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ORNL FOREIGN TRIP REPORT

ORNL/FTR-3477

DATE: December 4, 1989

SUBJECT: Report of Foreign Travel of S. J. Ball, ORNL Manager of NRC-Sponsored HTGR Safety Studies, Instrumentation and Controls Division

TO: A. W. Trivelpiece

FROM: S. J. Ball

PURPOSE: To visit nuclear installations in England, France, and West Germany to obtain primary source information needed for reevaluating DOE's research program plan for the modular HTGR in the areas of primary system components, reactor operations, fuel performance, reactor physics, heat transfer and fluid flow, fission product transport, safety analysis, and licensing criteria.

SITES VISITED:

11/8/89	Heysham-2 Reactor Heysham, England	Dr. Neil W. Davies, Thermal Reactor Coordination Manager, UKAEA
11/9/89	Nuclear Installations Inspectorate (NII), Bootle, England	Dr. Derek Goodison, Branch Chief
11/10/89	UKAEA Laboratories, Harwell, England	Dr. John R. Askew, Director, Gas-Cooled Reactor Program
11/13/89	Commissariat à l'Énergie Atomique (CEA), Fontenay aux-Roses, France	Mr. Daniel Bastien, Coordinator for Gas-Cooled Reactors
11/14/89	Centre d'Études Nucleaires (CEN), Grenoble, France	Mr. Jean-Francois Veyrat, Chief of Service
11/15-16/89	KFA, Jülich, FRG	Dr. Erwin Balthesen, Director, HTR Development
11/17/89	THTR-300 Reactor Site Hamm, FRG	Dr. Rüdiger Bäumer, VEW, Plant Manager

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ABSTRACT

The traveler was asked by the U.S. Nuclear Regulatory Commission (NRC/RES) to travel with Dr. Peter M. Williams, NRC MHTGR Project Manager, to assist in obtaining information from researchers and licensing authorities in the United Kingdom and Western Europe relevant to the NRC's ongoing evaluation of the DOE Modular HTGR (MHTGR) research program plan and licensability concerns. The NRC-sponsored ORNL program for HTGR safety reviews, of which the traveler is manager, has made significant use of foreign resources in conducting safety research, developing independent safety analysis capabilities, and assisting NRC in preparation of safety analysis reports. The additional information derived on this trip from detailed discussions with researchers and licensing authorities, laboratory and reactor site tours, and literature received will be very valuable in carrying out the NRC program. Specific information and insights were obtained in the areas of primary system component performance, reactor operations, control and safety system design and performance, fuel performance and fission product transport, safety analysis, heat transfer and fluid flow, reactor physics, advanced designs, and licensing criteria and methods.

HTGR RESEARCH AND LICENSING TRIP TO ENGLAND, FRANCE, AND WEST GERMANY

Heysham-2 AGR

The site visit to the Heysham-2 Advanced Gas Reactor (AGR) provided useful information on AGR operation, analysis, licensing, and design features and problems, much of which is applicable to HTGRs. The AGRs are a good example of how the evolution of a design can result in a much-improved, smoother operation. Recently, however, a serious licensing problem has cropped up. Previously, British reactors were licensed on the basis of "deterministic calculations of maximum credible accidents," while for the newest AGRs, probabilistic risk assessments (PRAs) are used. In the AGRs, the predominant risks are from refueling accidents, and the sum of the faults must be $<10^{-6}$/y. Heysham-2 is now coasting down in power, unable to refuel, because the calculated probability of dropping a fuel assembly during on-line refueling is too large. The problem surfaced in a safety review, wherein it was decreed that since two "independent" digital safety systems used the same type of processor hardware and the same programming language (although designed by different groups), the probability of failure is a factor of 10 higher than claimed. The Heysham case is substantially weakened by the fact that an assembly was dropped during a refueling at another AGR. To license or not to license, based on low-probability PRA numbers, appears to be a very risky business due to the large uncertainties in the calculations and the vulnerability to criticism.

