the RCCS, and with the vague concept of sub-specification or "weak" fuel. Another recommendation is to require building and operating a demonstration plant to verify the CB design; however, it appears somewhat naive to expect a prototype or demonstration project to resolve questions about problems that might arise every few million plant years.

If NRC were to implement a plan to determine the requirements of major testing programs during the final design review, based in part on the results of supporting R&D programs, it would not be sufficiently responsive to the needs of the designers and plant owners. For example, if the NRC were to make a conservative decision at that point that a very expensive multi-year testing program is required, it could easily kill that program in particular and the incentive for U.S. industry to develop advanced, safer reactors in general.

2. Review of Foreign Regulatory Policies on Containment

The nuclear regulatory body in the United Kingdom, the Nuclear Installations Inspectorate (NII), mandates that a low-leakage CB should be provided unless adequate protection is provided otherwise. Waivers have been granted in the case of the Advanced Gas Reactors (AGR), as they do not have unvented CBs. NII also allows for the use of containment pressure relief systems provided that they can be shown to have a "useful safety advantage.[1]"

The German regulatory policy is not "uniform" since each German state maintains its own licensing entity. However, the German equivalent to the USNRC's ACRS (the "RSK") has recommended that a sealed CB should not be required for the HTR-MODUL (MHTGR) design[2]. For the German HTR-2 design, the "confinement envelope," in normal operation, has no filtering. In case of a major depressurization, the HTR-2 discharge is vented to the stack via dampers; and in a slow depressurization, the discharge is passed through iodine filters. The two German HTGRs, THTR and AVR (both in the process of decommissioning), had no sealed CBs.

3. Possible Action Scenarios for Three Different Approaches to the Containment Building Problem:

Situation A: Assume a Conventional Containment Building Is Required.

1. All "events" requiring a conventional CB are of such low probability that the source term is not readily definable. That is, all postulated accident sequences that can be defined mechanistically result in small releases to the inside of the CB. Therefore, in order to determine the CB requirements, it would be necessary to arbitrarily assume the fuel failure fraction that needed to be contained and the circumstances attendant with its release.

2. Since the source and probability are not readily definable, calculations of the "dollars per person Rem averted" for a CB cost-benefit analyses would not be feasible.

3. If a CB were required (arbitrarily) for an MHTGR, it should not be required to be as "capable" of retaining massive fission product releases as those required of the conventional LWRs. Hence an entire new set of CB requirements and regulations (construction, cooling,

testing, etc.) would have to be developed for a fictitious source term. In general, it would be difficult to draw comparisons for LWR and MHTGR containment concepts, since:

- LWR CB = Big release X Large attenuation
- MHTGR "confinement" = Small release X Little attenuation

Situation B: Assume a Containment Building Is Not Required.

1. A good case could be made for covering all of the MHTGR's severe (and therefore unlikely) accidents by ad hoc mitigation measures, since in almost all cases any need to contain fission products would come late (several days) in the accident sequence. [Exception: a depressurization which occurs if/when there is a high circulating activity. This is the key point in the "weak fuel" issue, which involves both dry and wet depressurization accidents.] This might be made to be an acceptable argument as long as the ad hoc measures were accommodated in the design, tested for feasibility, and included in the emergency operating procedures.

2. There are some "lingering uncertainties" about the need for a CB to contain graphite fires. These fears could be dispelled by testing:

a. Unnecessarily pessimistic assumptions are made about a ruptured vessel's free access to clean air deep in the silo. Tests in silos could establish relative burn rates for underground vs. normal situations. Features of a proposed silo test would include mocking up the MHTGR silo air-access flow paths (both normal and disrupted), with upper bounds placed on cavity air access via realistic scenarios for a leaky reactor cavity cooling system (RCCS); and

b. Mechanisms for "complete" functional failures of the RCCS should be investigated in more detail. (Currently, such a failure appears to be non-mechanistic, but more detailed design information about the RCCS would be needed to assure this.)

Situation C: Treat the CB Problem as a Political One.

1. Acknowledge that the CB issue is more political than technical, where public education and debate would be beneficial.

2. The points to be made in "debates" are that: a) reactors with a high degree of inherent safety should not be subjected to arbitrarily stringent regulations; b) inherent safety features can be demonstrated; c) the overall societal risks involved can be shown to be less than those for alternative forms of electrical energy supplies; d) a CB might degrade rather than improve safety, since a passive and desirable air-cooled RCCS may not be possible with a conventional CB; and e) there is concern that an NRC hard regulatory line (on the CB issue, for example) may scuttle the MHTGR concept (economically) and thus force the U.S. to adopt less attractive energy supply alternatives.

3. Possible forums and audiences: Workshops for public officials (federal, state, and local governments), Public Utility Commissions, the media, and the "general public;" press releases and articles in the popular press (our local Oak Ridge newspaper editor is interested in the

idea, for example); and interviews with knowledgeable NRC personnel on "good" TV shows such as Nightline, 20/20, etc.

4. EC-III & IV probabilities are so low that sabotage and external events, as accident initiators, would play a major role in the overall risk assessment. Since the severe versions of these initiators are of very low probability and difficult to quantify, decisions on how to factor them in would be mainly political.

4. Responses to NRC Containment Study Questions (Ref. P.M. Williams Letter of April 13, 1989)

a. Undetected bad fuel batch (weak fuel particles): Since the particle coatings constitute the primary barriers against the release of MHTGR fission products, an inspection and testing program of coated fuel particle integrity should be required that is functionally equivalent to the intent of the requirements for LWR containments [10 CFR Part 50.54(0) and Part 50, Appendix J]. Micrographic and other appropriate surface and volumetric inspection techniques are required of statistically valid samples from fresh fuel batches. Samples should include both the newly fabricated loose coated particles, particles in various stages of fabrication, and the final product of the fuel sticks inserted into the prismatic elements. These inspections should verify the assured physical integrity of the particle coatings at all stages of fuel fabrication and the degree of heavy metal contamination external to the coatings.

In addition, prior to exposure in the reactor, batch samples should be subjected to (accelerated) irradiation in research reactors or other in-pile test facilities, and then subjected to post irradiation heatup testing at least to 1600-1700° C to quantify the functional integrity of the particle coatings after irradiation. Also, a post irradiation fuel surveillance program should be implemented for depleted elements from the MHTGRs to verify by inspection and heatup testing both the physical and functional integrity of the particle coatings. The inspection and testing program should achieve the same level of confidence in the continued integrity of the particle coatings as is proposed for FRG's pebble bed HTR, for which heatup functional testing is to be performed on randomly selected irradiated pebbles that are continuously being defueled. Sampling and testing requirements could be adjusted depending on the learned likelihood of substandard batches.

Continuous on-line monitoring of the MHTGR circulating radioactivity in the coolant should be used to detect unexpected failures of particle coatings at normal operating temperatures, and the helium purification system and the liquid nitrogen system should be subject to quality control, surveillance and maintenance as the functional equivalent to a containment atmosphere cleanup system for accommodating unexpected particle coating failures during normal operations.

b. Unrecoverable and total RCCS failures: Analysis has shown that the RCCS can function to prevent both fuel damage and excessive temperature of the reactor vessel with as little as 10% of normal flow; these analysis should be confirmed by verification of the analytical models. The analysis of total loss of RCCS flow has shown that the passive heat transport to the cavity wall and surrounding earth is effective in preventing fuel damage but may not assure vessel integrity for long time periods. Ad hoc measures to seal the silo or to provide enhanced cooling mechanisms should be shown to be sufficient either to prevent gross vessel failure and subsequent damage to fuel elements, or to inhibit releases from delayed fuel failures.

The presence or absence of a CB for a disaster that would entail a "complete" structural failure of the RCCS (and perhaps the reactor vessel and reactor cavity components as well) would probably not make any difference in mitigating the effects, since the CB would probably fail as well.

c. <u>A prompt critical event causing fuel failure and other disruptions</u>: Rod ejection should be precluded by physical design. Water ingress events are not likely to cause prompt critical excursions even under the worst-case assumptions, but more verification is needed for water/steam reactivity relations, especially, to better bound the results of the analyses. [Ref. O. L. Smith, "Magnitude and Reactivity Consequences of Moisture Ingress into the MHTGR Core," NUREG/CR-5947, ORNL/TM-12237, in publication.]

d. <u>Reactor cavity failures (vessel support, concrete integrity, unpredictable consequences)</u>: A comprehensive failure modes and effects analysis (FMEA) should be performed and utilized to develop a preservice and inservice inspection and testing program. Since such failures are likely to be probabilistically at such a low frequency of occurrence as to be beyond design basis. Since such failures would probably tend to enhance heat transfer from the vessel to surrounding structure, and since the analysis of the least optimum configurations for RCCS failure indicates that fuel damaging temperatures are not reached, ad hoc provisions for sealing or cooling the cavity appear to be appropriate. Again, for such extreme disasters, a CB may not survive.

e. <u>Catastrophic reactor vessel failure</u>: Per the findings of the ACRS in 1974, preservice and inservice inspections consistent with ASME Code Sections III and XI are required on current generation LWR vessels. These inspections assure that the frequency of LWR vessel ruptures of a size and location that can exceed the capability of the ECCS is less than 10^{-7} per reactor year. Equivalent inspection requirements for the MHTGR vessel are judged to be sufficient to assure that "catastrophic" vessel failures are at a frequency beyond the MHTGR design basis events, namely less than 5×10^{-7} per reactor year. However, less than catastrophic failures of the MHTGR vessel, such as through-wall rupture or major penetration failures that would be unacceptable for LWR vessels, would not lead to major rearrangement of the MHTGR fuel elements and would not preclude MHTGR vessel heat removal either via the RCCS or to the surrounding structures and earth. Thus, fuel damage is not expected for MHTGR vessel failures that could occur at frequencies well beyond 10^{-7} per reactor year.

f. <u>Graphite fire</u>: The resolution of the graphite fires issue may require experiments as noted in our earlier proposals. Public acceptance may play a large part here in that two fires have occurred in graphite reactors without a CB (and were caused by deliberate, but incorrect, reactor operator action), although the reactor designs are very much different.

g. <u>Major operator errors</u>: This appears to be a non-problem in the current MHTGR design, although the case can be made that in some beyond-DBA circumstances, operator action would eventually be needed for manual scram or depressurization functions.

h. Water ingress reactivity insertions: Water ingress issues need further investigation. The predictions of reactivity increase due to both large and small amounts of water in the core need verification. A preliminary computation of a large water ingress accident predicts that portions of the fuel may fail very early in a simulated transient with a possibility of radioactivity release. Whether or not a CB could mitigate a large water ingress accident requires further analysis, since the combustible gases of hydrogen and carbon monoxide generated by the steam-graphite chemical reaction may damage the CB if sufficient oxygen is available.

5. Summary of Comments, Concerns, and Opinions (not necessarily those of NRC) Developed Previously in the RES (A9477) Program:

1. All of the postulated accidents that would require a "leak-tight" Containment Building (CB) are of such a low probability that:

   - The mechanistic source term is not readily definable;
   - CB design requirements are unclear; and
   - Therefore, cost-benefit analyses are questionable.

2. A CB's need to contain fission products (FPs) comes LATE in unlikely accidents:

   - Exception: a depressurization which occurs if/when there is a high circulating activity. This is the key point in the "weak fuel" issue, which involves both dry and wet depressurization accidents.
   - Ad hoc mitigation of fission product releases, which may be feasible with a "no-CB" design:
     * Manually operated relief valves or louvers
     * Roll-up (ad hoc) filters
   - Ad-hoc sealing of the CB should not impair functioning of RCCS

3. Some uncertainties about the need for a CB would be resolvable by R&D, including testing and experimental R&D tasks:

   - R&D on the weak fuel and hydrolysis concerns, including a satisfactory fuel QA/QC program (most significant issue)
   - R&D to validate models for the liftoff and washoff of FPs in the primary system assumed during depressurization accidents
   - Relative graphite burn rates for underground vs. normal atmospheric conditions (Silo tests)
   - Proposed graphite oxidation experiments (ORNL, Sandia/NPR)
   - Cavity air access via leaky RCCS
   - Mechanisms for "complete" functional failure of RCCS; means for RCCS repair/restoration of function following incidents which could damage it.

