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

January 24, 1989

file
Trip
Reports

MEMORANDUM FOR: Distribution

FROM: James R. Shea, Director
International Programs, GPA

SUBJECT: COMMISSIONER ROGERS VISIT TO THE FRG AND VIENNA, NOVEMBER 3-12, 1988

The enclosed trip report by Jack Scarborough, Technical Assistant to Commissioner Rogers, summarizes for your information the major facilities visited and discussions held during Commissioner Rogers' visit to the FRG and the IAEA from November 4-12, 1988. It replaces an earlier version which some of you may have received.

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TRIP REPORT
VISIT TO FEDERAL REPUBLIC OF GERMANY AND VIENNA, AUSTRIA
FOR INTERNATIONAL SYMPOSIUM ON REGULATORY PRACTICES
AND SAFETY STANDARDS FOR NUCLEAR POWER PLANTS
NOVEMBER 3-16, 1988

J. C. Scarborough
Office of Commissioner Rogers

1. INTRODUCTION

This report summarizes the main points acquired and impressions gained during a two week visit to the Federal Republic of Germany and Vienna, Austria, concurrent with Commissioner Rogers' participation in an IAEA International Symposium on Regulatory Practices and Safety Standards for Nuclear Power Plants in Munich from 7 to 10 November, 1988. The report is organized in seven parts as follows:

Section	Topic/Facility
1	Introduction and Summary
2	Upper Plenum Test Facility, GKW Mannheim
3	LWR Severe Accident Research, KFZ Karlsruhe
4	Nuclear Plant Materials Research and Development, MPA Stuttgart
5	IAEA International Conference, Munich
6	IAEA Directors Meeting, Vienna
7	FRG Nuclear Waste Disposal, Gorleben and Asse
8	TUV-Hannover

During the International Conference visits were made to the Siemens AG-KWU Group in Erlangen and the Isar 2 Nuclear Power Plant of Kernkraftwerk Isar GmbH. These trips are also reported in Section 5.

In each section management persons with whom discussions were held, the facility description and purpose, key research and development results obtained, future plans, and relevant literature are presented. This report, supplemented by technical descriptions and literature for each facility, captures the main points only and is not intended as a comprehensive report of discussions and the numerous technical programs and experimental facilities we were shown.

The writer acknowledges with appreciation contributions to this report by his colleagues E. Don McPherson of Nuclear Reactor Regulation and Karen Henderson of International Programs. In addition, Hans Schechter of International Programs contributed his notes and comments on the overall report and prepared Section 8 in its entirety.

2. UPPER PLENUM TEST FACILITY, GROSSKRAFTWERKE, MANNHEIM

Upon our delayed arrival at Frankfurt in the late morning on November 3 we drove our rented van to Mannheim and Grosskraftwerke (GKE) where the Upper Plenum Test Facility (UPTF) is located. The visit had been arranged through Dr. Krewer, a senior official of the German Bundesministerium fuer Forschung und Technologie (BMFT), the federal agency responsible for German commercial nuclear development.

A. Key Management Contacts

Dr. Karl H. Krewer, BMFT
Dr. Manfred Banaschik, Manager, Research, GRS
Mr. Franz Baumueller, Director, GKM
Dr. Werner Leibfried, GKM
Dr. Hans-Juergen Spoehrer, GKM

B. Facility Description and Purpose

The Upper Plenum Test Facility (UPTF), the German contribution to the trilateral 2D/3D Project between Japan (JAERI), USA (USNRC), and the Federal Republic of Germany (BMFT), simulates on a 1:1 scale the primary cooling of the Siemens-KWU 1300 MWe PWR. A schematic of the facility is shown in Figure 1. Detailed descriptions of the UPTF are provided by Weiss, Sawitzki, and Winkler in Kerntechnik, Vol. 49 and Weiss and Hertlein at the 14th Water Reactor Safety Research Information (references). Objectives of the test program are summarized in Appendix A.

The large mass flows of saturated steam (up to 750 kg/s at 220 degrees C and about 20 bar) required for each test is supplied by an adjacent fossil-fired power station operated by Grosskraftwerke (GKM) Mannheim. The UPTF process control scheme, test instrumentation, and data recording system employ elaborate state-of-art techniques. Each test requires approximately one month of preparation time followed by extensive post test analytical interpretation and qualification of two- and three-dimensional thermal-hydraulic codes by the principal country sponsor. Of some 30 tests originally planned (18 separate effects and 12 integral), approximately nine remain to be accomplished.

The 21 tests conducted to date have provided time-dependent spatial thermal-hydraulic benchmark data for integrated LOCA models and have permitted confirmation of correlations of various phenomena. The downcomer separate effects tests, for example, confirmed 1:13 scale tests by showing that ECC penetrates the downcomer under LOCA conditions and refills the lower core plenum even while the primary system continues to depressurize. The Richter, and separately, Ohnuki modified Froude number correlations also yield good agreement with downcomer counter-current conditions in the hot leg under a LOCA.

In summary, the UPTF have shown that predictions from the largest subscale tests (1/13 scale based on area by Richter, et al) are quite accurate (+ or - 5 percent), and current ECCS plant systems and design margins in general are conservative.

D. Future UPTF Plans

Upon completion of the remaining nine tests, including the two integral B&W OTSG configuration simulations, the UPTF will be used in support of future confirmatory studies for the FRG Reactor Safety Commission in the following areas:

- Major component failure modes,
- Plant transients and potential severe accident sequences,

- Man-machine interaction studies and potential vulnerabilities,

- Further German risk assessment studies related to containment loading.

E. References

(1) P. Weiss, M. Sawitzki, F. Winkler, "UPTF, A Full-Scale PWR Loss-of-Coolant Accident Experiment Program," Atomkern-Energie Kerntechnik, Vol. 49 (1986).

(2) P.A. Weiss, R.J. Hertlein, "UPTF Test Results - First 3 Separate Effects Tests, Proceedings of 4th WRSRIM, USNRC, October 27-31, 1986.

(3) P.S. Damerelkl, N.E. Ehrich, K.A. Wolfe, "Use of UPTF Data to Evaluate Scaling of Downcomer (ECC Bypass) and Hot Leg Two-Phase Flow Phenomena," Proceedings of 15th WRSRIM, USNRC October 26-29, 1987.

3. LWR SEVERE ACCIDENT RESEARCH, KARLSRUHE NUCLEAR RESEARCH CENTER

The following day (November 4) we drove a short distance to the Karlsruhe Nuclear Research Center (KFZ) for discussions with Dr. Hans H. Hennies, Director and Member of the Board. We were briefed for approximately two and a half hours by Dr. Hennies on KFZ's overall nuclear research program with particular emphasis on their severe accident safety research program.

A. Key Management Contacts

Dr. Hans H. Hennies, Director KFZ
Dr. Hermann Rininsland, Chemical Engineering Department, KFZ
Prof. Kessler, KFZ
Dr. Koerting, KFZ
Mr. Wilhelm, KFZ

B. Facility Description and Purpose

The principal programs at KFZ Karlsruhe are described in a brochure entitled Main Activities, Karlsruhe Nuclear Research Center in the trip report file.

They include fast breeder technology, uranium enrichment (Becker nozzle) process, nuclear fusion, reprocessing and waste management, nuclear waste repository, environment and safety, solid state and materials research, nuclear and particle physics, microtechnology, handling technology, and other research activities. Current emphasis in nuclear fission technology is on:

- Severe LWR accident core melt concrete interactions with the large scale BETA experiments,

- Detailed fuel assembly thermal degradation under severe overpower/undercooling transients with the large scale CORA experiments,

- Time dependent aerosol production, transport, and deposition within containment, and

- Containment aerosol filter development under accident conditions.

C. <u>Key Research Results</u>

The initial condition of a Siemens-KWU PWR severe accident core melt (oxidic and metallic) is shown in Figure 3. The core melt-concrete interaction has been simulated at KFZ by the BETA test facility, shown schematically in Figure 4. A 10 MWe radiofrequency induction coil surrounding the concrete crucible deposits the requisite time-dependent thermal energy in the core thermite melt to simulate fission product decay heat for the 5 to 6 minute test run. Detailed simulations of the core melt and concrete oblation under different initial conditions (temperatures), oxidic and metallic compositions, concretes (silicates and carbonates), and crucible geometries have been performed with off-gases and aerosols carefully monitored.

Detailed three-dimensional analysis of material and energy flows throughout the concrete oblation of each test has been performed with the WECHSL code. WECHSL invokes at least four different models for heat transfer from melt to concrete during different phases of the oblation and melt cooldown to the solidus temperature. In addition to crucible spatial temperature measurements, off-gas analysis provides molar fluxes of H_2, H_2O, CO, and CO_2 (release of CH_4 or other hydrocarbons is negligible). The aerosol production (source term) throughout the BETA experiment is quantitatively measured. (Detailed fuel assembly melting under prototypic overpower and under cooling core conditions is simulated in the separate CORA facility.) Detailed results of BETA and CORA experiments and WECHSL analyses are presented in Appendix B.

For conditions prevailing in current FRG PWRs, WECHSL calculations normalized to BETA experiments indicate that for the core melt-concrete interaction phase, concrete erosion cannot be stopped by long-term sump water ingression into the (dry) reactor cavity and subsequent surface cooling of the interaction region. Under worst case assumptions Hennies indicated that local penetration of the 6 m deep concrete basemat cannot be excluded after some five to six days.