In the AGRs, the large graphite moderator blocks surrounding the (clad) fuel assemblies are cooled to nearly the inlet gas temperature by a major (60%) sidestream flow that later joins with the rest to cool the fuel. Hence, the bulk moderator temperature is relatively independent of power, so even though the moderator temperature coefficient of reactivity is positive and a factor of 8 larger than the (negative) fuel coefficient, the power coefficient stays negative throughout the fuel life. For fast transients, the fuel coefficient dominates and effectively terminates the spectrum of "allowed" reactivity transients. Total-loss-of-flow accidents are kept to an acceptably low probability by having four independent cooling system quadrants, any one of which could provide adequate shutdown cooling.

Heysham-2 has an interesting on-line computing system, which calculates the varying risk of fuel damage as a function of post-trip cooling equipment condition and availability. For configurations in which the base estimate for risk increases by a factor of 10, routine maintenance is allowed, while for factors of 100, emergency maintenance and 36-h shutdowns are mandated.

The 88 control rods are positioned automatically to keep assembly outlet temperatures to within 10°C of the average, with manual assistance from operator control of the assembly inlet flow orifices (or "gags"). Individual region power control is limited by calculated pellet-clad interaction failures that would result from the design-basis depressurization accident. Heysham-2 had tried a more automated plant control system (for startup, power maneuvers) but went back to the more traditional one because of problems with unreliability.

Coolant chemistry for the AGR primary system (carbon dioxide) is quite different from that of the HTGRs (helium). The normal moisture level is ~120 ppm (vs <1 ppm) and they make sure it stays >10 ppm to avoid dry-atmosphere friction problems, and have a high-level trip at 350 ppm. Methane is added to inhibit corrosion, which is considerable (20% graphite weight and strength loss is expected over the plant lifetime). The moisture detectors (fogged mirrors) are satisfactory. Primary coolant leakage is quite high (~1%/day) through particulate filters, with provisions for diverting leaking coolant through charcoal filters in an accident. The primary system is normally very clean, as they are able to do hands-on maintenance of the steam generators and circulators. All AGRs recently added a trip on high circulating activity.

Nuclear Installations Inspectorate (NII)

Plant licensing is done quite differently in the United Kingdom, but like the USNRC, the NII maintains a highly competent technical staff. The NII sets policy, reviews safety-related designs and operational problems, and grants operating licenses. It does not do independent research or safety analyses, but does advise on safety research done by UKAEA. Oversight of plant operations is limited to much higher levels (no resident inspectors, for example). One of NII's most effective administrative tools is apparently prosecution of a utility in the courts, since plant personnel tend to see themselves as personally involved in such lawsuits.

The U.K. equivalent of U.S. tech specs for a plant are a much broader set of "Operating Rules," which are subject to NII approval. At the next level are the "Identified Operating Instructions," which are more like the U.S. tech specs, but which are not subject to NII approval; and likewise for the most detailed documents, known as the "Station Operating Instructions."

Plant and safety system designs accommodate the operator action guidelines, which mandate that operators must not take action for the first 5 min following a scram, and must not be required to take action in the first half hour. The NII is generally more concerned with maintenance errors than with operator errors, and is developing rules for diversity of maintenance personnel (and procedures?) to avoid common-mode failure problems.

In the area of severe accidents, several items of interest were noted: (1) NII staff pointed out that the cladding (reactivity) worth is ~$15, so it is essential that it stays intact to avoid prompt critical accidents. It was later noted at Harwell that experiments do not show the clad "falling away" when the fuel bundle is heated to very high temperatures; (2) graphite fires are excluded due to the low probability of multiple failures in the vessel; (3) water ingress accidents do not result in any significant positive reactivity insertion. A writeup was obtained on the one and only major AGR water ingress, a steam generator tube rupture at Hartlepool-1 in March 1987 that resulted in an ingress of several tons of water; and (4) finally--on the Magnox reactors at least--NII allows leak-before-break assumptions for the vessels based on inspection criteria.

I obtained a writeup on NII policy and guidelines for accident code validation and verification, which were developed during the 2.5-year-long Sizewell-B (PWR) public hearings. This policy could be useful to the NRC in drawing up a similar guide for use by

U.S. licensees. Other useful documentation received included the U.K. regulatory policy on source terms and siting, policy statements (reports) on risks and safety assessment principles, and independent safety evaluations commissioned by NII of the U.S. and FRG advanced HTGR and LWR designs.