4. Inherent difficulty in making comparisons between MHTGRs and LWRs for containment and CB concepts:

- LWR "leak-tight" CB - Big release X Large attenuation
- MHTGR "confinement" - Small release X Little attenuation

5. The CB issues are as much political as technical:

- Event categories EC-III & IV probabilities are so low that sabotage and external events play a major role. Decisions would be mainly political.
- Need public education and debate on risks and CBs.
- Need to get public (& PUCs, State/local govt.) acceptance.
- Policies relating to emergency planning (evacuation planning) must be developed in conjunction with the CB design.
- Licensing a "special" CB would be time-consuming and costly.
- An NRC hard line may scuttle the concept and lead to less-safe reactors and less-environmentally-attractive coal-fired plants.

6. Because the water-cooled RCCS (possibly required if CB is "leak-tight") is less passive, harder to maintain, test, and license than the air-cooled RCCS, and it is the controlling factor in the PRA conclusions, it is likely to result in a net degradation of overall safety (vs. air).

REFERENCES

1. Health and Safety Executive. "HM Nuclear Installations Inspectorate Safety Assessment Principles for Nuclear Power Reactors." In Nuclear Safety, London; Her Majesty's Stationery Office, 1982 (Amended Jan. 1989).

2. Prof. Hubertus Nickel, et al, "Recommendation on the Safety Concept of a Modular High-Temperature Power Plant," translated by B. Elkendorf and I. Weisbrodt, HTR-GmbH, March, 1990.

III. ASSESSMENT OF DOE-HTGR-88311 (CONTAINMENT STUDY FOR MHTGR)

1. Evaluation of Containment Study Assumptions

Introduction In DOE's Containment Study (DOE-HTGR-88311), an assessment of containment alternatives lead to several reactor or containment building design options. These were described with respect to their relative capabilities for retaining radionuclides released from the MHTGR via depressurization accidents.

The DOE approach is based on the assumption of a source term, or radionuclide release to the reactor cavity, consistent with that in the Preliminary Safety Information Document (PSID, DOE-HTGR-86-024). DOE showed that significant reductions could be made to the already very low estimated releases from the reactor building and doses at the exclusion area boundary (EAB) by progressive enhancements of the confinement capabilities (with corresponding increases in costs). The alternatives ranged in four successive steps from the reference design (vented, no filtering, 100%/day leak rate) to a near-conventional sealed containment building (CB) with an unvented, high pressure, 1-5%/day leak rate design. This latter design uses a water-cooled RCCS and having an incremental capital cost of ~$90M over the reference design. The major differences in the alternatives are summarized in Table 1-1.

In Alternative 2, filters are added to the vent path (reactor building roof structure). In Alternative 3, a steel liner is added to give a moderate leakage of 5%/day. Alternative 4 includes two options, both unvented and capable of withstanding a moderate pressure, making use of common expansion volumes connected to the CB areas of each module via check valves. One variation (a) uses a large expansion volume to mitigate depressurization accidents, where the maximum CB design pressure is 10 psig. The second variation (b) uses a smaller expansion volume, with a maximum CB design pressure of 25 psig. In the latter case, a water-cooled RCCS panel is substituted for the standard natural-convection air-cooled panel in the reactor cavity. The CB design pressure for Alternative 5 is 80 psig.

The NRC's initial response to HTGR-88311 (letter from B. M. Morris to S. Rosen of DOE, May 9, 1990) requested a reevaluation based on the assumption of larger prompt and delayed source terms. While the DOE containment study provided a reasonable set of CB design alternatives for accommodating progressively larger assumed source terms, it did not consider or estimate the larger releases. Hence, the reevaluation of the containment question should include a clarification of how the source term is calculated, and further provide for considerations of progressively larger, less likely, source terms.

The crux of the matter is fuel reliability, along with the development of an adequate fuel performance model. NRC contends, and we concur, that the PSID fuel performance goals have not been substantiated by the testing program to date, and that there is not yet adequate assurance that the proposed QA/QC program for fuel production would guarantee that sub-specification or "weak" fuel would not be produced and go undetected. The following discussion is intended to contribute to a justification for the position that larger source terms, or potential releases to the reactor cavity and building environs, should be assumed at this time.

Table 1.1 Summary of DOE Containment Study Design Alternatives

Alternative : leakrate	Schematic (HTGR-88311)	RCCS	Incremental Capital Cost	Site Boundary Doses (REM) Whole Body/Thyroid: Maximum	Site Boundary Doses (REM) Whole Body/Thyroid: DBA
1. Reference Design Vented, Unfiltered:100%/day	Fig. 4.3.1-1	air	0*	0.3/3.0	0.02/0.2
2. Vented, Filtered:100%/day	Fig. 4.4.1-1	air	$5 M	0.2/2.0	0.01/0.1
3. Vented, Filtered:5%/day	Fig. 4.5.1-1	air	$15 M	0.2/0.7	0.01/0.02
4. Unvented, Moderate Pressure, Expansion Volumes:5%/day	a. Fig. 4.6.1-1	air	$39 M	0.02/1.0	0.0006/0.008
	b. Fig. 4.6.1-2	water	$33 M	0.02/0.5	0.0006/0.008
5. Unvented, High Pressure: 1–5%/day	Fig. 4.7.1-1	water	$90 M	0.02/0.5	0.0006/0.008

*Total capital cost for Reference Design Reactor Building (Nth of a kind) = $55.4 M.

Fuel Reliability Fuel "reliability" may be defined as behavior in quantitative accord with model predictions in both normal service and under accident conditions. (See "Evaluation of MHTGR Fuel Reliability" by R. P. Wichner and W. P. Barthold, NUREG/CR-5810 - ORNL/TM-12014, July, 1992 for a more detailed discussion.) When applied to accident consequence estimates, the current fuel behavioral models indicate that reactor safety goals are met without the sealed containment. Extensive in-pile fuel tests, as well as both U.S. and foreign reactor operation experience using fuel similar to the current selection, were used to support this assessment. Preliminary analyses of some recent in-pile test data, however, have indicated that the current fuel performance models may not be valid (i.e., they may be significantly non-conservative).

Several considerations encourage a conservative or cautious approach on the part of the NRC. It is noted that the current fuel design is relatively new, and while enhanced future performance is promised, no extensive testing record on the current fuel design now exists. The existing fuel behavior models are based on older fuel designs, principally the high quality fuel produced in Germany for the AVR and THTR HTGRs. In addition, a prototype fabrication facility for the selected fuel design does not currently exist. Bench scale facilities were used to fabricate the fuel for most of the limited testing of the current fuel conducted thus far.

There are four essential elements of "fuel reliability:"

(a) proper fuel design to satisfy performance specifications;

(b) fabrication and process control techniques which accurately produce the fuel within design specifications;

(c) well constructed QA and QC procedures; and

(d) behavior of the product in service that is in accord with expectations. Hence, an evaluation of fuel reliability must cover all these areas.

The "Weak Fuel" Concept As an expression of caution, the NRC adopted a concept termed "weak fuel" to explore the effect of errors regarding the degree of fuel integrity. The "weak fuel" concept is a penalty placed on consequence estimates in which a poorer fuel response is assumed vs. what is predicted by the behavioral models. In the scenario of particular interest, it is assumed that a bad or "weak fuel" batch exhibits no detectable normal-operating-condition abnormalities; however, during a design basis accident (DBA), many more than the expected number of failures occur. The excessive failures could be due to the standard fuel particles' having an unexpected susceptibility to high (but less than 1600° C) temperatures for dry DBAs, or to hydrolytic attack (and high temperatures) for wet DBAs. The "weak fuel" concept is currently being applied, on an interim basis, to MHTGR concepts which do not employ a sealed containment vessel.

Fuel Fabrication Lab-scale MHTGR-quality coated fuel particle fabrication has been demonstrated overseas. The German company HOBEG has fabricated MHTGR-quality fuel particles that were irradiated, followed by heatup tests. This fuel performed as expected. However, at this time, (1) no U.S.-made MHTGR-quality UCO fuel has been fabricated in more than capsule test quantities and (2) this fuel has not been irradiated under prototypical conditions. The performance of U.S.

MHTGR fuel is inferred from earlier U.S.-made UC_2 fuel and German-made, mostly UO_2, fuel.

The fabrication process for coated particle fuel is highly complex. Temperature, pressure, gas composition and flow rates, coating rates and raw product compositions have an impact on the fuel product attributes and need to be tightly controlled in the fabrication process. The process parameters affect the geometry of the fuel particle, the density of its components, and attributes like microporosity, isotropy/anisotropy and SiC phase composition.

The coated particle is a complex fission product containment system, consisting of multiple interacting coating layers whose properties change under irradiation. How simultaneous changes that stay within the specifications (e.g., coating thickness) affect fuel performance under normal operating and accident conditions is so far only inferred from models developed for different fuel types operating under different conditions. There is no universal fuel model, but models developed so far are design-specific, and how the fuel designs perform depend also on specifications, fabrication, quality assurance and control.

The fuel compact fabrication process also introduces the possibility for coating failures. While the protective pyrocarbon (Ppyc) layers are intended to provide protection in the compact fabrication process against excessive loads that could crack coating layers, the relationship between coating failure and the need for a Ppyc layer has not been clearly established.

<u>QA/QC Methods</u> While many quality assurance and quality control (QA/QC) methods have been developed and used in the past, there are questions about the reliability of the quality control of the burn-leach process, which is used to determine the integrity of the very important SiC coating layer.

Most of the currently available quality control techniques, including burn-leach, are destructive methods. Because the initial fuel defect fraction is to be in the 10^{-5} range, large samples sizes in the 10^5 range need to be analyzed for proof of the low defect fraction. In this regard it is very important to emphasize that even a perfect QA/QC program can only ensure as-manufactured fuel quality, but it cannot ensure that the fuel is also reliable.

While different sample sizes have been proposed for the sampling of different attributes, no relationship between sample size, attribute and impact on performance has been documented, i.e., the current QA/QC plan does not explicitly prioritize quality control activities. Such a prioritization needs to be based on the relative (and quantifiable) importance of particular measurements with respect to achievement of the fuel design requirements.

At this time it is also not known how uncertainties in quality control techniques have to be accounted for in quality assurance. Furthermore, it is necessary for fuel QA/QC to distinguish between an evolving fabrication technology and a mature fabrication technology. For a first of a kind reactor, different constraints have to be applied than for the n-th reactor.

The proposed QC methods test the measurable properties of the fuel, such as thickness and uniformity of the coatings. However, none tests the functional requirements for the fuel particles and compacts, i.e., the overall radionuclide retention capability of the complex multi-layer particle design and fuel compact.

Conditions for Reducing the "Weak Fuel" Burden Because of these risks and uncertainties, the imposition of a weak particle penalty appears to be justified, based the current state of the art in fuel and fission product technology. As the technology base is broadened with the understanding of fuel performance and fission product transport, the weak fuel penalty could be lessened.

The following conditions are recommended to reduce and possibly eliminate the "weak fuel" penalty for MHTGRs with no sealed containment:

1. the existence of a mature MHTGR fuel fabrication industry with an established record for producing fuel which performs to expectations in reactor service;

2. a good comprehension of fuel failure mechanisms to provide an unambiguous interpretation of capsule and in-reactor test data;

3. a good understanding of fission product transport; and

4. a successful testing/demonstration program of sufficient scope on the selected fuel design, produced using prototypical methods.

2. Consideration of Alternative Containment Design Features

Three alternative containment building (CB) design features are proposed for consideration:

1) "Stronger" air-cooled RCCS in lieu of water-cooled RCCS

For alternatives in which the DOE design has called for a water-cooled RCCS, we recommend considering use of a "stronger" air-cooled RCCS. It appears that the criterion used by DOE to switch to water is the maximum expected cavity pressure (10 psig limit on air RCCS design, vs 25 psig limit on the water RCCS design). In parametric studies using the ORNL MORECA code (Ref. S. J. Ball, "MORECA: A Computer Code for Simulating MHTGR Core Heatup Accidents," NUREG/CR-5712, ORNL/TM-11823, October 1991), the use of thicker panels caused no significant degradation of the predicted heat removal capabilities of the RCCS original design. Due to the better reliability, simplicity, and maintainability of the all-air-cooled system, we believe it to be a better choice even for the moderate pressure design CBs. It should be noted that the recently-released Amendment 13 to the PSID has a description of a new air-cooled RCCS reference design. While a reevaluation of this design is not within the scope of the present review, it would appear that the same conclusion would apply.