HEPA Filter Development

Related development emphasis has been placed on a high efficiency particulate aerosol (HEPA) filter which would function reliably under severe accident conditions of high containment temperature, pressure, and moisture. A modular HEPA unit consisting of an approximate one square meter filter "pillow" of 2 to 30 micron diameter stainless steel fiber "fleece" has been developed as a highly efficient (DF of 10^4) aerosol filter under the environmental extremes (up to 250 degrees C) of a severe accident, pressure vessel breech, and core-concrete interaction. Performance of these filters has been reported at the 18th through 20th DOE-NRC Nuclear Airborne Waste Management and Air Cleaning Conferences. The new filters have been installed in 4 PWR and 2 BWR German nuclear power plants since June 1988.

Observations of H.H. Hennies

During the course of the discussions Dr. Hennies offered a number of observations concerning LWR nuclear safety initiatives as follows:

- Standardized Advanced LWRs have already been developed and deployed in the FRG - the latest Siemens-KWU "Convoy" PWRs and current BWRs.

These evolutionary, fourth generation ALWRs contain many of the design features of the large APWR and ABWR which have been developed by Westinghouse and General Electric with Japanese industry and utility participation.

- Dr. Hennies believes the newest class of German ALWRs is "safe enough," even though reliance is placed on active engineered safeguards systems for beyond design basis accidents. PSA studies show that core melt probabilities from internally initiated events are below 10^{-6}/reactor year. Development of passively-actuated engineered safeguards systems in his opinion is unnecessary with highly reliable robust active safeguards systems.

- Improvements can be made in the below reactor vessel concrete basemat design to better cool and contain the molten core resulting from a severe accident. The next generation Siemens-KWU LWRs will incorporate improved basemat cooling as a potential passive safeguards feature.

- The current Siemens-KWU Convoy PWR containment can withstand the internal pressure buildup for at least 5 days from adiabatic flashing of the primary coolant inventory and WECHSL-computed gas generation and aerosols from core melt-concrete interactions. The WECHSL-computed gas generation and aerosol source term, based on extensive BETA and CORA experiment benchmarking, is the significant contributor to containment loading. During the 5 day period before containment venting is necessary as shown in Figure 6, more than 90 percent of the iodine, tellurim, and biologically significant rare earth aerosols deposit on the extensive concrete surface area within the containment (aerosol deposition has been confirmed by other extensive tests under representative conditions).

- The deposition results in off-site doses at the plant boundary after containment venting which are within FRG regulations for normal operation. Assurance of containment integrity for up to an approximate 5 day period following a severe accident, based upon the KFZ BETA-WECHSL and

CORA research and development program, is considered by Hennies a major step forward in closure of the German severe accident research program. Hennies noted that the concrete aerosol deposition surface area within containment is much greater in the Siemens-KWU PWR than in typical U.S. PWR designs.

- Dr. Hennies believes that economic incentives in Germany, where economies of scale are well documented from their construction experience, favor continued emphasis on large LWRs since available reactor sites are limited. (The FRG, a nation having roughly the land area of the State of Oregon, has 11 PWR and 7 BWR operating power stations.) Hennies does not see a role for downsized U.S. advanced modular reactors for electricity production; the German modular HTGR on the other hand may have a role in German process industries.

- Though our discussions did not focus on fuel cycle research, Hennies is a proponent of mixed oxide thermal recycle in German LWRs as a 30-40 year "transition regime" for fast breeders and a plutonium economy.

D. Future Plans and Programs

KFZ future plans are generally described in the Karlsruhe brochure cited earlier.

E. References

(1) H. Alsmeyer, "Melt Concrete Interaction During Severe LWR Accidents," Karlsruhe Nuclear Research Center, to be published.

(2) B. Kuczera, "LWR Severe Accident Research Activities and Trends at KFK," undated paper.

(3) V. Ruedinger, C.I. Rickets, J.G. Wilhelm, "The Realization of Commercial High Strength HEPA Filters," KFK.*

(4) H.G. Dillmann, H. Posler, J.G. Wilhelm, "Off-Gas Cleaning Devices for Containment Venting System," KFK.*

(5) C.I. Rickets, V. Ruedinger, J.G. Wilhelm, "The Flow Resistance of HEPA Filters in Supersaturated Airstreams," KFK.*

(*Proceedings of 20th DOE/NRC Nuclear Air Cleaning Conference.)

4. MATERIALS TESTING INSTITUTE (MATERIALPRUEFUNGSANSTALT, MPA), STUTTGART

After tours of research facilities and luncheon at the Karlsruhe Research Center we drove to our appointment with Professor Kussmaul at the Materials Testing Institute, University of Stuttgart the afternoon of November 4. A history of the 100 years of testing, research, and teaching of the MPA Stuttgart is contained in a brochure in the trip file. A member of the German Parliament, Dr. Lauf, was on hand to meet Commissioner Rogers at the beginning of our MPA visit. He stressed the severe negative impact on German nuclear programs of any nuclear plant problems in other countries <u>including</u> the U.S. He also urged the harmonization of international nuclear regulatory codes and practices as a necessary step in gaining international public acceptance of commercial nuclear power.

A. <u>Key Management Contacts</u>

The MPA was separated administratively from the University where it was located in 1980 and assigned to the Ministry of Trade and Transport. Since 1976 Professor Dr. Ing. Karl Kussmall has been director of the Institute and lecturer on the subjects Materials Testing, Materials Science, and Strength of Materials within the University's Mechanical Engineering curriculum with Professor Dr.-Ing. H. Dietmann.

B. <u>Facility Purpose and Description</u>

Present MPA activities comprise general material and component investigations and material development as well as welding techniques, forged nozzle-forged shell "joint" technology, high-pressure testing, Fracture mechanics, corrosion, tribology (wear), failure analysis, reactor materials safety research, plant engineering, and calculation/investigation of large components and systems under operational and accident conditions. Such investigations require state-of-art computational facilities and the MPA facilities include a CRAY 2. MPA assists local and federal authorities in licensing issues and procedures which require national and international cooperation due to their scope. The current MPA staff numbers approximately 360, of which 130 are scientists. Tables 1 and 2 summarize the scope of activities of the MPA Stuttgart.

C. Key Results

We were shown a unique 12 mega Newton High Speed Tensile Testing Machine which was designed and built at MPA for examining very large dynamic load transients for safety-relevant components of nuclear power plants. Its purpose is to determine the influence of very high loading rates on the stress and strain behavior of unwelded and welded ferritic and austenitic materials. The machine is driven by a propellant charge and generates a maximum tensile force of 12 megaNewtons with a poston velocity of 25 m/s after a stroke of 20 mm, or a maximum velocity of 60 m/s after a stroke of 400 mm. A test velocity of about 50 m/s coupled with a gauge length of about 1 m results in a strain rate of up to 50/s, a realistic upper limit for dynamic loading of nuclear power plant components.

The design of the machine, its mode of operation, the sophisticated measurement technique for test specimens, and typical dynamically-tested specimen stress, strain, and elongation data are described in a paper by Kussmaul and Sturm, "A New Generation of High Speed Tensile Testing Machines." In an extensive program of high speed tensile tests of wide plate chrome-nickle alloy (6 CrN 1811) material in which dynamic strain rates were varied from less than 10^{-3}/s to approximately 60/s, no unfavorable material behavior was observed. These data, which are shown in Figure 6, are typical of the dynamic loading encountered in aircraft crashes, certain postulated fast reactor excursions, and reaction forces from explosions. (The machine may have applications for testing new cask design for air shipment of plutonium in 100 gram quantities.)

D. Future Plans

The High Speed Tensile Testing Machine has and continues to be used to assure the integrity of nuclear reactor pressure vessels and main coolant piping systems in the presence of material flows under postulated high energy release rates and/or high dynamic loading conditions (e.g., the effect of an aircraft crash or the reaction forces caused by waterhammer and earthquakes). Since the weld seams of major components are generally considered to be the safety-relevant areas of the structure, it is usually necessary to test welded plates of large cross-section with and without flaws.

E. References

(1) "MPA Stuttgart, New Testing Facilities 1980."

(2) Karl Kussmaul and Dietmar Sturm, "A New Generation of High Speed Tensile Testing Machines," University of Stuttgart.

(3) P. Julisch, H.J. Hadrich, W. Stadmuller, and D. Sturm, "Dynamic Examination of Wide Plate Specimens to Cover Incidents on Components with High Energy Rate," University of Stuttgart.

(4) P. Julisch, H.J. Hadrich, W. Stadtmuller, D. Sturm, "Load Bearing and Deformation Behavior of Dynamically Loaded Wide Plate Specimens," 13th MPA Seminar, October 8 and 9, 1987, University of Stuttgart.

(5) P. Julisch, H.J. Hadrich, D. Sturm, "Dynamic Examination Technique for Large Specimens to Cover Incidents on Components with High Energy Rate," 8th International Conference on "Structural Mechanics in Reactor Technology," Brussels, Belgium, August 19-23, 1985.

(6) D. Sturm, P. Julisch, H.J. Hadrich, Nguyen-Huy, Tan, "Dynamic Testing Techniques for Components and Large Specimens to Cover Incidents with High Energy Rate," International Conference and Exposition on Fatigue, Corrosion Cracking, Fracture Mechanics and Failure Analysis, 2-6 December 1985, Salt Lake City, Utah.