UKAEA Harwell Laboratories

We met with Dr. John Askew and the program's Deputy Director, John Wilson, who will take over as Director when Askew leaves next spring. They commented on the very recent change in England's power plant privitization sale to exclude all nuclear plants. Reports on BBC had stated that, considering costs of decommissioning, the nuclear power cost would be three times that of coal plant power. Askew said UKAEA estimates that included complete dismantling and disposal after decommissioning (which he thought unnecessary) put nuclear costs only about 10 to 15% higher. England had not planned to build any more AGRs, and recently announced that no more PWRs will be started after Sizewell-B.

Two areas of possible collaboration were discussed. First, Askew agreed to write up a proposal for a contract to retrieve and analyze the data from Winfrith critical experiments that would be relevant to MHTGR needs. Data from heated assembly and simulated steam ingress tests are included. Second, from discussions it appeared that Harwell's primary investigator for fission product (FP) experiments, Mr. Faircloth, could be useful in either planning or evaluating the proposed experiments in the MHTGR R&D program.

Dr. Askew also described an interesting reactor design that he has proposed to IAEA: a small combined-cycle HTGR in which the use of a steam generator in place of a (direct cycle) recuperator would result in reduced pressure drop and capital cost, plus higher efficiency.

Commissariat à l'Énergie Atomique

Discussions at CEA included detailed descriptions by Mr. Bastien and other CEA staff members of the French experience with gas-cooled reactor operation, which is considerable (>150 reactor-years). Currently, only three of the Magnox plants are still operating, and the last of these is to be shut down in 1994. To date, four accidents have resulted in significant fuel damage, and the lessons learned from these were discussed. The CEA expert on plant decommissioning and dismantling, Mr. Bernard Giraudel, noted that dismantling a plant is much easier for PCRVs than for steel vessels (he has done both). This news should be of interest to Fort St. Vrain and THTR-300 owners.

Centre d'Études Nucleaires

The purpose of the CEN visit was to review the proposed DOE-sponsored fission product (FP) transport experiments in the COMEDIE in-pile loop at the Siloe reactor. Siloe is a 35-MW pool-type reactor used mainly for physics research and materials testing. The COMEDIE loop is being modified to accommodate depressurization tests to study FP liftoff from simulated steam generator (SG) tubes.

CEN personnel noted that they were providing a service to DOE wherein they are given the loop design, test, measurement, and chemical analysis requirements by DOE and do not get involved in the interpretation of the resulting data. For example, they were not aware of the geometrical differences between the MHTGR and test SG tubes (helium crossflow outside the tubes for the MHTGR vs inside for the test). Mr. Veyrat and Mr. Dupont (the COMEDIE loop project manager) have had considerable experience with FP experiments. They characterized the loop as a miniature "chemical plant" where impurity concentrations could have significant (and complex) effects on FP transport behavior. Hence the "models" used to design the test conditions will be very important, and certainly crucial to the interpretation of the results. They also stressed that detailed planning for the tests is still in progress, with the first real (with FPs) blowdown test scheduled for early 1991. Subsequent tests will look at effects of dust and moisture. They are designing a dust density probe using a laser system to measure opacity.

Kernforschungsanlage (KFA), Jülich

Dr. Balthesen arranged our meetings at KFA and the subsequent trip to the THTR. He is responsible for managing the HTR development programs in the FRG, and is very knowledgeable about most aspects of HTR activities. He works for BMFT, which is the FRG Ministry for Research and Technology.

In the FRG, the state is the licensing entity for all reactors in that state. The ministry of a state can (and does) get help from independent experts, such as the TÜVs, which have been established in 7 of the 11 states. TÜVs are also called on for generic licensing studies. For example, TÜV-Hannover wrote a safety evaluation report for the HTR-MODUL at Interatom's request (and in this case, BMFT supported its completion when the contract with Interatom was cancelled). The Federal involvement in licensing is through BMU (formerly BMI), which is the environmental ministry that supervises state authorities. Licensing ground rules in the FRG are consistent to the extent that licensing is done in steps (as in the U.S.), and while PRAs are considered, they are not used as a basis for licensing. State and federal courts also play a major role in licensing. For example, a blowdown test was planned at AVR, but a citizen's complaint resulted in a judge's decision that blocked the test.