2) Use of the main reactor building as a holdup volume for releases

An economically-attractive alternative to the use of an expansion volume (Alternatives 4a and 4b) for holdup of radioactive gas discharged via a primary system depressurization might be the use of the main part of the reactor building. In this design, however, the building would be used only as an atmospheric-pressure holdup vessel. Per a rough scaling of the building drawings, hydraulically "attaching" the main building to the current holdup/discharge volume of the reference design would increase the effective holdup volume by about a factor of three. Even with no special measures to reduce the leakage rate (from the current <100%/day), the extra holdup time would provide significant extra decay time for the noble gasses and short-lived iodines, as well as extra surface area for fission product plateout and deposition.

To protect operating personnel that may be in the reactor bay area at the time of an accident, it may be prudent to install normally-closed adjustable louvers between the current discharge space and the bay area. If the discharge is due to a steam line break (with no radioactive release), the louvers would remain shut, with the discharge to the outside exiting via the dampers (as in the Reference Design, Alternative 1). Following a primary system depressurization, however, should higher than expected releases to the atmosphere be encountered, the louvers could be opened (manually), providing the subsequent discharges with more holdup time in the already-evacuated bay area. The timing for this operator action would not be critical (i.e., within hours of the accident).

3) Primary relief valve filter train:

In Section III.8 there is a discussion of an alternative CB design which incorporates a HEPA and charcoal bed filter downstream of the primary system pressure relief valve(s). This additional feature is recommended because the probability of primary system leaks due to vessel failure is likely to be several orders of magnitude smaller than accidents involving depressurizations which discharge through a stuck-open relief valve. Small leaks in the vessel are also not likely to be associated with other concurrent failures, and hence the discharge would probably not contain radionuclides other than those normally circulating in (and lifted off from) the primary system. In those cases, the normal heat removal systems would be expected to function to maintain fuel temperatures at or below normal temperatures. On the other hand, relief valve depressurizations are most likely to be caused by a sizeable steam generator tube rupture accident, which could compound the problem by introducing:

a) a power/temperature surge due to positive reactivity insertion;

b) disruption of the normal heat removal capabilities; and

c) an increase in the chances of fuel failure, especially of weak fuel, due to elevated fuel temperatures and hydrolytic attack.

Evaluations of a relief valve vent train filter option should account for:

a) the potential advantages and disadvantages of filtering only the relief valve(s) discharge, noting that in a moderate (100%/day) leakage building with the reference design filters, a sizeable fraction of the discharge may bypass those filters;

b) the possibility that a single (reconfigurable) filter train may be sufficient for a multi-module plant;

c) the fission product chemistry and plateout phenomena that would affect site boundary doses (for both wet and dry depressurizations); and

d) the need to account for temperature and humidity (for wet depressurizations) and pressure drop limitations of the filter train. Charcoal filters have an ignition temperature of $^\sim 330^\circ C$, for example, so the discharge path for the primary system helium may have to incorporate some high heat capacity material to prevent filter damage. Also, expansion volume(s) would be necessary to limit the pressure drops across the filters to account for filter design limitations.

3. Evaluation of Dose Reduction and Cost Estimates

A. Dose Reduction Estimates

1) Effect of "Scaling" Risks

The basic theme of the DOE Containment Study (DOE-HTGR-88311) with respect to dose reduction is presented in its Table 2-1. This asserts that the safety factor for latent cancer risk relative to the safety goal is 10^5 for the base case reactor building. Therefore, according to this view, adding costs to further augment this already high safety factor would be unwarranted.

The risk estimates for all reactor building alternatives (2 through 5B) are developed from the base case (Alternative 1) by use of "scaling factors", as outlined beginning on p. 4.2.1-3 of the Study. The procedure uses the base case results as a starting point. Accident initiators are the same for all cases and event sequence probabilities are obtained by comparison with the base case.

The precise procedure for "scaling" the consequence estimates is not clearly given. However, again the base case appears to be used as a starting point, with release reductions for Alternatives 2 through 5B allocated by comparison using the particular features of each case.

In this situation, the key point is the validity of the base case risk estimates. All else depends on the base case results. All building alternatives beyond the base case tend toward higher levels of containment. Therefore, generally we may expect progressively lower risks for Alternatives 2 through 5B relative to the base case.

A more pertinent study would be directed at proving the validity of the base case risk profile. Such a review would require a document which covers the assumptions and transport models used to determine reactor building retention and methods for estimating associated probabilities. The report should include sufficient detail on assumptions and models to allow a technical review of the results. The current report is too general and vague on these points to permit such a technical evaluation. (Note, even the PRA [PRA, 1987] may not have provided sufficient detail for technical assessment.)

2) DOE Responses to NRC Containment Criteria

Containment Study pages 3.1-9 and following present DOE's comparison of NRC Containment Criteria with the base case MHTGR situation. The DOE response to NRC Criterion 1 asserts that the MHTGR provides multiple barriers against radiation release. The fuel kernel, moderator graphite, and reactor building are included as barriers. Normally porous materials and a fairly open building are not considered to be true barriers. These are attenuators to varying degrees, but certainly not barriers equivalent to fuel pin cladding, the pressure vessel boundary, or a sealed containment vessel. The true barriers for the MHTGR base case are the SiC coating layer and the pressure vessel boundary. If one considers only noble gases and iodine, then the OPyC layer may be considered as an additional barrier. (However, fuel behavior models predict about 3% failures of the OPyC during normal operation.)

Criterion 3 asks for the modified QA, surveillance, and ISI needs of the MHTGR. The response is extremely brief to this very difficult inquiry. Wichner and Barthold (1992) present a discussion of the QC procedures available for MHTGR fuel. There is a major difficulty in this area, namely, the lack of a suitable test of the fuel particles in the compact. This would be a QC procedure strategically equivalent to the helium leak test of the LWR fuel pin.

3) Comments on the Assumed Source of Radioactivity in the Reactor Building

Section 4.1.1 in the Containment Study discusses the potential source of radioactivity in the reactor building. The last sentence of paragraph 2 on p. 4.1-2 may lead to a misunderstanding which could in turn lead to an underestimate of the fission product release from fuel. This states "....the only significant radionuclide source available for release is that which is outside of standard particles." According to the definition in section 6.2.2 of Reg. Plan (1987), a "standard particle," while assumed to be free of defects, can nevertheless fail in service by means of all the known failure mechanisms. As noted by Wichner and Barthold (1992), this definition is somewhat ambiguous because defect-free particles are not likely to fail in normal service. They suggest an alternate set of fuel particle definitions, using the term "ideal particle" to signify what appears to be intended here by the term "standard" particle.

The point here is that the radioactive source from fuel described by the Containment Alternatives Study appears to exclude fuel material exposed by expected in-service failures. This may be the largest radioactive source from the fuel. Hence, a misunderstanding regarding the definition and behavior of "standard particles" could have lead to an underestimation of the source term.

The text following on the same page also reveals the same possible misunderstanding of the definition of "standard particles." As used, the meaning is more in line with the term "ideal particle," as defined in Wichner and Barthold (1992). An "ideal particle" also contains no defects, and will not fail in normal service.

Table 4.1-1 in the Containment Study lists the potential I-131 source into the building separated into four inventory categories. The circulating and plateout levels match closely the estimates of Wichner (1991, Table 5.4-1) for I-131. However, according to Wichner, plateout of other iodine isotopes brings the total up to about 45 Ci, of which about half is I-131.

Table 4.1-1 contains the same possible misunderstanding of the definition of "standard" particles as noted above. This may be corrected by use of the term "ideal" particle instead. However, in that case, the radioactive source from particles broken in normal service needs to be added to the list. This may be the largest potential source of radioactivity release from the core.

In addition to fuel exposed, as expected, by normal service conditions, an additional degree of failure may be incurred under harsher accident conditions. Although current models predict negligible additional failures, a containment study perhaps should prudently investigate the effect of possible enhanced failures under accident conditions.

Footnote 1 of Table 4.1-1 of the Study and the related text state that "approximately 7% (the UC_2 fraction) of the inventory in non-intact particles is subject to release in a time frame of minutes under hydrolyzing conditions encountered in rare MHTGR accidents." This may need correction on two counts. First, the 7% value appears to be the mid-duration UC_2 fraction in the kernel, initial fraction being 15% and the final about 0%. The UC_2 fraction in a two-zone core could therefore be as high as 11%, which would occur following a reload.

More importantly, hydrolysis test data have been reported (Myers, 1991) which present a different progression of the hydrolysis event than assumed by the Containment Study. According to Myers, all the exposed fuel "hydrolyzes", not just the UC_2 fraction, although the chemical mechanism for UO_2 "hydrolysis" is not clear. Briefly, the results show that all the stored gases (including iodine) in exposed fuel would be released in approximately 1-hour following steam addition, extrapolating steam pressures to levels expected in a steam generator tube rupture event.

The above discussion seems to indicate that major adjustments may need to be made in the radioactive source to the building assumed in this containment study. At the very least, the assumed source to containment needs to be clarified. The description given in section 4.1.1 is not sufficiently explicit. The listing of the potential radioactive source should include all important nuclides and the assumed chemical forms of each element. The latter impacts the attenuation mechanisms existing in the building.

4) Dose Reduction Mechanisms for the Alternative Building Designs

The DOE containment study offers no discussion of dose reduction mechanisms for the base case. In addition, only a brief outline is provided on the "scaling" rules used to determine dose reductions for alternative building concepts from the base case results.

a. Alternative 1 (Base Case)

Maneke (1988) provides a fairly good description of the base case reactor building, including description of the internal rooms and flow paths for releases from the primary containment to the atmosphere. Figures 3.1 and 3.2, from Maneke, show an isometric of the building and the leakage paths through the building, respectively. Note that a leak from the reactor vessel flows downward, past the cool RCCS surfaces and into a series of chambers underneath the vessel. A blowout panel allows flow to proceed upward into the steam generator cavity, and from there through baffles into a circular plenum, the ceiling of which is at ground level. The path in the plenum leads first through hinged louvers, then upward through fixed louvers, from which the flow is discharge to the atmosphere at elevation +30 feet.

Discharge to the atmosphere can be separated into two time periods. Initially, the discharge is impelled by helium pressure from a failure in the pressure vessel boundary or by steam pressure from a failure in the steam generator pressure boundary. When pressures equilibrate, a leakage rate from the building of 100%/day is assumed. Though not stated, presumably this means that the volumes involved in the leakage pathway described above are combined, and one complete exchange occurs with outside air per day. However, the precise meaning of the assumed 100%/day leakage rate as well as its basis need to be stated. The principal mechanism affecting this exchange is probably natural convection.