(7) P. Julisch, H.J. Hadrich, W. Stadtmuller, and D. Sturm, "Dynamic Large Scale Testing With a 12 MN-High Speed Propellant Driven Tensile Testing Machine," IAEA Specialists' Meeting on Large Scale Testing, University of Stuttgart, May 25-27, 1988.

5. IAEA INTERNATIONAL CONFERENCE, MUNICH

The International Symposium on Regulatory Practices and Safety Standards for Nuclear Power Plants, held from 7 to 10 November in Munich, involved more than 200 regulatory authorities from approximately 36 IAEA Member States and three international organizations. The program of the symposium is in the trip report file along with most of the papers presented. Commissioner Rogers participated as an observer on 7 and 9 November and chaired a panel discussion on Implementation of Developments in Safety Standards and Practices on 9 November. Side trips to Siemens-KWU headquarters engineering and research facilities in Erlangen were made on 8 November, and to the Isar Nuclear Power Plant on 10 November. These trips are reported in this section under separate captions.

The International Symposium was preceded by a formal dinner for participating senior government representatives at the home of Dr. A. Birkhofer, Chairman of the FRG Reactor Safety Commission (GRS), which Commissioner Rogers and Mrs. Rogers attended.

International Symposium

Mr. K. Toepfer, the Minister for Environment, Nature Conservation and Reactor Safety in the FRG, was the keynote speaker the opening day and made the following points in his address:

- Every effort must be made by Member States to prevent a severe reactor accident if commercial nuclear power is to gain public acceptance. Should a severe accident, however, occur, every effort must be made to limit environmental effects.

- International harmonization of regulatory standards and practices need not be on a "least common denomimator" basis. Revised IAEA NUSS codes should be accepted voluntarily and recognized as binding safety standards by Member States.

- INSAG Safety Principles have been used in the FRG to improve communication between the public and federal regulatory agency (BMW).

- Engineered safeguards systems in operating reactors should be examined and improved where necessary; for new advanced reactor designs, NUSS codes should be invoked.

- Member States should encourage development of a "safety culture" among plant operators.

- OSART international plant inspections have proven useful; while the concept could be extended to include review of the regulatory function of Member States by "expert" international teams, internationalization of nuclear energy is not necessary, desirable, or practical. Situations in which international regulatory oversight is misinterpreted and lead to a lack of individual national responsibility must be avoided.

- A new NUSS code should be considered to specify comprehensive plant safety assessments such as PSAs. Basic NUSS codes and IAEA safety principles should be accepted by countries.

- Further research should be expended on risk management, including human error, measures of accident control, containment inerting and venting, etc.

- Human error is inevitable and plants should be designed to allow for such degree of human error.

- Additional emphasis should be placed on training of nuclear plant personnel. Safety resides with individual operators; Member States must ensure that operators perceive their responsibilities within respective national regulatory frameworks.

- While nuclear safety principles are well known and international experience contributes to these, nuclear safet is a dynamic process, and safety practices should be advanced within the framework of international cooperation and partnership. Since current nuclear standards and requirements cannot be fully met by older nuclear plants, case by case determinations based on national circumstances must be made by individual Member States.

Hans Blix, Director General, IAEA, then spoke briefly and endorsed many of the suggestions in Minister Toepfer's address. The individual papers in each of the five sessions of the international symposium and the discussions of the various panels are not reported here since copies of the papers are in the trip report file. In addition a summary of the conference has been prepared by Dr. G. D. McPherson of the NRR Director's Office. Impressions from the symposium include the following:

- There is substantial but not unanimous opinion within Member States that greater "rationalization" (commonality of international safety practices and standards) is necessary or at least desirable. The current lack of uniform international regulatory criteria was highlighted by an IAEA survey of nuclear regulatory practices in Member States.

- The peer process by which the IAEA Staff conducts OSART plant inspections shown in Figure 7 is of interest and seemingly could be adapted as well to other nuclear facilities such as Member State waste repositories.

- The concept of international peer review of a Member State's nuclear facility or regulatory agency may increase the confidence of adjacent or nearby Member States in the safety of the plant and competence of the particular agency to perform its function. National regulatory structures are somewhat inflexible. Member State regulatory structures mesh with their particular form of government, and a single, representative norm does not exist. It may be useful, however, to examine regulatory organization divergence in international forums.

- An IAEA NUSS standard addressing the role of PSAs as a regulatory tool is a likely possibility.

- Increasing attention is being given to the prevention and mitigation of severe accidents in operating power plants, including severe accident management.

- The importance of human reliability and role of human factors in PSAs is increasingly being recognized.

- Greater commonality of safety goals and their reduction to practice in design, construction, and operation of plans is necessary.

The NRC papers by Murley et al, Beckjord et al, and Houston were well received by the participants.

Siemens AG-KWU Group Visit

On November 8 we were driven by Siemens AG-KWU Group to Siemens' Headquarters in Erlangen for discussions on the evaluation of the Siemens' Standardized Convoy 1300 MWe PWR, the Interatom "HTR-Module," and the Siemens-Bechtel alliance to provide full-spectrum nuclear plant services to U.S. utilities. After lunch we toured a number of Siemens nuclear research and development laboratories at Erlangen before returning in the evening to Munich.

A. Key Management Contacts

Dr.-Ing. Wolfgang Keller, Senior Vice President, Siemens AG-KWU Group
Dipl.-Ing. Adolf J. Huettl, General Director, Siemens AG-KWU Group
Dr. Wolf Gruner, Senior Director, Power Plant Concepts, Siemens AG-KWU
Dipl.-Ing. I.A. Weisbrodt, Division General Manager, Advanced Reactors, Interatom
Dr.-Ing. Otto Gremm, Senior Director, Main Department Plant and Systems, Safety Concepts, Siemens AG-KWU Group
Dr.-Ing. Leonard Slegers, Siemens AG-KWU Group
Dipl.-Ing. Ernst Liss, Deputy Director, Department Manager, KWU-Bechtel Services, USA, Siemens AG-KWU Group
Prof. Dr.-Ing. Erich Tenckhoff, Head of Subdivision, Materials and Chemistry, Siemens AG-KWU Group

B. Facility Purpose and Description

In October 1987 Kraftwerk Union AG became the seventh group within Siemens together with the Transformer Union and parts of the Siemens Power Engineering and Electrical Installations and Automation Systems Groups. The merger and incorporation into the Siemens organization has consolidated the company's position as a supplier of power plants and energy systems on the world market. Siemens AG annual sales are about $30.2 billion as compared with General

Electric's $35 billion. The new KWU Group employs approximately 25,000 at nine FRG locations and contributes 12 percent to Siemens' revenues as described in The KWU Group brochure in the files.

Siemens-KWU has designed and constructed in Germany 25 PWRs (<25,000 MWe), 11 BWRs (8,500 MWe), 3 PHWRs (1,170 MWe), and the Kalkar 300 FBR (327 MWe). KWU has been modifying its 1300 MWe PWR for Japanese conditions for the past 7 to 8 years in collaboration with Toshiba and Hitachi for a joint Siemens-Toshiba-Hitachi commercial offering to TEPCO. The PRC is expected to contract for two additional PWR units, and Siemens-KWU continued collaboration with the Chinese in the detailed design of the two Quinshan 600 MWe units in 1986 provides KWU an advantage in the international bidding.

A discussion on the evolution of the Siemens Convoy 1300 MWe PWR, the modular HTGR, and the Siemens-Bechtel alliance to serve the U.S. utility market is presented in Appendix C.

After luncheon and discussions with Dr. Wolfgang Keller, Senior Vice President, Siemens-KWU Group, a tour was made of various Siemens-KWU materials and chemistry labs. Brochures on Siemens capabilities in these areas are in the trip report file. The Siemens facilities in Erlangen, a company town, are extensive, appear to be of good to excellent quality, and are highly integrated. Siemens-KWU nuclear design and technology as embodied in current technical offerings (PWRs, BWRs, MHTR, FRRs, PHWRs, and district heating power reactors) is distinctly impressive.

Isar Unit 2 Visit

On November 10 we were driven by Kernkraftwerk Isar GmbH, owner and operator of the Isar Power Plant, to the station on the Isar River about 14 km downstream from Landshut in Lower Bavaria. Dr. Maurice Katz, Counselor for Nuclear Technology, U.S. Mission to UN organizations (Vienna) and participant in the IAEA conference, accompanied us on this tour. The plant site is ideal for integration of the plant's output into the Bavarian 380 kv grid. A description of the plant and isometric drawing of the Siemens-KWU 1300 MWe Convoy PWR are in the trip report files.

We were given a tour of the station, including containment entry, with the plant at power (1400 MWe) by Mr. Maximilian Rank. The control room was orderly, functionally well laid out, and business like. Only four annunciators were lit (blinking), though three others (of the few hundred) were tagged out. Siemens micro process instruments are employed in the main central board in a semi-circular arc permitting process flow-diagram visualization of the nuclear, turbine generator, and conventional portions of the plant. At each major component node the "process" conditions (temperature, pressure, flow rate, chemistry, radioactivity, etc.) can be directly checked by the analogue or digital micro-instrument.