KFA [which just recently had the Kern (nuclear) part of its name removed] previously devoted 50% of its efforts to gas reactors, while the total now is 10%. KFA is divided up into a number of "institutes;" two of the principal ones we interacted with (Nuclear Safety Research and Reactor Development) are about to merge.

In the meeting with the Institute of Nuclear Safety Research, we heard presentations (and received reports) on the current status of KFA's work in accident selection and analysis, source terms, containment, the role of operators in accident mitigation, experimental confirmation of heat transfer and fluid flow analyses, and FP transport calculations and experiments.

For the HTR-MODUL design, the main reason they have restricted the U-235 loading is to mitigate the positive reactivity effect from water ingress from a postulated single-SG-tube-break accident. Water ingress accidents are the major contributors to risk. The calculations for this design basis accident assume a reactor scram, blower trip, and SG isolation and result in acceptable consequences. There are three independent ways to trip on this accident, each with three independent channels: high moisture (800 ppm, capacitance probes), high pressure, and high power (140%). KFA maintains that generation of a burnable or explosive gas mixture is not likely in the case of a single SG tube break. KFA has also looked at failure of water-side isolation valves and inleakage of steam from other modules. For these accidents, the major source for the release is from washoff of FPs from the SG tubes, where they assume 2% of the tubes are wetted and all of that 2% gets off. The normal relief path is unfiltered, but credit is taken for the operator switching to the filtered one (after 30 min). The design includes two helium purification plants and takes credit for the operator's ability to switch to the spare (after 30 min). The loss-of-forced-convection accident relies on an active (vs passive) cavity cooling system, but long-term outages can be postulated (without damage) to allow for repairs on emergency diesels, reconnection of the outside grid, and/or pump repair.

KFA has done a lot of work on code verification. Its LUNA loop was described as a test facility for HTR thermal-fluid flow codes; currently, possibilities of prismatic core tests are being discussed with DOE.

Most of the KFA fuel testing has used uranium oxide (vs uranium carbide in the U.S. design), with their lower enrichment (8% vs 20%) and lower design burnup. KFA staff said that while the carbide fuel has some advantages for fertile loadings (which they don't have) and is less susceptible to the amoeba effect (which they haven't seen at their power densities), the oxide fuel is easier to manufacture to the required quality. In their fuel tests, they have seen significant coating failures and deterioration at 1700-1800°C. They have also seen more significant failure rates at burnups somewhat higher than their design burnup (but less than ours). At 1600°C, the holdup of FPs in the graphite contributes significantly to a reduction in the release. The relatively fast diffusion of silver through the coatings at lower temperatures is of no consequence to the overall risk. The KFA lab for testing fuel is very impressive.

KFA staff have also been working on models for effective retention of releases into the reactor building. They noted that the amount of activity attached to dust is hard to calculate, that reasonable models for the dust/FP transport phenomena don't exist, and that their planned depressurization and dust release experiment at the AVR was cancelled (via the citizen lawsuit). They also claim that there are big uncertainties in their FP washoff and "steam-off" models. They would be interested in collaborating in the DOE COMEDIE loop experiments.

KFA has classified the massive air ingress accident as being of too low a probability to be included in its risk study but is looking at it anyway (Chernobyl!) and have done some very interesting parametric studies. Microchip manufacturers are assisting with the development of a process for coating the pebbles with a thin layer of silicon carbide. In some air furnace tests to date, those without the coating disintegrated while those with the coating looked "undisturbed."

KFA also has a very active program in the metals (including graphite) institute and has an active collaboration program with DOE/ORNL (Phil Rittenhouse, Ray Kennedy, and Tim Burchell). It is notable that the HTR-MODUL design uses only Incoloy 800H for their SG tubes to avoid thermal stress problems at the bimetallic welds.