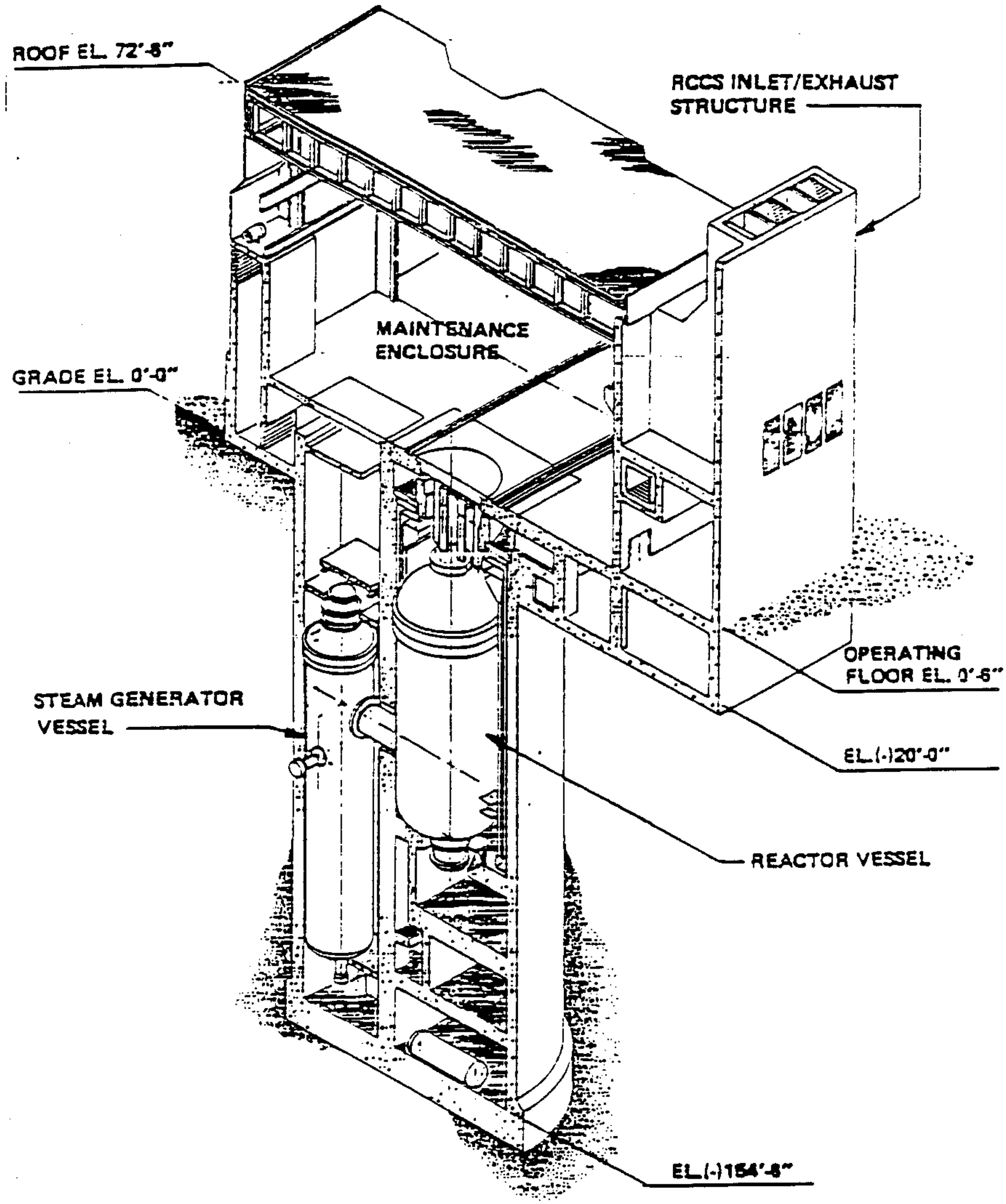

Fig. 3.1. Isometric view through reactor building (Maneke, 1988)

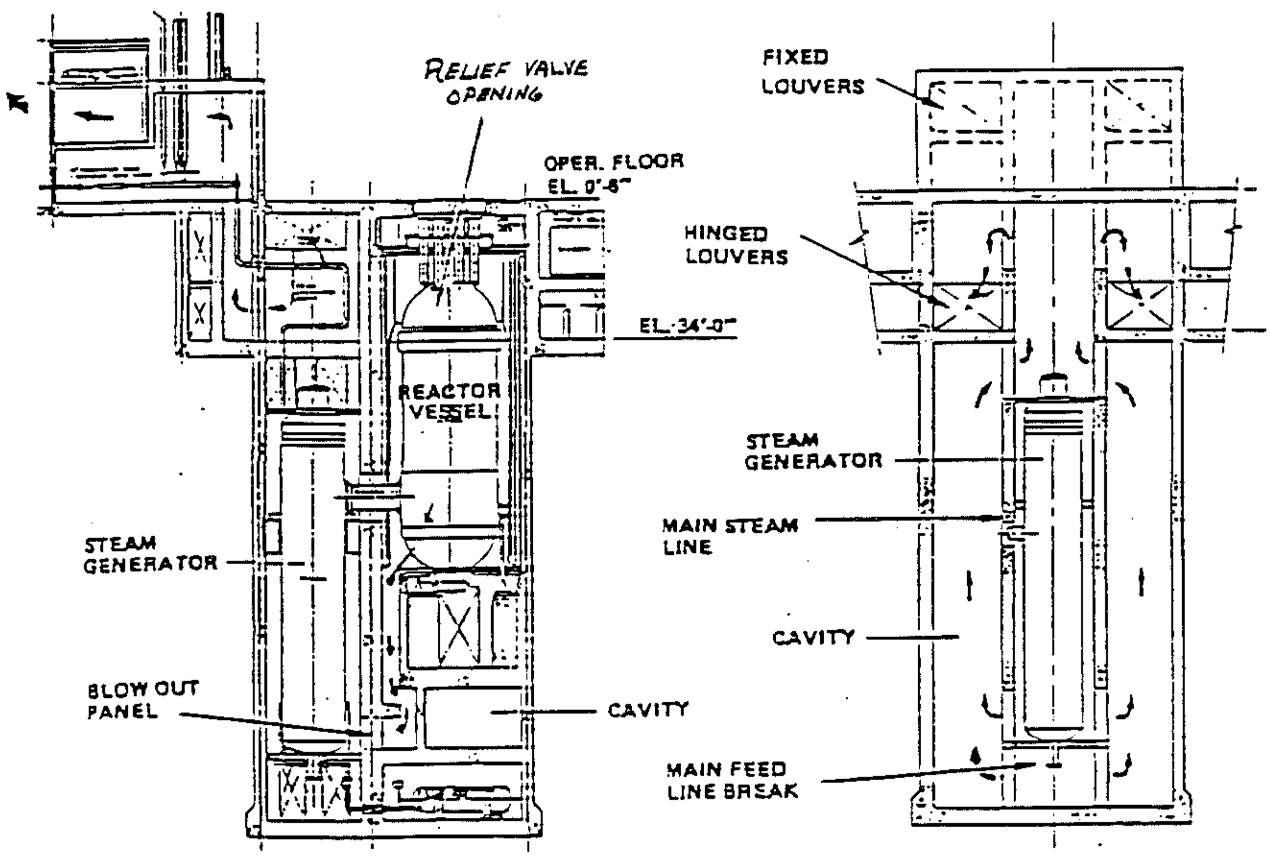

Fig. 3.2. Vent pathway schematic (Maneke, 1988)

The radioactivity attenuation mechanisms that are effective in this flow path depend on the chemical form of the fission products entering the building:

(1) Noble gases will be attenuated only as a result of radioactive decay during holdup;
(2) Non-noble gases, iodine being the principal one, may chemisorb on all available surfaces. The cool, metallic RCCS surface should be an effective sorption surface for iodine. However, iodine sorption is expected to occur on all other surfaces, including bare concrete and painted surfaces;
(3) Radioactivity associated with particulates will deposit to some extent on all surfaces. The principal transport mechanisms depend mainly on the particle size, but also on wall temperatures and local convective flowrates; and
(4) Additional attenuation mechanisms come into play for accidents involving steam discharge.
    - 4a. There will be an enhanced transport of gases and particulates to surfaces on which steam condenses (diffusiophoresis).
    - 4b. The presence of condensed water allows retention of many fission product compounds due to dissolution.
    - 4c. Wet walls enhance retention of particulates.

It should also be noted that conditions may exist in some accidents which enhance release of deposited material. Elevated temperatures or the presence of steam will cause some degree of iodine desorption.

Evaluation of the degree of attenuation along the base case leakage pathway (and the pathways for all other options) requires knowledge of the chemical form of gaseous species and particle characteristics of radioactivity associated with aerosols. The major uncertainty lies with the uncertain degree of association of iodine with particulates. An estimate by Wichner (1991, Table 5.3-2) based on dust samples from the Peach Bottom HTGR indicates that from 64 to 97% of the iodine circulation in the primary system exists as gaseous iodine, with the balance as adsorbed iodine on particles. This would be the form of iodine entering the reactor building in the near term as a result of depressurization. Near term iodine releases from the "plated" iodine on primary system surfaces due to "blowdown" are likely to be quite small (Wichner, 1991).

Subsequent releases of iodine and other fission products from the fuel occur for accident sequences which involve core heatup. A portion of this subsequent release escapes from the pressure vessel boundary and enters the reactor building space. Retention factors for the building depend on the physical/chemical form of the radionuclides. However, except for iodine and the noble gases, the predominant form is most likely to be particulates, principally oxides, due to low building temperatures and the predominantly air atmosphere.

Nonetheless, the magnitude and physical form of iodine entering the building needs to be verified by an appropriate test series.

An important feature of iodine speciation in the reactor building that is not present within the normal primary system is the degree of organic iodide formation. Sorption of organic iodine on surfaces is significantly less than the inorganic form. Therefore, the retention factor of organic iodine is significantly lower. The degree of organic iodide formation becomes especially important for building Alternatives 2 and 3 which include charcoal filtration. Sorption of organic iodide is significantly lower than for I_2 on charcoal, even when specific additives are used to enhance sorption.

b. Alternative 2 (Vented Filtered, Moderate Leakage)

Alternative 2 is identical to the base case, with the exception that the exhaust from the exit plenum is routed through filter assemblies which include HEPA filters and charcoal absorbers. The type of assembly used may be similar to the Standby Gas Treatment System assembly used in BWRs. An example of such a filter assembly is shown in Fig. 3.3 (from Wichner, et. al., 1983). As shown for this BWR application, fans are used to drive flow through the filter assembly. A new assembly requires about 13 inches of water pressure difference to drive about 8500 SCFM through the filter, for the particular case shown.

Each filter assembly for the BWR case consists of the following components in series: a moisture separator, a humidity control section, a roughing filter, an upstream HEPA filter, a charcoal absorber section, and a downstream HEPA filter. Some further discussion of this system as used in a BWR is given in the cited reference.

The moisture separator and humidity control sections are needed because the sorptive capacity of the charcoal bed for iodine is diminished by excessive humidity.

According to the Containment Study, leaks of less than 1 in^2 size in the primary system, which raise building pressures less than 2 psig, are routed through the filtration system. Larger leaks bypass the filters to avoid damage due to excessive pressure. This is accomplished by designing the building pressure relief dampers to open at $\sim$1 psig. Thus the initial discharge from a large leak will still be directly to the atmosphere. The dampers will close when building pressure falls to <1 psig.

The Containment Study assumes than the HEPA filters in the filter assembly retain 90% of the nuclides associated with particulates, and the charcoal bed retains 95% of the halogens. These are low estimates compared with the intrinsic capabilities of HEPA filters and charcoal beds. For example, see Tables 2.3 and 2.4 of Wichner, et.al. (1983). Two HEPA filters in series are expected to retain 99.99% of particles. A new charcoal bed may retain 99.99% of iodine gas as I_2, and properly impregnated, 99% of the organic iodide.

The principle uncertainty regarding realistic fission product removal fractions for the MHTGR reactor building relates to determination of the true leakage flow path. At low building pressures, the fraction of leakage flow that will actually go through the filters is not clear. The cited assumed values of 90% and 95% removal for particulates and gaseous iodine presume that about this fraction of the leakage flow actually goes through the filter assembly. This is not clearly the case, and may not be consistent with the assumed open building structure of 100% per day leakage rate.

An additional question relates to filter assembly behavior for steam line break and steam generator tube rupture accidents. HEPA filters can weaken on contact with high moisture levels. In addition, iodine adsorption on charcoal is reduced by high humidity levels. Therefore, the humidity control potion of the filter assembly may need to be selected for the case of a steam line break accident.

In summary, all retention mechanisms cited for the base case in section 2.1 apply to Alternative 2. In addition, discharge flows that pass through the filter assembly will experience radioactive retention, subject to the above discussion.