Isar is the first application of Siemens-KWU's advanced PRINS (Process Information System, consisting of all information equipment in the control room) and of PRISCA, the Computer Aided part of PRINS, an enhancement of Siemen's already powerful plant process computers. PRISCA provides defined sets of visually interpretable sets of displays, shown simultaneously on banks of CRTs, which provide a comprehensive overview of reactor behavior. A unique feature of the new system is the capability for the operator's own selection of multi-colored picture display, fast access to related information, and a large amount of available detail. At Isar one bank of eight, two banks of six, and several single screens are provided. (The PRINS/PRISCA process information system is described in "Testing Time for Siemen's/KWU PRINS/PRISCA," Nuclear Engineering International, November 1988.)

Several facts obtained during the Isar visit included:

- A significant amount of plant operational data for major nuclear systems is being transmitted to licensing authorities in the State of Bavaria. The operational data are listed in Figure 14.

- The operator shift hours and shift rotation for the Isar 2 unit, shown in Figure 15.

- The lack of an on-site training simulator and reliance by the Station Superintendent on demonstrated comprehension of plant equipment and systems through progressively more responsible shift positions by operating crews. Operators do receive initial simulator training at the Siemens-KWU facility in Erlangen, but not annual refresher training on a plant specific simulator as in the U.S.

- Initial operator reluctance to accept PRINS/PRISCA due to lack of shift training and shortage of documentation about it in plant operations manual, but growing acceptance among shift crews as PRISCA's capabilities were being demonstrated.

- A large $3 million detailed 1:25 scale model of the Isar 2 plant constructed in moveable sections is routinely used for operational training and preparation for maintenance purposes. The model replicates all components, systems, and structures and plant small bore piping down to about 1 in O.D., including cable trays, ventilation ducts, lifting beams, and anchor plates.

After luncheon at the plant with the Station Manager, Hans Beuerle, we were driven to the Munich airport.

C. References

(1) Isar 2 Isometric Drawing with Plant Characteristics.

(2) W. Gruener, "The Convoy-Projects: A Case Study from Germany," Technical Session 13.14, ANS Winter Meeting, Los Angeles, 1987.

(3) Kernkraftwerk Isar - a monograph on the Isar 2 plant.

6. IAEA HEADQUARTERS VISIT

A. Key Management Contacts

Director General Hans Blix
Deputy Director General Jon Jennekens
Deputy Director General William Dircks
Deputy Director General Leonid Kostantinov
Deputy Director General Noramly bin Muslim
Associate Deputy Director Morris Rosen

B. Visit and Discussions Held

The U.S. Mission in Vienna had arranged interviews with the above senior Agency management, and Mr. Ted Sherr (formerly of NRC-NMSS) served as our Vienna contact and host. The Agency discussions were in the morning and afternoon. At a luncheon hosted by Commission Rogers for U.S. IAEA and Mission personnel, Commissioner Rogers summarized current NRC initiatives and responded to numerous questions from attendees.

C. Summary of Main Points Discussed

The main points which emerged from discussion with the IAEA senior management were the following:

- Developing countries are not likely to build commercial reactors due chiefly to lack of capital and trained cadres; the Build Operate-Transfer (BOT) concept appears the most probable means of introduction of commercial nuclear power in such countries.

- OSART inspections of operating reactors have advanced international safety and should be helpful in restoring public confidence in commercial nuclear power.

- The value of Probabilistic Safety Analysis (PSA) techniques is limited chiefly to designers and regulatory bodies; very low frequency events having serious consequences are not generally understood by the lay public.

- Greater acceptance of IAEA NUSS standards by Member States would represent a major step in international safety of operating plants, since they codify international safety experience and are easily interpretable by designers, constructors, and operators. A recent survey, however, revealed that NUSS standards are not universally embraced by most Member States.

- The Agency has been too passive in its consideration of disposing of radioactive wastes and of encouraging development of advanced reactors incorporating safety enhancements.

- The international safeguards regime is strained in the current period of "zero growth" but continued technology diffusion to non-weapon States. Technical and institutional improvements must be adapted to maintain adequate international "timely warning" and deterrence capabilities. In this respect the advent of large commercial reprocessing facilities for LWR fuel at La Hague (UP-2, UP-3, UP-4), Sellafield, and Tokai (JNFS) which are under the IAEA safeguards regime are of less concern than those Member State facilities not under IAEA safeguards, such as Argentina's enrichment and reprocessing facilities, Brazil's conversion and laboratory scale enrichment facilities, and Pakistan's enrichment facility.

- A need exists for better understanding of international design bases for filtered containment venting under severe accident scenarios, and possibly revision of the NUSS standard on functionality requirements of LWR containment structures.

- The possible extension of IAEA peer review teams to include plant safety assessments (e.g., PSAs), Member State regulatory agency effectiveness, and Member State design/operation of radioactive waste facilities.

A detailed summary of individual discussions is presented in Appendix D.

7. FRG RADIATION WASTE REPOSITORY VISITS: GORLEBEN AND ASSE

Commissioner Rogers returned to the U.S. following the IAEA Vienna meetings, and Dr. Schechter and the writer proceeded to Hannover for two one-day trips by car to the FRG radioactive waste repositories at Gorleben and Asse in Lower Saxony on November 14 and 15, respectively. The Gorleben salt dome, with a length of about 14 km, a width of up to 4 km, and a base of 3,000 to 3,500 m deep, was designated in 1977 as the provisional site for a federal nuclear waste repository. Exploration of its suitability is the responsibility of the DBE, the German Company for Construction and Operation of Waste Repositories. DBE is performing the planning, construction, and mining investigations required for the Gorleben project. A description of the surface and underground exploratory investigations of the Gorleben salt dome (and the Konrad iron ore mine intended chiefly for low and intermediate level wastes) is presented in a brochure, "Final Disposal and Related Waste Management," by DBE and Siemens-KWU in the trip report files.

At Asse the domed salt mine is well known, having served as a prototype repository for low and intermediate level wastes since 1967. Asse served as a former commercial potash (KCl) mine from 1898 until the mid 1920s. Research and development at Asse is conducted by the government owned Company for Radiological and Environmental Research (GSF). The Asse dome is approximately 8 km in length, more than 2200 m deep, and of from 300 to 1700 m in width with increasing depth. A description of the Asse mine and its geological formation is presented in a brochure, "The Asse Salt Mine, Research for Final Disposal," by GSF in the report files.

Extensive discussions and tours of the Gorleben and Asse facilities were held on November 14 and 15 and supplementary technical literature was obtained.

A. <u>Key Management Contacts</u>

<u>Gorleben</u> (November 14)

Dr. Ing. Christian Schrimpf, Senior Engineer, Technology and Development, German Company for Construction and Operation of Waste Repositories (DBE).

Dipl.-Kfm. Reinhard Koenig, General Manager, Brennelementlager Gorleben GmbH.

Prof. Dr. Klaus Froehner, Assistant Manager, Brennelementlager Gorleben GmbH.

<u>Asse</u> (November 15)

Prof. Dr.-Ing. Klaus Kuehn, Director of the Institute for Underground Disposal, Company for Radiological and Environmental Research (GSF).

Dr.-Ing. P. Hild, Manager, Repository Safety, GSF.

B. Facility Purpose and Description

A brief summary of the facility purpose and current research program is presented in Appendix E. More detailed technical information is contained in the references.

C. Key Research Results

Gorleben

Current research demonstrations at Gorleben are focused on dry intermediate storage of fuel elements in CASTOR and TN 1300 spent fuel transport and storage casks in the engineered storage hall with natural air convection.

The main objective of these demonstration programs was the creation of a data base for large scale applications including:

- verification of cask-specific design data, e.g., shielding and heat dissipation;
- expanding fuel element behavior data base for dry storage;
- testing and optimizing cask handling;
- functional tests on casks, components and special loading equipment and training and handling of a new generation of transport and storage casks.

Results to date can be summarized as follows:

- Cask design parameters have been verified in practice.
- In-pool loading and unloading of transport- and storage casks have been successfully demonstrated.
- Radiation levels to operators are extremely low.
- No rods have failed due to dry storage even for a wide temperature range.

Asse

A GSF publication, The Asse Salt Mine, Research for Final Disposal, states that for the DOE Sr-Cs vitrified HLW emplacement demonstration program:

"The first test disposal of high level radioactive sources in the Asse salt mine is planned for 1989. It is intended to fill the lower third of six 15 m deep boreholes with glass-filled canisters with a diameter of 300 mm and a height of 1200 mm. the glass canisters will contain Sr-90 and Cs-137 as radioactive emitters. There will be two additional boreholes equipped only with electrical heaters, allowing comparative measurements to be made to determine radiation effects other than heat in the boreholes. To ensure that the waste simulates can be retrieved at any time, the boreholes have to be encased, but this is basically not different from the conditions in a future repository for high-level radioactive waste. The test will continue for five years and will allow an extensive measurement program to be carried out over a comparatively long period."

In the electrically-heated and Co-60 simulation tests which have been completed, it was found that "the plastic properties of the salt which are important for the tight enclosure of the wastes are activated in a particular way by rock pressure and heat. It also appears that the water present in the rock salt horizons (in quantities less than 0.05 weight percent) is liberated in the immediate vicinity of the emplacement boreholes and migrates into the borehole. This means that under normal disposal circumstances there is insufficient moisture to allow corrosive attack on the waste canisters."