In our talks at the Institute for Reactor Development, it was noted that sources for their physics data for the water ingress accidents included those from Austrian (10 years old) and Swiss (new) experiments. They quoted a surprisingly low error (5%) estimate for the reactivity vs water ingress models. They referred to a "Chernobyl Syndrome" effect that a professor in the Green Party has brought up. The curve for reactivity vs water in the HTR-MODUL core peaks at about $4.5 with 1000 kg of water, so he has postulated that if one starts out with more water and quickly empties it out, enough reactivity could be inserted to go prompt critical. Reasonable mechanisms for effecting this have not been postulated. I obtained copies of reports on their accident analysis code TINTE (in German).

During our visit to the AVR, most discussions were about AVR operating history and dust data experiments. AVR personnel were surprised at how small the dust particles were (the peak in the size distribution curve is at <1μm). For more information on the dust experiments, we were referred to Prof. VonderDecken (at KFA). The AVR is currently awaiting its decommissioning license, and no further operation is planned.

Several other miscellaneous topics were discussed:

1. A lot of concern was expressed in FRG (and France) about how the 1992 Common Market normalization process would be implemented. Most seemed to think that the problems are too far from solution to be solved by then. In FRG, there is concern that they will not be able to compete with cheap French (nuclear) power, even at home in coal country. Apparently, France's current and near-term planned capacity is great enough to supply a lot of FRG's needs;

2. It was mentioned that the primary concern at KFA about the U.S. MHTGR-NPR program was that FRG public support for their HTR would be seriously eroded if it were shown that a modular HTR was capable of being used as a weapons producer; and

3. The FRG reactor safety committee analogous to the U.S. ACRS completed a study of the HTR-MODUL and concurred with the no-containment-building design. Prof. Nickel, who is on that committee, will forward us the report when it becomes available (January 1990).

<u>THTR-300 (HKG)</u>

The THTR has been shut down since November 1988 and is awaiting a decommissioning license. It is a very sad situation, since THTR obviously had a great potential for a long, productive, "safe" life but operated only about 2 years. Its demise was mainly political. It is located in a primarily "Green" coal-country state, where the local authorities felt that they were mislead about the usefulness of THTR as a potential process heat reactor (which could make use of local coal). [An FRG study showed that the best (only) potential uses of HTR process heat were for aluminum manufacture and refineries.] Also, the THTR fuel supplier was shut down due to a scandal, the estimated time and money needed to build up an alternative was excessive, and apparently neither the state nor the federal government was willing to guarantee the needed financial support.

During THTR operation, some design deficiencies were uncovered, most of which were corrected. A problem in the pebble discharge circuit design prevented refueling at powers (flows) higher than 40%. Several other factors contributed to undesirable in-core pebble flows: a miscalculation of the pebble friction at high temperatures, control rods "shielding" the outer ring of pebbles from flowing to the center, and an improper design of the angled core floor that impeded discharge flow. Also, an arbitrary high-temperature limit set for an auxiliary room limited power output on hot days. After the last shutdown, personnel discovered several (33 of 2600) damaged Incoloy 800 bolt heads on the cover plates in the hot lower plenum ducting. The failures were at the points of highest temperature gradient (differences of 100°C were measured across the ducts). They claimed that while this was not a safety problem, it needed monitoring, and an on-line means of detecting a detached plate was described. (I would guess that repairs would have been possible using remote welding techniques.)

The THTR nuclear and thermal pre-operation predictions were very good. While their initial critical loading predictions were very close, they didn't have any on-line reactivity calculation, probably because of the large uncertainties in fuel distribution (fuel burnup and temperature vs position).

The THTR got good service from its capacitance probe moisture monitors (capacitance), and staff members said that if the moisture level stayed at or below 1 ppm there would be no corrosion problems. The primary leak rate was 1/3 of the inventory per year. The plant had excellent maneuvering capability, and could (by test) sustain a turbine trip without needing to shut down the reactor. The staff conducted a variety of dynamics tests for code validation.

There are several problems with the THTR decommissioning, including uncertainties in the core configuration and fuel loading distributions upon emptying the core. The fixed detector locations may also make reliable monitoring of criticality a problem as well. I made several "helpful" suggestions, which I plan to follow up.