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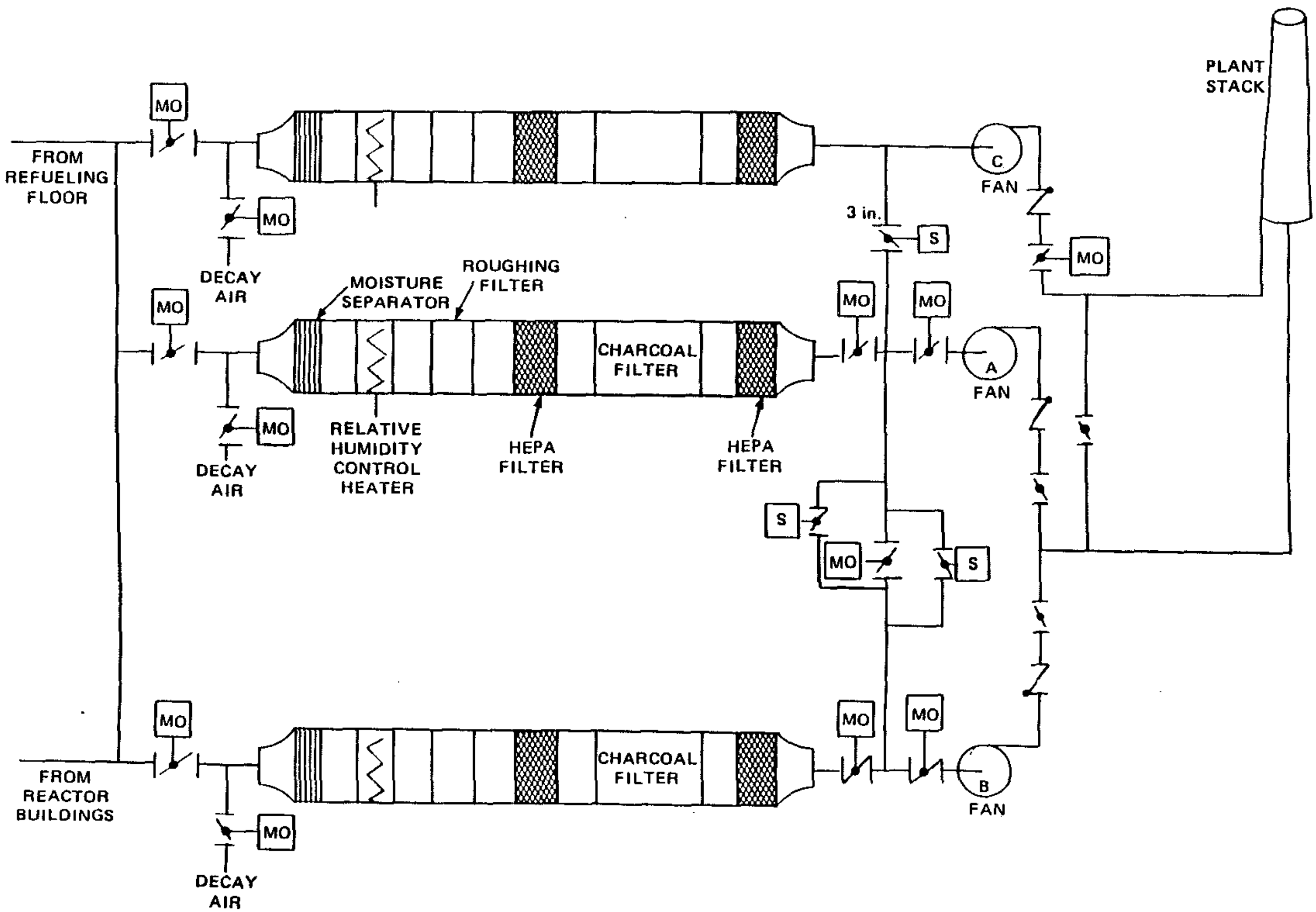

Fig. 3.3. Browns Ferry standby gas treatment system (Wichner et al, 1983)

c. Alternative 3 (Vented Filtered, Low Leakage)

Alternative 3 is similar to Alternative 2 in that pressurized discharges above 1 psig pass directly to the atmosphere. However, the building in this case is provided with a steel liner which limits leakage to 5% per day instead of 100% as in Alternative 2. Thus once the HVACs dampers close at <1 psig, a significantly higher portion of the subsequent leakage to the atmosphere may be expected to pass through the filter assemblies instead of bypassing the filters via building leakage.

Expect for this one difference, identical radioactivity retention mechanisms are in effect for Alternative 3 and Alternative 2.

d. Alternative 4 (Unvented, Moderate Pressure, Low Leakage)

Alternative steel-lined expansion volumes are provided in building Alternative 4, connected through a check valve to the reactor cavity volume. Alternative 4A provides for a total expansion volume of 2 million ft^3 and a maximum pressure of 10 psig. Alternative 4B has a total expansion volume of 600,000 ft^3 and would result in a maximum pressure of 25 psig. In both cases a maximum gas leakage rate from the expansion volume of 5%/day, presumably under full pressure conditions, is assumed. According to the study, a water-cooled RCCS would be required for Alternative 4B due to the need to withstand the higher pressure. (See discussion in Section III.2 of this report.)

The radioactivity removal mechanisms cited for Alternative 1 are all effective for Alternative 4, with appropriate modifications of rates to account for the different physical circumstances.

Lower noble gas leakage relative to Alternative 1 would occur due to (1) containment of the initial depressurization flow, (2) dilution of the subsequent leakage from the core into a larger volume, and (3) provision for a lower leakage rate of 5%/day, vs 100%/day in Alternative 1.

Iodine release to the atmosphere would be reduced relative to Alternative 1 for the same for the same reasons as cited above. Additionally, chemically reactive iodine would be removed from the gas space, and hence from the inventory available for leakage, to a greater degree than for Alternative 1 by the natural mechanism of chemisorption on the walls. The reason for the added degree of removal by chemisorption is simply more time and more surface area. Iodine associated with aerosols would settle within the expansion volume to a greater degree due to the longer residence time.

All other radioactivity in the expansion volume is expected to be associated with particulates due to low the temperature and the air atmosphere, which tends to form oxides of the active metal fission products. Therefore, an enhanced degree of removal is expected for all non-gaseous fission products relative to Alternative 1 due to the longer residence time.

In addition to enhanced natural removal mechanisms, Alternative 4 provides an opportunity for increasing removal rates in the expansion volume by installation of conventional containment air cleaning systems such as used for LWRs. However, the Alternative 5 does not currently include such a provision.

e. Alternative 5 (Unvented, High Pressure, Low Leakage)

Alternative 5 essentially provides a conventional containment vessel for each MHTGR module. A leakage rate of 5%/day is assumed for Alternative 5A. Alternative 5B is identical to 5A; however a lower leakage rate of 1%/day is assumed. It is estimated that peak pressures of about 80 psig would occur in the 249,000 ft^3 vessel as a result of a major helium and steam release event.

The above discussion of radioactivity removal mechanisms for Alternative 4 applies also to Alternative 5.

5) Characteristics of Containment Filtration Systems

Characteristics of a typical BWR filtration assembly are briefly noted in section 4) b., above. It is likely that an MHTGR filtration assembly would contain similar components and possess similar flow and removal behavior.

The important points to note about this behavior with respect to MHTGR application are the following:

a. A significant pressure drop is required to drive a large gas flow through the assembly. This is provided by a fan for the BWR application. Figure 3.4 (from Wichner, et.al., 1983) shows a typical fan head curve and assembly pressure drop. It is not clear what the effective flow through the filter would be for a pressure-equilibrated case with essentially no pressure driving force.

b. Filtration performance drops with the presence of large amounts of moisture. The sorbancy of charcoal for iodine is reduced by moisture. Also, the strength of the HEPA filter is adversely affected. Thus, the filter assembly would have to be designed to accommodate a selected moisture ingress event.

c. Would the filtration system be classified as an "engineered safety feature?" This may be a debatable point, but it seems that if its function is to reduce radioactive leakage to the atmosphere, it may be so classified. As such, it would be subject to all the design, testing, and maintenance criteria required of Class 1E equipment. Application of these criteria would be expected to significantly increase its cost.

d. The filter assemblies require scheduled maintenance and testing. If the assemblies are classed as "engineered safety features", the maintenance, design and testing criteria described in NRC Reg. Guide 1.52 (Reg. Guide, 1978), must be observed.

e. The sorbency of the charcoal bed for iodine diminishes with temperature. Iodine sorbed at low temperature during the initial stage of a core heatup sequence may desorb at later stages if building temperatures rise. This behavior is illustrated in Fig. 3.5. Note that exposure to 300°C air at a superficial velocity of 2 cm/s can completely desorb the charcoal bed in about 200 minutes.

6) Summary of Dose Reductions

The whole body and thyroid dose estimates for reactor building Alternatives 1 through 5B are summarized in Table 3.1. Values listed under columns labeled "DBA" and "MAXIMUM" were taken from tables in the DOE Study presented at the end of sections 4.3 through 4.8. Dose values listed in the last column were taken from the risk profile curves for each case for the annual accident probability of 10^{-8}.

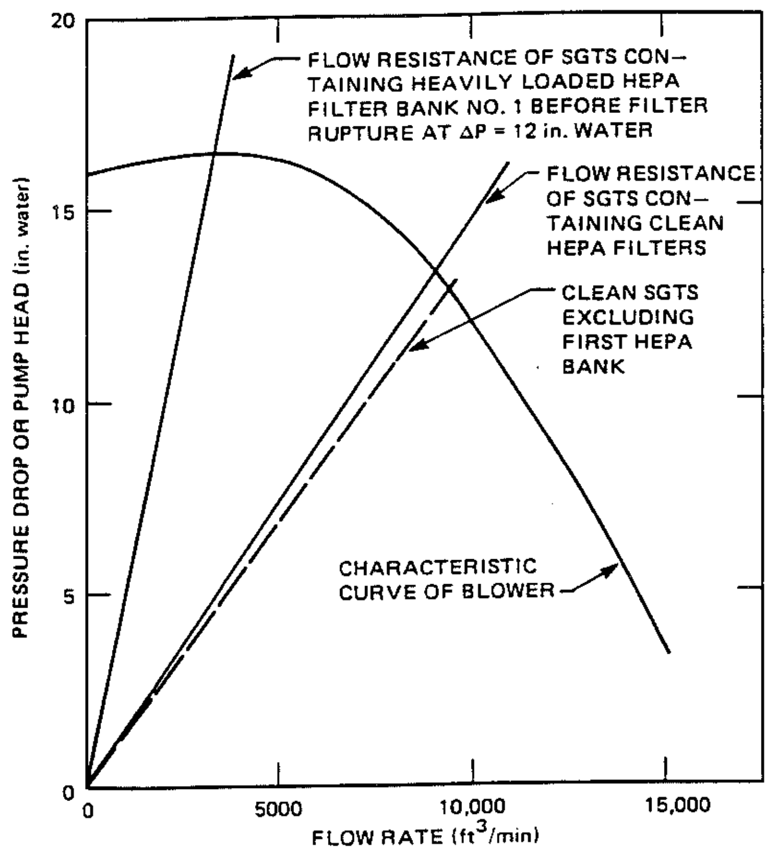

Fig. 3.4. Flow characteristics of a single SGTS train

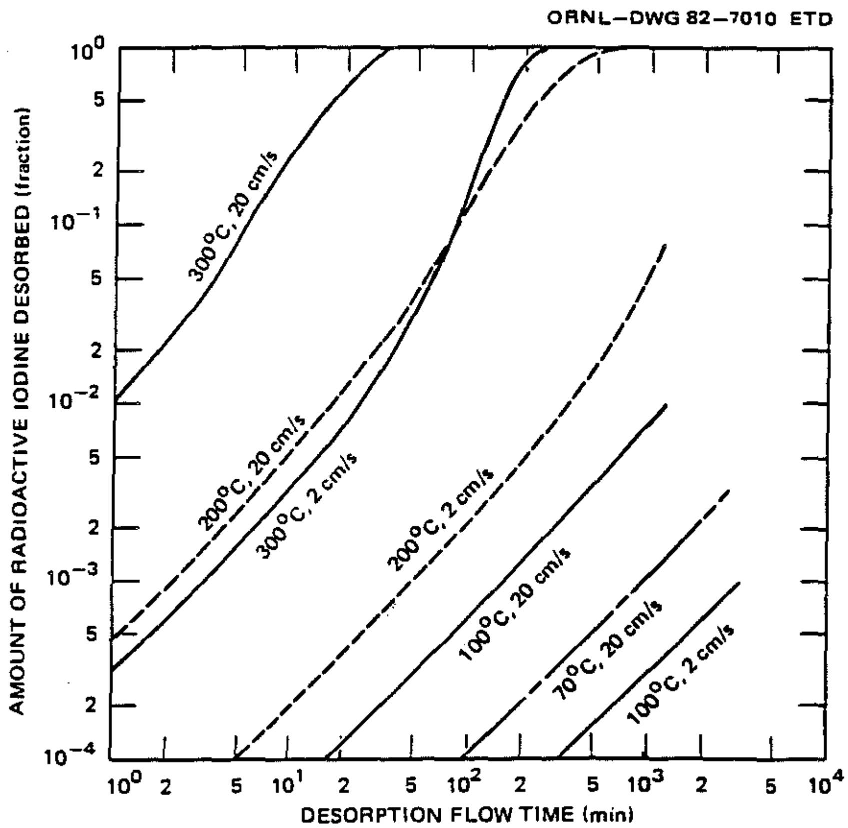

Fig. 3.5 Desorption from a 2-in. deep charcoal bed

	DBA		"MAXIMUM"		$P = 1x10^{-8}/y$
Building Alternative	Whole Body	Thyroid	Whole Body	Thyroid	Thyroid
1	0.02	0.2	0.3	3.0	45
2	0.01	0.1	0.2	2.0	40
3	0.01	0.02	0.2	0.7	10
4A	6 E-4	8 E-3	0.02	1.0	80
4B	6 E-4	8 E-3	0.02	0.5	10
5A	6 E-4	8 E-3	0.02	0.5	75
5B	5 E-4	2 E-3	0.02	0.5	75

Table 3.1. Predicted doses at Exclusion Area Boundary (REM)

The limits in 10 CFR 100 apply to site boundary doses incurred as a result of DBAs, and hence should be compared to values in the DBA column. Note that the predicted whole body dose for Alternative 1 of 0.02 rem is a factor of 1250 less than the 10 CFR 100 guidance of 25 rem. The predicted thyroid dose for Alternative 1 of 0.2 rem is a factor of 1500 less than the 10 CFR 100 guidance.