In Co-60 simulated HLW canisters with an overall output of 36,000 Ci (1.3 X 10^{13}Bq), no previously unknown radiation effects on the host rock salt were noted. (The writer observed that discoloration of the salt had occurred up to a distance of about 1 m from the canister. The radiation absorbed by the borehole wall was about 30,000 Rad/hr.)

D. Future Plans and Programs

Gorleben

The facilities future activities will include:

- conditioning (disassembly/cutting) and encapsulation of spent fuel in final disposal casks;
- encapsulation of radioactive waste in a form suitable for final storage;
- unloading of radioactive waste containers, e.g., vitrified HAWC from transport casks into storage casks suitable for interim and/or final storage;
- maintenance of transport and storage casks.

Due to the pilot function of the facility the throughput is limited to 35 MTHM/year, which is sufficient for the execution of the development programs.

An important demonstration at Gorleben will include permanent storage of LWR fuel elements for which reprocessing is either technically not feasible or economically not justifiable. A reference concept for licensing purposes of permanent geologic disposition of spent fuel is based on development work conducted in the project PAE "Project for Alternative Disposal Techniques" (an R&D program of the German Ministry of Research and Technology, BMFT) during 1980-84 by Nukem/DWK [and continued by DWK on their own in the ensuing years].

The Pollux containment concept (suitable for long term intermediate storage and final disposal of spent fuel) is based on a double shell concept, defined by the final disposal cask and the shielding overpack. (The final disposal cask consists of a gas-tight welded steel containment and assures: safe containment of spent fuel, protection against mechanical loads and impacts and protection against corrosion, which uses a Hasteloy C4 resurfacing welding process. The shielding overpack assures, in addition to the radiation shielding function, further mechanical protection.)

The Pollux cask system has the flexibility and potential to be suitable either for borehole or for tunnel storage in the repository, depending on the fuel mass in the cask and the use of the shielding overpack as additional shielding.

Taking the Pollux cask system as an example, the process in the pilot plant is characterized by the following steps:

- arrival of transport cask at pilot plant;
- dry unloading of spent fuel into hot cell;
- buffer storage of spent fuel;
- dry fuel rod consolidation and loading of fuel rods into containers;
- buffer storage of containers;
- loading of containers into Pollux cask system (final disposal cask and shielding overpack);
- sealing of the final disposal cask by a primary lid;
- transport of Pollux cask system out of hot cell to welding gantry;
- closing steel containment by narrow gap welding of secondary lid in direct operation mode at welding gantry;
- closing anti-corrosive coating at the welding;
- closing shielding overpack by screwing in shielding lid;
- transport to intermediate storage facility.

The process building is about 59 m long, 50 m wide, and 21 m high; its total 80,000 m^3 includes a hot cell volume of 2000 m^3.

The design work and the licensing procedure is progressing so that granting of a first construction license is expected at the beginning of 1989.

<u>Asse</u>

Disposition of intermediate and low level wastes (ILW-LLW) by direct injection and in-situ solidification with grout is an on-going development program at Asse. According to the Asse Salt Mine brochure, the main features of in-situ solidification are:

- direct emplacement of pre-treated granulated LLW-ILW from surface proportioning, mixing, and conveying plants into an underground cavern, using a vertical pipeline for conveying,

- in-situ solidification of the product in the cavern to form a quasimonolithic block.

The following steps are carried out above ground:

1. Granulation according to specifications of the untreated LLW-ILW (slurries, powders, etc.) at the site or origin. The granulate has a grain size range of 0.3 to 5 mm and must be allowed to harden for 7 to 10 days during which time the heat of hydration disperses.

2. Transport of the granulate from the waste producer to the cavern site in special containers with loading and unloading devices.

3. Unloading of the granulate at the above ground mixing plant where a liquid mixture of equal proportions of granulate and cement paste is produced. Besides water, tritiated waste water with approximately 5.5×10^{12} Bq/m^3 (150×10^3 Ci/m^3) can be used for making the cement mixture.

4. Underground operations consist of:

   Vertical transport of the mixture from the surface through a fairly narrow pipe (50 - 60 mm diameter) into a salt cavern (1000 m below the surface) with a volume of 75,000 m^3. Supply is purely by gravity which, with a flow rate of appr. 0.5 m/s, balances the frictional resistance in the pipe.

5. The mixture spreads out in layers on the cavern floor and solidifies in-situ, i.e., at the disposal site. Each layer corresponds to an emplacement phase.

Schematics illustrating the techniques involved are shown in the brochure.

E. <u>References</u>

<u>Gorleben</u>

(1) K. Einfeld, "Pilot Plant for Spent Fuel Conditioning at Gorleben," <u>Nuclear Europe</u>, Vol. 3, <u>4</u>, 1987 (p. 21).

(2) "AFR Interim Fuel Storage Facility, Gorleben," DWK-TWL Status Report, September 30, 1988.

(3) "The Gorleben Interim Store," BLG, October 1986 (brochure with photographs).

(4) Klaus-Detlef Closs, "German R&D on Direct Disposal of Spent Fuel," Nuclear Europe, August-September 1988.

(5) DBE (brochure with photographs and schematics), undated.

(6) CASTOR - Transport and Storage; TN 1300 - Transport and Storage, DWK, (brochures with photographs and details), undated.

Asse

(1) K. Kuehn, et al, "Status of the Nuclear Waste Disposal Program in the FRG," R. Ollig et al, "Status - High-Level Waste Programs, undated.

(2) K. Kuehn, T. Rothfuchs, "High-Level Radioactive Waste Test Disposal, Asse Salt Mine, FRG, GSF, Repository Design and In-Situ Testing, (undated).

(3) K. Kuehn, "Are we ready to construct and operate an underground repository?" IAEA-SM-289/51, International Symposium on the Siting, Design, and Construction of Underground Repositories for Radioactive Wastes, Hannover, 3-7 March 1988.

(4) GSF Report Summary, 1986.

8. TUV-Hannover

The evening of November 14 Mr. Scarborough and Dr. Schechter met in Hannover with two officials of TUV-Hannover, Dipl-Ing Mazur, whom we had met in Munich, and his deputy, Dipl.-Ing. Helmers. Mr. Mazur is head the Division of Nuclear Technology and Radiation Protection, and Mr. Helmers is head of the Project Management Department under Mr. Mazur, with responsibility for supervising several nuclear power plants and nuclear fuel cycle and waste disposal facilities, including the Gorbelen and Asse mines visited by us. The purpose of the meeting was to learn more about the responsibilities and activities of the TUV.

Like all German TUVs, the Hannover TUV (founded around 100 years ago) is an independent, self administered, non-profit organization of experts involved in performing independent testing, inspection services and efficiency investigations for various clients -- private individuals, manufacturers and state and federal governmental authorities -- in the FRG and other countries. TUVs are governed by a board of directors, which provide overall policy guidance. Like many other German TUVs, TUV-Hannover's non nuclear divisions includes: Steam and Pressurized Plants; Environmental Protection and Materials Technology; Electrical Engineering, Mining and Conveyor Technology; Automotive Engineering (vehicle testing); and a Medical-Psychological Institute.

Nuclear activities encompass Radiation Protection and Reprocessing/Disposal Techniques; Stress Analysis and Fracture Mechanics; and Handling Devices and Mechanical Components. All of these disciplines are called upon in a matrix project management approach for reviewing the safety of nuclear facilities. For instance, Mr. Helmers' Project Management Department was called upon to investigate an accident which had occurred last year at Gorleben during shaft sinking work, and which resulted in a casualty caused by the collapse of a shaft wall reinforcement ring. TUV ordered all further shaft drilling work stopped until approved safety procedures have been developed by the facility operator.

Mr. Mazur was the official responsible for inspecting and approving the use , at Asse, of the palladium - titanium coated stainless steel cylindrical borehole liners which will protect the emplaced waste canisters (see Appendix E).

The TUVs are viewed without exception as scrupulously honest and technically extremely competent organizations, and their recommendations, while not legally binding, are hardly ever appealed or challenged by anybody.

APPENDIX A

UPPER PLENUM TEST FACILITY: TEST OBJECTIVES AND KEY RESEARCH RESULTS

The objective of the test program now nearing completion is the full-scale investigation of the three-dimensional single and two-phase flow behavior in the primary system of a PWR during the end of the blowdown, refill and reflood phases of a loss-of-coolant accident. Separate effect and integral tests have been performed to examine phenomena in the upper plenum, upper core tie plate, in the vessel downcomer region, and in the hot and cold legs of the primary system. The experimental program has included simulation of various Emergency Core Cooling System (ECCS) injection schemes common to the FRG, U.S., and Japan, such as cold leg injection, hot leg injection, simultaneous hot and cold leg injection and downcomer injection. Emphasis in all tests has been given to:

- Entrainment/de-entrainment processes in the upper plenum,
- Concurrent and countercurrent flow and core bypass in the downcomer region,
- Condensation and fluid mixing effects caused by cold water injection,
- Concurrent and countercurrent two-phase flow phenomena including water "breakthrough" in the upper core tie plate (fuel assembly nozzle) region.

Experiment simulation areas (for which mathematical models permit exact reactor-like initial and boundary conditions, thus precluding the need for exact physical modeling) include:

- The reactor core, replaced by a core simulator;
- The steam generators, replaced in the region above the SG tubesheet by SG simulators;
- The coolant pumps, replaced by pump simulators; and
- The reactor containment structure, replaced by a containment simulator.