The THTR has a wealth of operating data (on plant computer tapes), some of which may be of interest to the U.S. MHTGR Program. The THTR staff was interested in pursuing the possibility of a subcontract to retrieve and analyze some of the data. In particular, they have data that could be useful in code validation for coolant mixing analyses for the outlet plenum. Steam generator and other component operational data may also be useful.

Summary of Significant Findings and Recommendations

1. The new U.K. policy of licensing reactors on the basis of PRA calculations has gotten them into a bind with the AGR refueling safety case, perhaps unnecessarily. The United States should be very wary of adopting such a policy.

2. The U.K. NII is developing rules for diversity of maintenance personnel to avoid common-mode failure problems with maintenance, which is their major safety concern.

3. The NII policy statement for accident code validation and verification could be useful to the NRC in drawing up a similar guide for licensees.

4. The U.K. problem with the exclusion of nuclear plants from the privitization sale was due primarily to the uncertainties of the (large) costs for decommissioning the plants. The United States should work toward establishing reasonable policies and cost estimates for various decommissioning options. Collaboration with Mr. Giraudel of CEA would be useful (and would be particularly interesting for Fort St. Vrain decommissioning).

5. We should pursue the subcontract with John Askew to retrieve the U.K. physics data of interest to the MHTGR program.

6. NRC should follow the design and planning for the COMEDIE loop experiments. It appears to me that additional collaboration with groups experienced in fission product (FP) transport would be useful to everyone.

7. FRG is still putting a lot of effort into HTRs in spite of its recent setbacks, most notably the THTR shutdown and the lack of a near-term expectation for advanced HTR sales in FRG. The FRG R&D effort does not seem to be too dependent on U.S. work, but they are very interested in U.S. policies, public acceptance, and licensing criteria.

8. FRG considers water and steam ingress accidents to be the major contributors to risk in the HTR-MODUL design, and has modified the fuel design (fissile loading) to mitigate the positive reactivity insertion resulting from water/steam ingress. FRG is continuing an aggressive theoretical and experimental program to resolve the remaining problems. We should look for parallels to their "Chernobyl Syndrome" in the U.S. design.

9. Unlike the completely passive U.S. cavity cooling system design (the air-cooled RCCS), the HTR-MODUL design relies on redundant power supplies and pumps, with margin to allow for equipment repair or replacement periods.

10. FRG tests on its fuel are of much interest to our fuel performance evaluations, but the significant differences between the U.S. and FRG fuel designs preclude "direct" use of their data. The two major items of interest were the marked deterioration of the kernel's protective coating with burnup, and the observation that the FRG (oxide) fuel is easier to manufacture to the required quality than is the U.S. (carbide) fuel.

11. KFA noted that the current models for FP transport via dust and washoff are not reliable. KFA would be interested in collaborating in the COMEDIE loop tests.

12. FRG's development work on a silicon carbide coating for their pebble fuel to mitigate oxidation attack should be looked into both for reducing concerns about air ingress accidents for the U.S. design and for possible use as an additional FP barrier in a fuel stick.

13. The FRG safety committee analogous to the ACRS "approved" the no-containment building design for the HTR-MODUL. The report on this should be of both technical and political interest to NRC deliberations.

14. A wealth of data and experience is tied up in the THTR design and operation that is pertinent to MHTGR concerns, and the THTR staff is interested in pursuing collaborative work. This should certainly be pursued.

APPENDIX 1

Itinerary

November 6-7, 1989	Travel from Oak Ridge, Tennessee, to Liverpool, England
November 8, 1989	Visit Heysham-2 Reactor Site, Heysham, England
November 9, 1989	Meeting with Nuclear Installations Inspectorate, Bootle, England
November 10, 1989	Meeting with AGR Program Personnel, Harwell, England
November 11-12, 1989	Travel to Paris, France, for weekend
November 13, 1989	Meeting with CEA Gas-Cooled Reactor Personnel, Fontenay-aux-Roses, France
November 14, 1989	Meeting with CEN personnel, tour of Siloe reactor and COMEDIE loop, Grenoble, France
November 15-16, 1989	Meetings with HTR Program Personnel at KFA, Julich, Federal Republic of Germany
November 17, 1989	Meeting with THTR-300 Reactor Plant personnel and plant tour, Hamm, Federal Republic of Germany
November 18, 1989	Travel from Frankfurt, Federal Republic of Germany, to Oak Ridge, Tennessee