As expected, predicted whole body and thyroid doses steadily from Alternatives 1 through 5B, which represent progressively increasing degrees of containment.

Doses listed under the columns labeled "maximum", pertain to estimates corresponding to the $5x10^{-7}$ accidents per year probability. According to the DOE, MHTGR accidents of lower probability have no realistic basis for occurring, and hence are hypothetical. Note from the table that doses for Alternative 1 for the $5x10^{-7}$/yr probability accident are predicted to be about 80 and 100 times less than 10 CFR 100 guidance for whole body and thyroid, respectively.

The risk profiles presented for each building case show dose consequences down to 10^{-8}/yr probability. Dose values for this low probability case range from 10 rem to the thyroid for Alternative 4B up to 75 rem for Alternative 5, but show no consistent trend with building alternative. The point that the DOE study is making here is that the predicted 45 rem thyroid dose for the base case is still a factor of 6.7 below 10 CFR 100 guidance, even for a non-realistic accident of 10-8 probability.

7) Conclusions Regarding Estimates of the Dose at the Site Boundary

(1) Site boundary dose estimates for each building alternative depend directly on the magnitude and timing of the assumed leakage from the core. This source is only briefly described in the six paragraphs of section 4.1.1 and accompanying Table 4.1-1 in the Containment Study. This brief description indicates that the source from the core may have been underestimated. There is no indication that particles expected to fail under normal service were included in the category of exposed fuel. In general, the source from the core needs to be more clearly and completely defined, even if only by reference to some more comprehensive study.

(2) The assumed radioactive source from the core appears to have also excluded the possibility of fuel additional failures under harsher accident conditions. Although current models predict this to be low, some allowance perhaps may have been prudent for errors in the current fuel behavior models for accident conditions. In any case, the containment evaluation would profit by applying uncertainty factors to the assumed source from the core.

(3) Site boundary doses for DBAs, as presented, are about a factor of 1500 below 10 CFR 100 guidelines for the base case reactor building. Building Alternatives 2 through 5B could only result in even lower predicted doses. As a result, the key point is the validity of the base case dose estimates. If these are accepted, motivation for considering tighter containments is greatly reduced. Therefore, review and evaluation of the documentation for the base case dose estimates is highly pertinent, and perhaps should have been undertaken prior to evaluation of containment alternatives.

(4) The subject report contains very little technical detail on matters which determine site boundary doses. The technical basis for the base case is not presented. It is implied that base case results were taken from the PRA. Site boundary doses for Alternatives 2 to 5B were obtained by a "scaling" procedure using the base case as an initial standard. The precise scaling method is not clear (see pages 4.2.1-3 and -4 of the Study). For these reasons, a technical review of site boundary dose estimates is not possible based on material presented in this report.

References for Section III.3.A

Maneke, J. (1988), "Review of Radionuclide Retention Models and Data Applicable to the Reactor Building of the MHTGR," DOE-HTGR-88236, Nov. (AT)

Myers, B.F. (1991), "The Effect of Hydrolysis on MHTGR Fuel," DOE-HTGR-88486, August (AT)

PRA (1987), "PRA for the Standard MHTGR," DOE-HTGR-86-011, January

Reg. Guide (1978), "Design, Testing, and Maintenance Criteria for Post Accident Engineered Safety Feature atmosphere Cleanup System Air Filtration and Adsorption Units for LWRs," NRC Reg. Guide 1.52, Rev. 2, March

Reg. Plan (1987), "Regulatory Technology Development Plan for the MHTGR," DOE-HTGR-86-064

Wichner, R.P. (1991), "Fission Product Plateout and Liftoff in the MHTGR Primary System: A Review", NUREG/CR-5647, April

Wichner, R.P., and W.P. Barthold, (1992), "Evaluation of MHTGR Fuel Reliability," NUREG/CR-5810, July

Wichner, R.P., et. al., (1983), "SBLOCA Outside Containment at Browns Ferry Unit One: Volume 2," NUREG/CR-2672, September

B. Cost Estimate Evaluation

Five reactor building alternatives were evaluated by Bechtel National, Inc., in Containment Study for MHTGR, DOE-HTGR-88311. The major features of these alternatives are summarized in Table 1-1 (Section III.1). The purpose of this evaluation is to consider the completeness and accuracy of the capital cost impact of various reactor/confinement concept modifications as presented in the reference Containment Study (DOE-HTGR-88311). The cost impacts are presented in the reference document as cost differential only, i.e., the cost variation from the base MHTGR reactor building concept. The incremental costs were estimated by determining the physical quantities of materials impacted by the changes in confinement concepts and then applying unit commodity rates, composite labor crew rates, and unit installation manhours to these incremental quantities of materials. This appears to be a reasonable way to assess the cost contribution of confinement schemes in a scoping study. The adequacy of the proposed design changes with respect to achieving enhanced confinement objectives is discussed elsewhere.

Most of the unit material cost, the unit labor hours, and the unit labor costs used by Bechtel in the Containment Study were within bounds suggested by Gas-Cooled Reactor Associates (GCRA) for studies of modular high temperature gas-cooled reactors in DOE-HTGR-87-086. The unit cost and unit labor values suggested by GCRA were converted to 1987 dollars and compared in Exhibits 1-3 to the values used in the Containment Study. Comparative unit values from GCRA were not available for all items used in the Containment Study, the level of detail was not the same in both documents for the unit values.

Exhibit 1 compares the available unit material costs. The major variations for the unit material cost is in formwork for cylinder concrete (Bechtel value about 10% of the GCRA cost document value; however, this item is not a cost driver), in containment head (Bechtel value about 40% higher), and in overhead structural steel (Bechtel value about 40% of the GCRA cost document value). Exhibit 2 compares unit labor costs ($/MH), which agree, within 5%, for all unit labor costs which were available for comparison.

Exhibit 3 compares unit labor manhours. Major variations are seen in unit labor manhours for formwork for cylinder concrete (Bechtel value about 25% of the GCRA low value; however, this item is not a cost driver) in overhead structural steel (Bechtel value about 45% lower than the low value suggested by GCRA), and in steel liner plate (Bechtel value about half of the GCRA low value). These variations should not greatly impact the overall results of the cost of containment options presented in the reference containment document. However, some of the cost drivers, such as the cost of cadwelds, the cost of penetrations (process, HVAC, electrical) and the cost of the reactor cavity cooling systems (which are impacted by the various containment schemes) could not be compared with GCRA values.

The percentage of direct cost which was allocated to indirect cost was found to be different in the containment study relative to baseline MHTGR cost estimate. The original baseline cost estimate used 44% of the direct costs as indirect cost; the containment study used 27% of the incremental direct cost as incremental indirect cost for each containment option evaluated. This difference in the incremental indirect cost percentage would not affect the relative cost comparison between containment options. However, we believe the 44% estimate to be more realistic, in which case we would suggest increasing the incremental cost estimates by the difference, i.e. 44 - 27 = 17%.

Although there was not enough information available to make informed judgments about all of the cost elements in the alternative designs, since those elements that were evaluated were generally in reasonable agreement with estimates from another independent study (GCRA), we would conclude that the estimates quoted in the reference report for incremental capital costs are reasonable.

There may be significant differences, however, both in absolute and relative magnitudes, between the incremental capital and total costs for the various alternatives. Other costs that should be included in a detailed evaluation are the cost differences for licensing, maintenance, and operation. For example, the maintenance and operating costs for the air-cooled and water-cooled RCCS designs would clearly be different. Evaluations of operating costs should also consider the impact of the designs on plant operation, as well as on containment system operation (for example, to consider accessability for both operations and maintenance).

EXHIBIT 1

4x350 MW MHTGR CONTAINMENT STUDY
1987 DOLLARS

UNIT MATERIAL COSTS ($)

	UNITS	CONTAINMENT STUDY	GCRA COST DOCUMENT
REACTOR		DOE-HTGR-88311	DOE-HTGR-87-086 (REV 2)
EXCAVATION & DEWATERING(EXPN. TUNL)	CY	6.00	?
FREEZE & EXCAVATION (SILO)	LOT	100,000.00	
BASEMAT CONCRETE	CY	80.00	75.28
REBAR	TN	560.00	576.19
FORMWORK	SF	1.75	1.86
CYLINDER CONCRETE	CY	80.00	75.28
REBAR	TN	560.00	576.19
EMBEDS	LB	2.55	2.60
FORMWORK	SF	0.22	1.86
DOME CONCRETE	CY	80.00	75.28
DOME FORM OUTSIDE	SF	1.75	1.86
DOME REBAR	TN	560.00	576.19
CONTAINMENT HEAD	TN	4,000.00	2,853.08
ADD'L INT. STRUCTURE & SHIELDING	LOT	350,000.00	
STRUCTURAL STEEL OVERHEAD	TN	1,200.00	2,853.08
OTHER CONCRETE (EXP. TUNNEL)	CY	80.00	75.28
REBAR	TN	560.00	576.19
FORMWORK	SF	1.75	1.86
CADWELD	EA	47.00	?
STEEL LINER PLATE 1/4" THK	TN	3,000.00	2,853.08
REDUCE FORM FOR LINER	SF	0.22	
DOME LINER PLATE	TN	3,000.00	2,853.08
VENT RELIEF VALVES	EA	44,000.00	?
HEPA FILTERS	LOT	825,000.00	
WELDED SEAL PLUGS	EA	15,000.00	?
WELDED SEAL ACCESS PLUGS	EA	2,000.00	?
PENETRATIONS (PROCESS/HVAC/ELEC)	LOT	3,612,000.00	
AIR TIGHT PERSONNEL DOORS	EA	2,000.00	?
ALLOW FOR LEAKAGE INSPECTION (RSCC)	EA	0.00	?
ALLOW FOR BLDG. SERV (EXP. SPACE)	CFT	1.00	?