Additional details of the UPTF main components and various simulators are presented in the Kerntechnik reference in the trip report files.

The 34 LOCA tests may be categorized as follows:

Separate Effects	No. of Tests
Counter-current flow conditions at the (core) tie plate	5
Counter-current flow conditions in the downcomer	5
Flow conditions in upper plenum and downcomer when using vent values	3
Small break LOCA items: downcomer mixing and counter-current flow conditions in the hot leg	2
Special phenomena: lower flow path and loop oscillation behavior	5

Integral Tests	No. of Tests
German ECCS Injection	7
U.S./Japanese ECCS Injection	5
Brown Boveri-Babcock & Wilcox OTSG PWR Simulation	2

In addition to the separate effects comparisons, the PWR integral tests of large and small break LOCAs have revealed that reflux condensation occurs without holdup due to hot leg counter flow as predicted from smaller scale data. The UPTF reflux calculation results when applied to U.S. PWRs (the 3800 MWt Westinghouse and Combustion Engineering plants at 1160 psia) show that current regulatory and design margins are very conservative as shown in Figure 2; that is, there is significant margin between actual flows and the counter-current boundary as expected, and during large break LOCAs reflood runback is likely for water deentrained in the legs.

Other possible longer term applications in the author's opinion could possibly include VVER-1000 LOCA simulations under the recent FRG-USSR technical exchange agreement and the Westinghouse AP-600 LOCA simulation with "passive" full pressure ECCS injection under the present 2D/3D trilateral program. The costs of modifying the UPTF for possible VVER-1000 or AP-600 simulations, however, are unknown and thus their practicality is very uncertain.

APPENDIX B

RESULTS OF BETA AND CORA EXPERIMENTS
AND WECHSL ANALYSES OF CORE MELT-CONCRETE INTERACTIONS

The CORA facility electrically heats full length LWR fuel bundles to melting and failure, providing valuable data on bundle melting (individual fuel pin "candling"), melt relocation, and fuel rod fragmentation due to rapid water reflooding. One interesting finding is that Inconel grid spacers initiate Zircaloy liquefaction at temperatures well below the liquidus temperature of Zircaloy. CORA is providing data for validation of computer codes to be used in assessing the effect of accident management strategies involving injection of water into a damaged core, including the influence of quenching on clad oxidation and hydrogen production processes in damaged bundles.

Results from the BETA experiments include:

- Greater downward heat transfer (and oblation) into the concrete than predicted by earlier modeling (high heat transfer results in rapid temperature reduction of the core melt),

- Dispersion of the lower metallic melt into the upper oxidic melt under high gas fluxes,

- Low aerosol releases with the exception of an initial aerosol peak during and immediately after pouring of the melt,

- Fast preferential oxidation of chromium followed by iron oxidation reducing the freezing temperature of the molten concrete to about 1200 degrees C after some interaction time.

The most important conclusions from the BETA and CORA experiments and WECHSL analyses are:

- The slow pressure buildup would not fail the large PWR containment before 4 to 5 days, and overpressurization could be prevented by controlled venting. (Hydrogen production may require ignition or recombination to prevent deflagration.)

- Release of fission products into the containment after initial core melt-concrete interaction is very small (0.1

g/s); filtered venting of the containment would retain these aerosols to an acceptable level.

- Basemat penetration by the core melt to a depth of approximately 6 meters as shown in Figure 5 must be expected, and cooling at the underside of the basemat, either by engineered heat sinks (e.g., a sump water annulus extending further into the base concrete), may not necessarily stop the sideward progression of the melt into the basemat.

- Development of improved containment concepts with passive accident survivability offer a design challenge for the future.

APPENDIX C

SIEMENS-KWU GROUP COMMERCIAL NUCLEAR PLANT DEVELOPMENT

Siemens-KWU Standard Convoy PWR

Figure 8 depicts the chronological evolution of Siemens Light Water Reactors through Prototype, Demonstration, Consolidation, and Standardization phases. Schematic representations of both BWRs and PWRs in the various phases is depicted in Figure 9. The Siemens-KWU evaluated contribution to core melt for the 1300 MWe Standard PWR is shown in Figure 10 to range from approximately 0.5 to 4×10^{-7} per reactor year (internally initiated events). The stainless steel fiber fleece HEPA filters previously described retain 99.99 percent of aerosols and 99 percent of the radioactive iodine.

The PWR containment venting scheme is depicted in Figure 11 showing the primary containment rupture disk, the flanged pipe fitting that must be installed after the severe accident (within 3 to 5 days), the filter in the reactor auxiliary building, and the vent stack. The KWU PWR fiber fleece filter operates under practically dry conditions up to pressures of 6 to 8 bar and gas flow rates of 3 to 12 kg/s. For BWR plants with small containment volumes, KWU developed a combination scrubber and metal fiber filter unit shown in Figure 12. The BWR filtered venting system consists of a wet scrubber with venturi nozzles followed by a combined droplet separator and stainless steel fiber deepbed housed in a pressure vessel as shown in Figure 13. Operational details are presented in "Off-Gas Cleaning Devices for Containment Venting System" by J.G. Wilhelm et al at the 20th DOE/NRC Nuclear Air Cleaning Conference.

Dr. Gruener explained the evolution of the KWU PWR and BWR which has led to a PWR availability factor of 84.4 percent (78 to 92 percent range) and construction times of approximately 5 years (Neckar 2, Isar 2, Emsland). The 1300 MWe Standard PWR construction costs of about 3.5 billion DM, or $2 billion (per unit), yield a specific cost of about 1 [illegible] $/KWe. With German interest costs of 350 $/KWe (60 month co[illegible]ruction period), the total specific cost is about 1900 $/KWe.

Interatom HTR Modul

Isador Weisbrodt, Division General Manager - Advanced Reactors, of the Siemens-KWU subsidiary Interatom GmbH of Bergisch Gladbach, then discussed Interatom's development of its "HTR

Modul" or modular HTGR. The design is similar in its configuration to the General Atomic modular HTGR but is entirely above grade and uses spherical 5 cm diameter graphitic fuel spheres with on-load refueling instead of prismatic fuel and batch refueling off-load.

A brief description of salient safety features of the German plant was presented including data showing that post accident fuel temperatures under de-pressurized (LOCA) conditions would not exceed 1600 degrees C (about 300 degrees C below the onset of SiC fuel particle coating failures. Weisbrodt said that within 3 years Interatom would have sufficient fuel irradiation data to qualify current FRG production HTR fuel statistically to an 1800 degree C fuel limit.

The potential HTR Modul maximum graphite oxidation rate under "perfect chimney" assumptions (top and bottom breech of the 1.5 m reinforced concrete cylindrical aircraft crash barrier structure, two similar holes in the inner confinement structive, and a lower reactor pressure vessel breech with the pressure vessel head removed thus permitting an optimum thermal head for air ingress, core convection cooling and graphite oxidation) is 0.6 kg/h. This value is an acceptable rate for the HTR Modul core support design.

The HTR Module relies, however, on <u>active</u> components (valves, pumps, actuators) for residual heat removal by the <u>water-cooled</u> Reactor Cavity Cooling Panel which surrounds the reactor pressure vessel. Weisbrodt explained that passive air cooling of the reactor cavity as provided in the General Atomic Modular HTGR would not be licensable in the FRG due to concerns about the possible entry of combustible/ ignitible vapors from an external source, a design basis accident requirement for all German reactors.

In 1987 a Safety Study for the HTR Modul (2 unit plant) was submitted to FRG licensing authorities in Lower Saxony. Completion of the review is scheduled as follows: TUV Hannover, September 1988; BMU, February 1989; GRS, March 1989. Mr. Weisbrodt reiterated KWU-Interatom's standing offer to exchange the FRG Safety Analysis Report and Regulatory Authority's Evaluation Report for the DOE-sponsored General Atomic SAR and the NRC's SER, while acknowledging the present DOE restrictions on release of Applied Technology information. [Note: NRC should transmit the <u>nonproprietary</u> version of the GA Safety Analysis Report and the agency's SER upon Commission action through appropriate channels to the Germans and ask for the equivalent German documentation. It is the German BMU safety evaluation which would probably be of greatest interest to the NRC and which is not publicly available in Germany.]

Mr. Huettl, General Director of the KWU Group, summarized the current state of the Siemens-KWU and Asea-Brown Boveri (ABB) joint offer to the USSR to provide HTR equipment, design, and supporting technical services for dual purpose (electricity and process heat) for Soviet industrial applications. The approximate 500 million DM deal had been successfully arranged by Chancellor Kohl with Secretary Gorbachev in September 1988, but the detailed scope of services and commercial arrangements remains to be negotiated. Nevertheless, the agreement between the two Heads of State is notable in that it marks the first commercial export of a modular advanced reactor to the Soviet Union through a German-Swedish joint venture.