APPENDIX 2

Persons Contacted

Heysham-II Reactor	
Neil W. Davies	Thermal Reactor Collaboration Manager, UKAEA-Risley
John Birchall	Principal Physicist
Tom White	Assistant Operations Manager
Nuclear Installation Inspector	
Derek Goodison	Branch Chief
Ian Tate	Siting Criteria
Jim Murray	Accident Selection and Containment
Malcolm MacPhail	Source Terms
Bill Whiteley	Accident Mitigation
UKAEA Harwell	
John R. Askew	Director, Gas-Cooled Reactor Programs
John Wilson	Deputy Director, Gas-Cooled Reactor Programs
CEA-Fontenay-aux-Roses	
Daniel Bastien	Coordinator for Gas-Cooled Reactors
Marc Natta	Chief of Service
Bernard Giraudel	Group Leader for GCR Decommissioning
Gerard Chevalier	Department of Mechanics and Thermodynamics

CEN-Grenoble

Jean-Francois Veyrat	Chief of Service
G. Dupont	COMEDIE Loop Manager
Ted Beresovski	U.S. DOE Consultant

KFA-Jülich

Erwin Balthesen	Director, HTR Development
Michael Will	Interatom-GmbH
Helmut Helmers	Chief Engineer, Hannover e.V.TÜV
Wolfgang Kröger	Director, Institute for Nuclear Safety Research
Werner Katscher	Accident Consequences
Dr. Moorman	Fission Product Behavior, Source Terms
Dr. Wolters	Accident Analysis
Dr. W. Rehm	Thermal Hydraulics
Mr. Hennings	Reliability
Heinz Nabielek	Accident Analysis, Fuel Performance
Hubertus Nickel	Director, Institute for Reactor Materials
Mr. Haag	Graphite Behavior
Dr. Breitbach	Materials Testing
Bernd Thiele	Institute for Reactor Materials
Dr. E. Teuchert	Institute for Reactor Development
Dr. W. Scherer	HTR Accident Codes
Dr. Klaus Krüger	Reactor Analysis and Experiments (AVR)
Mr. Pott	Hot Cells-Fuel Heatup Tests

THTR-300 Reactor

Rüdiger Bäumer	Plant Manager
Ivan Kalinowski	Chief Physicist
Norbert Röhl	Chief of Production
Erwin Balthesen	Director, HTR Development (KFA)

APPENDIX 3

Literature Acquired

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4. Handouts for NRC/ORNL 9/11/89 meeting on NII Siting Policy:

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5. "The Inspectorate's Approach to the Assessment of Code Validation Submissions." Sizwell-B Inquiry. NII; March 1983. (8 pages)

6. "Final Report of Boiler Tube Leak on Reactor 1 on 17 March 1987." Hartlepool Station. 14 April 1987. (5 pages)

7. "Compilation of AGR Trips." (6 pages)

8. Spanton, J. H., Brighton, P. W. M., McLaughlin, D. Stevenson, and G. "Inherent Safety in Small and Medium-Sized Power Reactors: Review of Some Currently Proposed Small and Medium-Sized Reactor Designs." UKAEA Health and Safety Studies Committee; May 1988.

9. Hunt, Colin. "PRA: Modular HTR Systems." Reactor Physics Division; AEE Winfrith. GA-MHTGR-350.

10. Stevenson, G. W. (UKAEA Safety & Reliability Directorate), Mannell, S. J. (UKAEA Safety & Reliability Directorate), Hunt, C. (Winfrith Atomic Energy Establishment), Brighton, P. W. M. (UKAEA Safety & Reliability Directorate). "Ultimate Safety in the Interatom HTR-Modul." September 1988. (105 pages)

11. Hunt, Colin (Winfrith Atomic Energy Establishment). "Review of the Safety Case for the Interatom HTR-Module." Reactor Physics Division; AEE Winfrith; August 1988 (3 pages)

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