EXHIBIT 2

4x350 MW MHTGR CONTAINMENT STUDY
1987 DOLLARS

UNIT LABOR COSTS ($/MH)

	UNITS	CONTAINMENT STUDY	GCRA COST DOCUMENT
REACTOR		DOE-HTGR-88311	DOE-HTGR-87-086 REV 0
EXCAVATION & DEWATERING(EXPN. TUNL)	CY	18.00	17.68
FREEZE & EXCAVATION (SILO)	LOT	18.00	17.68
BASEMAT CONCRETE	CY	21.00	20.47
REBAR	TN	21.00	20.47
FORMWORK	SF	21.00	20.47
CYLINDER CONCRETE	CY	21.00	20.47
REBAR	TN	21.00	20.47
EMBEDS	LB	21.00	20.47
FORMWORK	SF	21.00	20.47
DOME CONCRETE	CY	21.00	20.47
DOME FORM OUTSIDE	SF	21.00	20.47
DOME REBAR	TN	21.00	20.47
CONTAINMENT HEAD	TN	23.87	23.16
ADD'L INT. STRUCTURE & SHIELDING	LOT	23.87	23.16
STRUCTURAL STEEL OVERHEAD	TN	23.87	23.16
OTHER CONCRETE (EXP. TUNNEL)	CY	21.00	20.47
REBAR	TN	21.00	20.47
FORMWORK	SF	21.00	20.47
CADWELD	EA	21.00	?
STEEL LINER PLATE 1/4" THK	TN	23.87	23.16
REDUCE FORM FOR LINER	SF	21.00	
DOME LINER PLATE	TN	23.87	23.16
VENT RELIEF VALVES	EA	24.88	24.53
HEPA FILTERS	LOT	24.88	24.53
WELDED SEAL PLUGS	EA	24.88	?
WELDED SEAL ACCESS PLUGS	EA	24.88	?
PENETRATIONS (PROCESS/HVAC/ELEC)	LOT	24.88	?
AIR TIGHT PERSONNEL DOORS	EA	24.88	?
ALLOW FOR LEAKAGE INSPECTION (RSCC)	EA	24.88	?
ALLOW FOR BLDG. SERV (EXP. SPACE)	CFT	24.88	?

EXHIBIT 3

4x350 MW MHTGR CONTAINMENT STUDY

UNIT LABOR MANHOURS (MH)

	UNITS	CONTAINMENT STUDY	GCRA COST DOCUMENT	GCRA COST DOCUMENT
REACTOR		DOE-HTGR-883	DOE-HTGR-87-086 (REV 2)	
			LOW	HIGH
EXCAVATION & DEWATERING(EXPN. TUNL)	CY	2.07	?	?
FREEZE & EXCAVATION (SILO)	LOT	3,000.00		
BASEMAT CONCRETE	CY	2.07	2.20	3.80
REBAR	TN	28.20	25.00	40.00
FORMWORK	SF	0.71	0.45	0.75
CYLINDER CONCRETE	CY	2.07	2.20	3.80
REBAR	TN	37.60	25.00	40.00
EMBEDS	LB	0.07	0.07	0.15
FORMWORK	SF	0.11	0.45	0.75
DOME CONCRETE	CY	3.70	2.20	3.80
DOME FORM OUTSIDE	SF	0.71	0.45	0.75
DOME REBAR	TN	40.00	25.00	40.00
CONTAINMENT HEAD	TN	50.00	54.00	70.00
ADD'L INT. STRUCTURE & SHIELDING	LOT	2,000.00		
STRUCTURAL STEEL OVERHEAD	TN	30.00	54.00	70.00
OTHER CONCRETE (EXP. TUNNEL)	CY	2.82	2.20	3.80
REBAR	TN	37.60	25.00	40.00
FORMWORK	SF	0.71	0.45	0.75
CADWELD	EA	2.40	?	?
STEEL LINER PLATE 1/4" THK	TN	28.20	54.00	70.00
REDUCE FORM FOR LINER	SF	0.11		
DOME LINER PLATE	TN	50.00	54.00	70.00
VENT RELIEF VALVES	EA	1,200.00	?	?
HEPA FILTERS	LOT	450.00		
WELDED SEAL PLUGS	EA	200.00	?	?
WELDED SEAL ACCESS PLUGS	EA	50.00	?	?
PENETRATIONS (PROCESS/HVAC/ELEC)	LOT	314,000.00		
AIR TIGHT PERSONNEL DOORS	EA	30.00	?	?
ALLOW FOR LEAKAGE INSPECTION (RSCC)	EA	5,100.00	?	?
ALLOW FOR BLDG. SERV (EXP. SPACE)	CFT	0.01	?	?

4. Evaluation of Baseline Reactor Building Cost Estimate

The cost of the baseline reactor building in the Containment Study for MHTGR, DOE-HTGR-88311, is not explicitly given but is embedded in Account 21 (Structures and improvements), which is shown in Table 4.3.3-1 of the reference report. The cost for Account 21 is $110.6M, which is for the Nth of a kind (4x350 MWt) reactor plant. In addition to the cost of the reactor building, Account 21 includes the cost of the turbine building, reactor auxiliary building, reactor service building, radwaste building, operations center, and a myriad of lesser expensive buildings. The Containment Study references as its cost source the base cost estimate for the MHTGR (DOE-HTGR-87-086, Rev. 0). In this document, Account 21 is broken down into its elements (Table D-4). The total of Account 21 is shown to be 110.6 M% for the Nth of a kind plant, as indicated above. The cost of the reactor building only, Account 212.1 is shown to be $59.6M. The cost of the reactor complex, Account 212 (which includes the reactor building, reactor service building, radwaste building, reactor auxiliary building, and personnel services building) is derived to be $77.4M.

The cost of the 4x350 MWt MHTGR reactor plant was revised in the MHTGR Cost Reduction Study Report (DOE-HTGR-88512), dated October 1990. In this review of the cost, it was discovered that the original cost estimate for the reactor building (DOE-HTGR-87-086, Rev. 0) erroneously reflected the cost for four excavations per reactor, instead of the cost of one excavation per reactor. The revised cost of the 4x350 MWt reactor building was $58.3M for the reactor building (Account 212.1) $75.2M for the reactor complex (Account 212), and $99.4M for all the structures and improvements (Account 21) for a lead plant with all costs express in 1990 dollars. The costs for the Nth of a kind reactor plant were also provided for the reactor complex, $71.5M (Account 212) and the cost for the total structures, $99.4M (Account 21). The cost of the Nth of a kind reactor building only was not given but was derived by ratioing to be $55.4M.

The revised cost (expressed in 1990 dollars) for the baseline reactor building ($58.3M for the lead plant; $55.4M estimated for Nth of a kind) is felt to be accurate for this level of a cost study, particularly since it has been through the second review by a leading architect engineering firm (Bechtel National, Inc.).

5. Assessment of Vent Rates from Reference Design Reactor Building

The design value of (maximum) leakage rate from the reference design reactor building (Alternative 1) is 100%/day. This building is characterized as being "vented with no filtering." The 100% (by volume) of the building air inventory per day refers to the case where there is no substantial positive pressure in the reactor building. The design of the dampers or louvers is to be such that they open during a large primary or secondary system leak, but close again (via gravity) after the building overpressure is relieved. Two sets of parallel dampers will be designed to open on 1 psid or less, where only the pressure differential is the motive force. No controller action is involved. They are to be constructed of lightweight metal such that even if the normal hinge mechanisms were to jam, they would be forced open and provide a vent path for pressures only slightly higher than 1 psid (Reference: PSID Amendment 11, R 6-3). DOE will propose a safety classification for the dampers, which must function to provide overpressure protection for the safety-grade RCCS (PSID Amendment 11, R 6-10).

DOE analyses which support the 100%/day design leakage rate assumption for the vented building are not available for review, but that rate has been characterized (by DOE) as being conservatively large. The rate is claimed to be based on nuclear industry practice and experiments. It would appear, however, that the actual rate would be strongly dependent on how much gas is emanating from the reactor vessel, as well as on the meteorology. References to equations used are noted in PSID Amendment 11, R 15-12: the Reactor Safety Study (WASH-1400), the AIPA (GA-A15000), and the BNWL Containment Systems Experiment (BNWL-2108). Additional information is available in DOE-HTGR-88009 ("Reactor Complex Building Design Description, MHTGR Plant").

Another consideration that ties in with an integrated average %volume/day building leakage would be the estimates of fission product retention and transport. DOE has committed to submit additional analyses which "will take into account developing information on fission product retention and transport as applicable to conditions in the reactor and reactor building" (reference PSID Amendment 11, R 6-10; see also RTDP TDN 6-1). Additional background information is available in DOE-HTGR-88236 ("Review of Radionuclide Retention Models and Data Applicable to the Reactor Building of the MHTGR").

There are four other important considerations related to the reactor building vent/leakage characteristics:

1. The opening of the dampers upon rapid primary system depressurization is accounted for in DBE/SRDC 10 in the PSID. These scenarios assume that the primary circulating activity, the liftoff of a portion of the plated-out activity, and the release of some small fraction of fuel activity all contribute to the radiological release. Another low-probability scenario that should be considered is the one involving a steam generator tube rupture and sustained moisture ingress into the core, which could potentially result in additional fuel damage, fission product release into the primary circuit, and a subsequent primary system steam-induced overpressure, followed by depressurization through a stuck-open relief valve;

2. In the case of damage to the RCCS which provides a direct path for venting the reactor cavity atmosphere to the RCCS stack, the effective building vent rate would be increased. DOE estimates for worst-case increases in offsite doses are factors of 25 and 8 for thyroid and whole body, respectively (PSID Amendment 4, R G-11.D);

3. The design leakage rate assumption of 100%/day presumes that the dampers reclose successfully after venting excess reactor building pressure. While the two parallel sets of dampers provide redundancy for opening in case of reactor building overpressure, if either one of the two failed open due to damage from a rapid release, it could potentially cause a larger leak rate. However, if only one of the two "stuck" during an overpressure, it would be likely for the pressure to be relieved successfully by the other, reducing the chances of any damage to the other which could cause it to fail open; and

4. In DOE's review of the LWR's General Design Criteria (GDC), several of the criteria were judged to be "not applicable" to the Standard MHTGR (reference) Design for "containment buildings" because it was in fact a moderate leakage reactor building, not a containment building. In case an alternative to the reference design is selected which includes filtering and/or leak reduction mechanisms, the review of these criteria (PSID

Amendment 6, G 3-1) should be revisited. Specifically, GDCs 38-43 and 50-53 should be reviewed. If these additional criteria are judged to apply, then additional licensing oversight would be required for both the design and operation phases, and its impact would need to be factored into cost estimates.

6. Assessment of Structural Differences Among Containment Alternatives

A detailed assessment of the structural differences between the containment alternatives in the Containment Study (HTGR-88311) is not feasible since there is very little information presented about the different structures. There are some general observations that can be made, however, regarding the features that are added as the containment evolves from the reference case (vented, unfiltered) to the one approaching the design of a conventional LWR containment (unvented, low leakage, high pressure).

1. For designs which utilize a water-cooled reactor cavity cooling system (RCCS), the additional weight of the water-filled radiator panels (vs. air) would make the seismic forces larger and more difficult to design for (especially considering the very high reliability required of this safety grade, ultimate heat sink). The makeup water tanks required for the water RCCS also add weight and inertia to the total structure and are typically difficult to design with respect to assuring very high reliability.

2. For designs which utilize steel liners in the concrete reactor building in order to reduce leakage rates, the liners would also add weight (inertia) to the structure, which would add to the seismic forces and the probability of structural failure.

3. The two designs requiring filtering of the vented gas (Alternatives 2 and 3) have the HEPA and charcoal bed filters located in the building roof structure. These would add to the seismic forces of the building, and increase the chances of structural failure.

4. For the two designs which include an expansion volume option (Alternatives 4a and 4b), there is the addition of a major "building" (the expansion volume) which adds to the structural and foundation loads. The large check valves which must ensure isolation of the module undergoing depressurization from the other normal modules must be designed for seismic loading as well.

7. Identification of Tests for Leakage and Plateout Assumptions

A. Leakage Testing Regulations

NRC guidance on containment leakage testing for LWRs is given in 10 CFR Part 50, Appendix J. This guidance should be applicable to MHTGR reactor building's alternative designs characterized as "unvented, moderate-to-low leakage" (5% to 1%/day). In the case of the vented buildings, however, certain requirements would clearly not be applicable, and NRC would need to translate the intent of Appendix J into a modified set of specific requirements. For example, there are no "containment isolation valves (CIVs)" as such in the vented designs. However, the dampers used for pressure-relief in case of rapid depressurization and steam line rupture accidents are to perform a function similar to CIVs after the pressure relief is completed.

Appendix J requires that the procedures for testing unvented containments follow ANSI N45.4-1972, "Leakage Rate Testing of Containment Structures for Nuclear Reactors." Required tests are divided into three categories, A, B, and C. Type A Tests are intended to measure the overall integrated leakage rate just before initial reactor operation and periodically thereafter. Type B Tests are to detect local leakage in areas of penetrations, closures, doors, etc. Type C Tests are intended to measure leakage of containment isolation valves.

A summary of NRC regulations on LWR containment leakage testing is given the Standard Review Plan (SRP) section 6.2.6 (Rev. 2, July 1981).