Mr. Huettl also stated that the Siemens-KWU commercial agreement with General Atomic had expired in October 1988 and would probably not be renewed in light of U.S. information restrictions related to DOE MHTGR New Production Reactor development. Messrs. Weisbrodt and Huettl stated that in Siemens-KWU opinion a conventional containment structure for HTGR plants was not required with current coated fuel particle technology, citing the THTR licensing with confinement in lieu of containment. In response to a question Mr. Weissbrodt opined that the addition of a conventional containment to the GA MHTGR would probably range from 5 to 7 percent of the plant capital cost. The Siemens-KWU management recognizes that the MHTGR-NPR development would contribute approximately $500 million in first of a kind costs to U.S. MHTGR commercial development which otherwise would be unavailable under present U.S. circumstances.

Siemens-Bechtel U.S. Alliance

Finally, Mr. Ernst Liss, Deputy Director of the KWU/Bechtel joint venture in the U.S., described the services to U.S. utilities which Siemens-KWU and Bechtel are offering. These include major equipment servicing, repair, and replacement; inspection and test services; decontamination and chemical cleaning; technical support services; radioactive waste management and processing; and outage planning, refueling services, and outage management services. Recent contracts had been received from Commonwealth Edison (Zion Station) for outage planning and management and Cleveland Electric (Perry Station) for refueling services. The work is being done by an 80 person KWU-Bechtel organization in Bechtel's Gaithersburg offices. KWU-Siemens offers comprehensive LWR technical experience in both the nuclear and conventional islands to Bechtel's design-construction capabilities.

APPENDIX D

DETAILED COMMENTS/DISCUSSIONS OF IAEA SENIOR MANAGEMENT
WITH COMMISSIONER K.C. ROGERS
IAEA HEADQUARTERS
NOVEMBER 11, 1988

DDG Jon Jennekens

The main points made by Deputy Director Jennekens to Commissioner Rogers were the following:

(1) U.S. recognition of limited indigenous fuel supply in Japan and the FRG, and need to utilize plutonium. The U.S.-Japan agreement recognizes this fact.

(2) There are very real limits on special nuclear material (SNM) accountability techniques. Redundant, diverse sources of information on a would-be proliferator's SNM diversion activities is a major component of an effective safeguards regime.

(3) NRC's communications to the Congress on the proliferation potential have not always been "correct," with particular reference made to the Chairman's 1987 testimony on the U.S.-Japan Agreement in which the potential uncertainty in plutonium material balance could be as high as 300 to 400 kg (or in Jennekens' explanatory comments, as many as 50 weapons).

(4) Harold Denton's memorandum to the Commission of March 15, 1988 in which he stated that the U.S. has no guarantee of the application of future state of the art safeguards accountability techniques by the IAEA or Japan to the proposed JNFS 800 MT/yr facility was "an unfortunate choice of words." The Denton comment, he said, had not been understood by the Japanese nor the Europeans.

(5) The international safeguards community would probably have to reexamine the long-standing requirement for material access for safeguarded facilities and place greater reliance on on-line instrumentation and near real-time material accountancy. This was the view of an American safeguards expert at IAEA, Dr. James Lovett, with whom Harold Denton had spoken. Implementation of on-line near real-time material accountability methods (including, for example, NRC's Larry Wirfs "running inventory difference"

technique), coupled with an IAEA on-site presence and analytical capability at commercial reprocessing, facilities such as Japan's JNFS, PFPF, and JNS facilities should provide an adequate level of timely warning and hence deterrence. [Note: The specific views of Dr. Lovett on this issue, if not completely known, may be of interest to NNMS and should be obtained.]

(6) Concern continues in political circles in the U.K., FRG, France, Japan (and obviously IAEA) over the U.S. position on the undesirability of commercial reprocessing. In light of the official U.S. position it was understandable that NRC capabilities in the technology of reprocessing are declining with time. More frequent NRC visits to THORP, La Hague, and JNFS he believed may help improve this situation.

(7) The IAEA safeguards program budget freeze for the past 4 years and de facto continuance for another 2 to 3 years, coupled with increasing diffusion of enrichment and reprocessing technologies, will in all likelihood require a restructuring of IAEA's safeguards program. This will probably result in fewer Vienna headquarters staff and greater reliance on international teleconferencing for project status reports on IAEA safeguards programs in Member States.

Mr. Jennekens conveyed with some emotion his personal concern about the robustness and potential vulnerability of the international safeguards regime without the continued, multi-pronged support of the principal Member States. In light of Mr. Jennekens' references to the NRC statements, Commissioner Rogers assured the Deputy Director General of the Commission's support of the IAEA's safeguards role.

DG Hans Blix and DDG William Dircks

Dr. Blix commented on the Munich International Symposium held the preceding week and the reservations he had on the use of PSAs due to the difficulty of understanding of such techniques by the general public. This led to discussion of French versus Swedish approaches to filtered containment venting as a severe accident mitigation strategy. The Director General expressed his opinion that further "discussion" (harmonization) on containment functional design was needed. He posed the rhetorical question: Was a revised NUSS standard on containment required? (Obviously it was in the DG's opinion.) Commissioner Rogers, in response to a question, informed

Dr. Blix and Mr. Dircks of NRC's endorsement of Boston Edison's plan to implement containment venting procedures at the Pilgrim Station as a safety enhancement.

The discussion broadened to such issues as the need for containment of advanced reactors with passive safety features and potential safety enhancements to existing U.S. plants as may be revealed through the IPE program. The Director General commented on the essential requirement of a continued cadre of trained personnel for national commercial nuclear power programs, citing John Gray's 1988 ANS-ENS Plenary address in which he noted increasing reliance by many Western countries on military training instead of traditional university education as a basic source of nuclear power expertise. Discussion then turned to developing countries and the Director General made the following points:

(1) Developing countries are not likely to build large commercial reactors for a variety of reasons, among them shortages of capital and trained manpower, lack of an effective regulatory regime, and public opposition. The Build-Operate-Transfer (BOT) concept of introduction of nuclear power by suppliers of industrialized nations in developing countries appears to be the most probable solution to this problem.

(2) Dr. Blix had recently visited Brazil and Argentina and, based on discussions with the political leadership of each country, felt confident about compliance of facilities under IAEA safeguards with existing protocols. He was concerned on the other hand that IAEA safeguards did not extend in Brazil to the military uranium enrichment facilities (as opposed to the FRG-supplied Becker nozzle-type laboratory and demonstration scale facilities at Resende) and in Argentina to certain government facilities, (the 2MT/yr heavy water production facility at Atucha, and the "Unit 2" conversion facility, the pilot and demonstration uranium enrichment plants, and the pilot reprocessing plants). He acknowledged that in Argentina the intent (as regards support of the international safeguards regime) was also unclear regarding the issue of development of Peaceful Nuclear Explosives (PNEs). Brazil's desire for naval nuclear propulsion capability posed a similar issue. Nuclear submarines were not, he pointed out, prohibited under the Nonproliferation Treaty (NPT). Brazil's new constitution on the other hand appeared to sanction a "domestic NPT" and appeared to be incompatible with national development of naval nuclear propulsion capabilities.

(3) As to Pakistan and India, he believed that adequate surveillance of <u>de facto</u> enrichment and reprocessing facilities in both countries could be maintained through scrutiny of "trigger list" items by IAEA Member States supplemented by IAEA inspections of these items and Member State explanations as appropriate. The IAEA, he said, was sensitive to the issue of Member State composition of IAEA teams which routinely inspect reprocessing plants. On the other hand, there should be no difficulty in principle with Libyan or Pakistani experts' participation in OSART evaluations of commercial LWR power plants.

(4) On balance, since no expansion will be possible in 1989-1990 in the IAEA Safeguards program due to budgetary considerations, improvements would be necessary in the effectiveness of IAEA's safeguards activities. These IAEA initiatives were necessitated in part by recent political attacks by the "Greens" in Western European nations (particularly in the FRG, where an expose in <u>Der Speigel</u> had alleged an inability by IAEA to maintain an effective international safeguards regime). The Director General stated the Agency's Safeguards Information Reports (SIRs) were to be made public as a counter to such institutional attacks to refute the allegation that the IAEA is a "high priesthood" whose function it is to "watch" all countries.

(5) As to the safety of the international commercial nuclear enterprise, he believed:

- OSART inspections, especially of the developing nations, would prove an effective tool for achieving enhanced safety (although IAEA Member States would have to underwrite the expense of Eastern European plant inspections).

- Commercial nuclear waste management and disposition and regulatory encouragement of development of advanced reactors represented two sectors in which IAEA efforts had been "too passive."

New initiatives were foreseen, such as IAEA peer reviews of regulatory practices by "sector" (i.e., nuclear waste, radiation protection, operator training, etc.). The intent would be to offer collegial assistance, not to "give grades."

(6) As to the issue of civil liability in instances of accidents or possibly contamination from faulty waste repository design or operation, such issues were resolvable within the framework of existing protocols

(Paris Convention) within the West. The USSR and East European countries were presently developing a similar convention between East European governments. He questioned whether the U.S. was for or against such conventions in principle; there was no intervention level specified in the previous (Paris) agreement to which the U.S. is a signatory.

(7) While he believed the Agency's NUSS standards codified common experience and wisdom, the recent IAEA Questionnaire had revealed their general lack of use by respondent Member States. The DG felt NUSS codes could serve as complementary standards to a State's existing codes, standards, and regulations.

Commissioner Rogers thanked Dr. Blix and Mr. Dircks for the opportunity to meet and discuss important issues the Agency faces and to assure them of the Commission's support of IAEA's mission.

Assistant Deputy Director Morris Rosen

Mr. Rosen reviewed discussions during a recent trip to Japan and the U.S. He had attended the INPO CEO Conference in Atlanta where senior utility executives had stated that in the U.S. commercial nuclear power as an option was tantamount to bankruptcy. In Washington World Bank officials not only pressed for OSART inspections in countries having "energy sector" loans, but proposed IAEA peer reviews of the regulatory infrastructure as well. For most countries the Bank recommended small gas-fired power plants as a preferred electrical generation option, citing thermodynamic inefficiencies from over-electricitization in developing countries. Where fossil fuel supplies were constrained, however, as in Poland and Hungary, Rosen believed nuclear power was needed. He made the following points:

(1) The World Bank would publish an environmental assessment of electrical generation in the next few months, commenting on the Greenhouse effect and the possible benefits of modular advanced nuclear plants featuring passive safety (i.e., financially less risky... no loss of investment) for electric power purposes. Indonesia, for example, was known to be interested in the Westinghouse AP-600, a 600 MWe Advanced PWR with passive safety features.

(2) One must expect delays in construction if domestic companies with limited expertise undertake parts of the

project. (e.g., faulty concrete pouring, etc.) In some developing countries (e.g., Brazil, Mexico, Philippines) a lot of money was tied up in grandiose projects.

(3) The adequacy of trained nuclear manpower in developing countries is not seen as a major problem since commercial nuclear plants contracted for would be through a BOT contract arrangement. On the other hand, an effective regulatory organization in the host country under such arrangements would be essential.

(4) In Brazil the recent OSART review of Angra Unit 1 (Westinghouse 600 MWe turnkey contract) had revealed operational issues, although an OSART review of an identical Westinghouse unit in Yugoslavia (NPP Krsko) had been very satisfactory. Clearly the distinction lay in the efficacy and will of operational management.

(5) USNRC-IAEA relations he thought were good; Chairman Zech had been a strong advocate of IAEA operational safety initiatives. He noted, however, some U.S. difficulty in accepting the concept of peer review of U.S. facilities by international organizations (e.g., the two year delay in accepting OSART inspections of U.S. plants). It would be helpful to schedule more frequent OSARTs in the U.S.; the 2 year gap since the Calvert Cliffs OSART is too long.

(6) PSAs can be an effective regulatory tool, but one should view the "bottom line" (core melt probability and off-site consequence) with caution. Implementation of NUSS codes is preferable, he believes, since they are understandable to operators. Performance indicators were useful for trend analysis, but are subject to misuse. Regulatory bodies should refrain from redoing calculations... let industry do the work and just review the methods and conclusions; i.e., check if the bottom line is reasonable. Emphasis should be on regulatory management judgement rather than mere technical competence.

<u>DDG Noramly bin Muslim</u>

Dr. Noramly bin Muslim, Head of the Department of Technical Cooperation, explained that for practical purposes the Agency's Technical Assistance and Cooperation Fund (TACF), amounting to $83.2 million for the years 1989-1990, attempts to address the needs of four geo-political IAEA Member State groups: the Peoples Republic of China, the European Eastern-Bloc countries, the European and Latin American Western-Bloc countries, and the "Bloc of 77" (developing countries). Proposed distribution of the TACF by region and by basic activity (AAPC codes) is shown in Figure 16.

Allocation by the four broad geographic regions is seen to be essentially equal: Asia and Pacific - 25 percent; Africa - 24 percent; Latin America -23 percent; and Middle East and Europe - 26 percent. Allocation by type of science or activity (AAPC codes) is revealing: nuclear technology excluding radiation protection accounts for only 8 percent, the balance devoted to food and agriculture (16 percent), human health (14 percent), physical and chemical sciences (20 percent), industry and earth sciences (13 percent), radiation protection (10 percent), and so on. Dr. Noramly explained that his department provided project oversight and funding for over 500 projects in the "Bloc of 77" developing countries and has since forwarded an IAEA report, "Technical Co-operation, Project Schedules for 1989-90," dated 11 November 1988, which contained project summary descriptions and funding by country and region. The department focuses on applications of the nuclear sciences to problems of nutrition, agriculture, health, and industrial development to developing countries. He acknowledged with appreciation U.S. support of TACF programs through providing cost-free experts. He singled out for mention the risks from selling nuclear equipment (e.g., sealed sources) to nations without adequate regulatory bodies.

DDG Leonid Kostantinov

Dr. Kostantinov will leave the Agency in April 1989. He said that he is awaiting an NRC proposal to examine the need for improved instrumentation in plant aging research, an outgrowth of his discussions with Commissioner Rogers following the International Symposium on Plant Aging in September 1988. He and Commissioner Rogers agreed the prospective activity should be based on the need for specific plant-aging instrumentation rather than broad programs to improve instrumentation. He said he would approve such an NRC-sponsored proposal if made. The Deputy Director went on to make the following points:

(1) He considered the establishment of WANO (World Association of Nuclear Operators) a significant accomplishment and an international plant operation safety enhancement mechanism. He saw evolution of WANO as a forum for international dialogue not only concerning plant operational procedure but design, construction, and maintenance issues as well, citing LWR instrumentation thimble wear and steam generator degradation.

(2) He saw as challenges to the international nuclear community the following:

- Economic evaluations of electrical energy alternatives for developing countries.

- Worldwide dissemination of power reactor information.

- Improvement of current levels of safety in the operation of nuclear power plants.

- Development and eventual deployment of a new generation of advanced reactors (HTGRs, LMRs, LWRs, HWRs). In this regard he prefers evolutionary designs as opposed to "revolutionary" concepts such as PIUS.

- Advanced countries should examine the Greenhouse effect; developing countries have too many immediate problems to handle this topic.

(3) While developing countries do not have the resources to undertake commercial nuclear projects on their own, the IAEA's computerized Model Assessment of Energy Demand (MAD) used by the World Bank clearly reveals that additional nuclear power is needed in the developed countries. He believes continued OSART inspections of existing nuclear plants can help stabilize adverse public opinion with time; he does not favor regulatory review by OSART-type teams.

(4) The Agency had consented to conduct a PSA design review of a Soviet power reactor in Gorky (an announcement of this was to be made by Director General Blix in mid to late 1989) to assuage public concerns about its safety. He wondered if this concept (IAEA international peer review) possibly would be useful in U.S. attempts to license the HLW repository at Yucca Mountain. He said that the U.K. has requested an IAEA review of its proposed Low and Intermediate Level Waste facilities in Britain. For HLW, the IAEA has prepared a code of practice and safety guides which will be presented to the Board of Governors by February 1989. France is still reviewing certain parts of this report.

The meeting concluded with Commissioner Rogers thanking Dr. Kostantinov for receiving him and discussing the Agency's safety agenda and planned initiatives.

APPENDIX E

FRG RADIOACTIVE WASTE REPOSITORIES: GORLEBEN AND ASSE

Gorleben

Surface exploration was completed in 1984, and exploratory shaft sinking by the deep freeze method began in mid 1986. An accidental worker fatality in late 1987 has temporarily halted progress. The effect of surface water from the overlying strata on the salt dome is always a concern in saltmining, and since the accidental inflow of water or brine can never be excluded with certainty, federal authorities required the inclusion of such an inflow apriori in the safety analysis. Reasonable scenarios were postulated on the basis of more than 100 years of German salt mining experience. DBF believes the postulated inflow would be critical from a safety point of view only for a limited time period in the post-operational phase in repository tunnels which remain permeable to water/brine intrusion even though they are backfilled. Figure 17 contains drawings and explanations of the Gorleben repository.

The final Gorleben facility will be completed in 2008, and vitrified HLW from La Hague and Sellafield will be emplaced at a projected cost of 1.5 to 2 pfenning/kwh (0.9 to 1.2 m/kwh). The capital cost of Gorleben is estimated to be 100 million DM ($59 million). The cost of emplacing 1,500 MT of spent LLR fuel elements as HLW in Gorleben for 40 years is 50,000,000 DM or 20,000 DM/MT (about $11,800/MT of HLW). Some 20 MT of LLW/ILW packages are now stored in an above-grade storage facility at Gorleben. Gorleben will thus serve as Away From Reactor (AFR) Storage until the Wachersdorf reprocessing plant is in operation.

Asse

Up to 1979 approximately 124,000 drums of low level waste and 1300 drums of medium level waste had been emplaced. Since January 1979 research and development projects have been continued without any further storage. Present experiments include issues related to heat-generating, highly radioactive waste. Tests have included emplacement and monitoring of simulated HLW vitrified waste with electric heaters. Additional experiments have included Co-60 sources in simulated HLW vitrified waste to investigate the combined effect of radiation and release of heat. In 1989 Pacific Northwest Labs (PNL) under a BMFT-DOE agreement will arrange for transport of 30 actual HLW (Sr and Cs) vitrified waste forms to Asse for a 5 year emplacement demonstration which will be extensively monitored. This project is described in "High-Level Radiioa-ctive Waste Test Disposal Project in the Asse Salt Mine (FRG)," by T. Rothfuchs and R. Stippler, March 1987 Symposium on Waste Management, Tuscon, Arizona. The paper is in the report files. Figure 18 is a schematic of the cross section of a HLW emplacement borehole, showing the 2 to 3 mm titanium palladium coating as corrosion protection on the high strength stainless steel borehole liner.