In the case of the General Design Criteria (GDC) from 10 CFR 50 Appendix A, DOE has taken the position that the two GDCs most closely related to leakage testing (GDC 52 - Capability for containment leakage rate testing, and GDC 53 - Provisions for containment testing and inspection) do not apply to the reference (vented) reactor building design because "there is no comparable structure" (Ref. PSID R G.3-1). While these criteria would clearly apply to the unvented designs, there may need to be some considerations applicable to the vented designs as well, assuming the 100%/day (maximum) leakage is required to meet EAB dose limitations.

In summary, depending on the features of the reactor building design chosen, certain revisions of the applicable containment licensing guidance will be needed to provide MHTGR designers latitude appropriate to the enhanced safety features of the MHTGR.

B. Leakage Testing Methods

Two excellent, detailed technical discussions of containment building leakage testing may be found in the Nuclear Safety Information Center ORNL-NSIC-5 report on U. S. Reactor Containment Technology, and the article in Nuclear Safety Vol 7, No. 2 (Winter 1965-1966) entitled "Leak Testing of Reactor Containment Systems." In particular, Chapter 10 (Performance Tests) of NSIC-5 discusses alternative methods along with the advantages, disadvantages, and reactor experiences with a wide variety of testing options. This is especially valuable in view of the need to adapt and certify testing methods developed for LWRs to the MHTGR case(s). Since design leakage rates of the various MHTGR reactor buildings are considerably larger than the typical LWR design leakage rates of ~0.1%/day, some of the problems encountered in LWR testing, such as compensating for changes in average inside air temperature, will not be as significant.

Of particular concern for the MHTGR is the potential need to test the 100%/day leakage rate requirement, which applies to the vented building after a depressurization accident pressure transient has been relieved. In order to test any non-zero leakage rate, there would need to be some assumed pressure driving force, and apparently none has been defined for the 100%/day leak case. It is recommended that some conservative positive differential pressure should be determined from analysis of the EAB dose-limiting case severe accident scenario, and thus specified as a design leakage test pressure.

Of all the methods described in the NSIC report, the most appropriate for the 100%/day leakage case appears to be the one involving measurement of makeup air (Section 10.4.10.1). In this method, the air flow rate required to sustain the rated differential pressure is assumed equal to the leakage rate. In this, as well as other recommended methods, it may be necessary to account for changes in average containment air temperature. A detailed evaluation of the wide range of potential methods should be made. It is, however, beyond the scope of this task.

C. Identification of Tests for Plateout Assumptions

Three types of information are required for verification of the plateout assumptions for the reactor building:

a. The rate, timing, temperature, and chemical form of material entering the building during the accident need to be specified. These items define the boundary conditions for evaluation reactor building deposition. Characterization of radioactive particles is needed in order to estimate deposition rates.

b. Building conditions during the accident need to be specified. Needed are air and wall temperatures and humidity levels, which affect sorptivities on solid surfaces. Humidity levels and condensed water ingress to the building are also needed for estimation of the degree of washoff and determination of radioactive inventory in water pools. Radioactivity levels are needed because they impact iodine volatility levels. Gas velocities are required for estimation of particle deposition and mass transfer rates.

c. The inherent deposition mechanisms in the building need to be identified and quantified. These include chemisorption on the various types of surfaces in the building, condensation, dissolution in water, aerosol deposition, and possible chemical interactions with building surfaces. The rate of formation of organic iodide under the building environmental conditions needs to be assessed.

Item "a" must be provided by the analysis of the primary system for the conditions of the accident.

Item "b" may be provided by the output of a suitable containment code, assuming the accident sequence and the information provided by "a" is known.

The information required for "c" can in most cases be provided by reports and papers which deal with containment deposition. Most of this body of data refers to LWR accident conditions. It is not clear if all the required information to perform the MHTGR analysis is available. If not, bench scale tests may be required.

8. Recommendations for Containment-related R&D & Tests (RTDP)

Recommendations for R&D tasks and testing that should be considered for DOE's Regulatory Technology Development Plan (RTDP) are split into two categories: 1) those which are directly related to the reactor or containment building (CB) design and performance; and 2) those with a potential impact on CB design requirements.

A. R&D tasks and testing directly related to CB design and performance.

1) Studies and validation of the features of the CB that are peculiar to multicompartment underground silos -

a) models for RCCS heat rejection both for normal operation and for long-term loss of function of the RCCS and/or other auxiliary heat removal systems during postulated accidents. Experimental validation of RCCS performance should include the development of tests which could be used for on-line performance validation in the plant. To validate calculations of accidents with loss of RCCS function, models for conduction of heat to the concrete silo and surrounding earth should also be checked. In long-term loss of forced convection (LOFC) events without the RCCS, heat removal via conduction to the soil surrounding the silo is not significant within the first week or so of the accident; however, for longer-term scenarios, conduction to the soil becomes important in the calculation of vessel temperatures. In the original PSID design, there was insulation in the RCCS between the downcomer and riser (to reduce regenerative heating). ORNL and BNL studies both showed that regenerative losses would be very small without insulation, and that for severe accidents involving loss of function for the RCCS, not having the insulation there would increase the heat removal rate from the vessel considerably. Validation of RCCS performance is also needed for accident scenarios in which there is steam or dust in the space between the vessel and RCCS. Degraded RCCS performance is one of the most sensitive parameters in determining vessel overtemperature in long-term LOFC accidents. Serious vessel overheating can occur locally if portions of the RCCS panel are not cooled;

b) consideration of alternative air-cooled RCCS designs for CB options involving the need to withstand higher pressures in depressurization accidents. It is advisable to avoid using water-cooled RCCS designs due to their lower reliability. (see discussion in Section III.2)

c) operator accessibility to critical equipment under postulated severe accident conditions [e.g., for restoration of reactor cavity cooling system (RCCS) function]. Certain features of the underground silo design may make some critical areas less accessible than for conventional designs in severe accidents involving high radiation fields and high ambient temperatures. On the other side of the coin, accessibility may be a sabotage concern;

d) ventilation characteristics [with and without HVAC operating properly], including air access to lower part of vessel [via a damaged RCCS?] in case of "graphite fire scenario," and other analyses and proposed tests related to graphite fires. The potential for a graphite fire in the MHTGR is vanishingly small; however, it remains a "safety" issue because of the Chernobyl accident;

e) seismic perturbations and the response of tall underground concrete silos. These may be crucial considering the very low probabilities for initiating events that are characteristic of beyond design basis events (Ref. synopsis of DOE seismic R&D plan and seismic event characterizations in PSID - HTGR 86-024, R G-15.F and R G-25); and

f) scenarios for flooding via leakage from the water table or pressurized water cooling systems (HTS or SCS). Very low probability scenarios involving flooding of the core may result in recriticality accidents.

2) Study of other CB features that affect DBAs or other accident scenario outcomes -

a) Blowout panel performance (including analysis/testing of the leakage through other possible paths between the reactor cavity and its neighbors). Performance of pressure barriers during primary system depressurization and steam line and feedwater line breaks may be crucial to the outcome of reactor cavity pressurization accidents with respect to the integrity of the RCCS;

b) Analysis/testing of effective CB leakage rates for fission product releases (including wet and dry depressurizations at various postulated rates, and leakages/releases during long-term severe accident scenarios after the primary system has depressurized); and

c) Analysis (including PRA) of benefits of alternative filtering options. For example, the use of filters on the primary system relief valve(s) vent train, which appears to be the most likely path for depressurization, may be more effective than the building (roof level) filters considered in the containment study (see discussion, Section 2).

B. R&D tasks and testing affecting CB design.

Another set of recommended RTDP tasks affect CB design in that they contribute to the determination of functional requirements for the CB or reactor building. These tasks are not as well-defined as those in part A because they involve such things as extremely low probability mechanistic source terms, relatively vague concepts such as "weak" or sub-specification fuel, and the subjective "public acceptance" of a reactor without a traditional low-leakage CB, and/or without evacuation planning or drills.

1) Prompt source term: Additional R&D is needed to determine the prompt fission product (FP) release from the primary system due to a rapid depressurization event, which would be characterized by lift-off (and wash-off, in the case of a wet depressurization) of a portion of the radionuclides held up, deposited, and plated out in the primary loop.

2) Delayed source term: CB functional requirements should be based in part on the NRC's "Containment Design Controlling Event Sequence" (CDCES), which involves a steam generator tube rupture and loss of forced convection (LOFC) accident, with a subsequent primary system overpressure that eventually results in a rapid depressurization through a failed-open relief valve. Depending on the timing of the depressurization, there may be additional fuel failure due to overheating and hydrolysis. The CDCES scenario may include a substantial release of radionuclides from weak fuel, which is postulated to fail at temperatures likely to be reached in this accident, e.g. 1200°C, but lower than the 1600°C "limit." Additional releases from weak fuel may be attributed to a successful

hydrolytic attack on weak particles. (Ref. B. M. Morris to S. Rosen letter of May 9, 1990); and

3) Development and verification of fuel performance models is needed to characterize "weak fuel." R&D on the fuel manufacturing and QA/QC process is needed in addition to fuel performance testing to determine the failure characteristics and probabilities of occurrence of weak fuel.

9. Summary Assessment of DOE Containment Study for Resolution of MHTGR Containment Needs

In addition to summarizing earlier containment discussions and evaluations from work on FIN A9477 in previous fiscal years, Part III, Sections 1-8 of this letter report presents assessments of various aspects of the DOE Containment Study (HTGR-88311).

Section 1 summarizes the DOE assumptions used for the five major design alternatives, starting with the reference (PSID) design of a vented, unfiltered reactor building. Sequentially, in four steps, the design alternatives progress to a high pressure, low-leakage containment building (CB) similar to a conventional LWR CB design. Using the PSID assumptions for source terms (releases to the reactor cavity), the design-basis accident (DBA) and worst-case releases to the exclusion area boundary (EAB) are both minimal for the reference design and, of course, even less for the progressively tighter designs. As noted, however, the more pertinent issues of clarifying the proposed source term and considering the possibilities of larger releases to the reactor cavity were not addressed. Currently, there are no postulated accidents within the design basis or emergency planning basis categories that would result in releases which exceed the allowable 10 CFR 100 limits at the EAB. Incremental attenuation measures such as HEPA and charcoal filters may provide sufficient additional safety margins for the bounding accident cases should they be deemed necessary. To properly consider the potentially larger source terms, however, it is recommended that there be a reevaluation of both the prompt and delayed releases based on NRC's concerns about "weak fuel."

Section 2 briefly describes three proposed design features that may enhance the design capabilities.

In the evaluation of the DOE dose reduction estimates (Section 3.A), several areas of concern were noted. Specifically, the apparent ambiguity in the definition of a "standard particle" may have lead to an underestimation of the source term. The DOE report also suffers from a lack of technical detail about how site boundary doses were estimated. The cost estimate evaluation (Section 3.B) was cursory due to limited documentation on the assumptions and methodology used for the incremental cost estimates. Evaluations of the structural differences and cost estimates for the design alternatives were found to be reasonable to the extent they can be checked. It was pointed out, however, that additional costs such as those for operation, maintenance, and licensing, could add significantly to the total (differential) costs, and should be considered in a more detailed cost evaluation.

The evaluation of the baseline reactor building cost estimate (Section 4) concluded that the Study's estimate appeared to be reasonable.

The design building vent or leakage rate estimate was assessed in Section 5. Further information is needed on the rationale and quantification of the "100%/day leakage rate" specification. The structural differences assessment (Section 6) noted some of the structural problems that would need to be considered as the building design alternatives approached that of a standard LWR containment.

In the assessment of building leakage testing regulations and methods (Sections 7 A and B), the need for special interpretations by NRC was noted, and a brief survey of applicable leakage testing methods was given. Tests needed for verification of plateout assumptions were listed in Section 7 C.

In Section 8, recommendations were made for containment-related R&D tasks that should be included in the Regulatory Technology Development Plan (RTDP).

In summary, it appears that the satisfactory identification of the source term to the reactor building, including a reasonable margin of error, is the principal uncertainty in the selection and evaluation of an appropriate building design alternative. The current reliability problem with the U.S. fuel should factor in to the selection of the margin of error determination for the source term. A determination of the sensitivity of the containment design requirements to parametric variations of the source term over this error band should be useful in selecting a satisfactory building design alternative.