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ORNL

FOREIGN TRIP REPORT

ORNL/FTR-3753

DATE: September 24, 1990

SUBJECT: Report of Foreign Travel of William R. Corwin, Manager, Heavy-Section Steel Irradiation Program, Metals and Ceramics Division

TO: Alvin W. Trivelpiece

FROM: William R. Corwin

PURPOSE: To visit the Prometey Complex in Leningrad, USSR, to assess additional potential US/USSR collaborative nuclear safety research; to participate in the International Seminar on Irradiation Embrittlement of Reactor Pressure Vessels sponsored by the Gesellschaft für Reactorsicherheit in Cologne, FRG, and to visit both the Materialprüfungsanstalt, Stuttgart, FRG, and Siemens, A.G., Erlangen, FRG, to discuss sampling and testing techniques presented earlier in the week at the irradiation effects seminar used to assess existing irradiation embrittlement of reactor pressure vessels still in service.

SITES VISITED:

8/29-31/1990	Leningrad, USSR	Prometey Complex	G. Karzov
9/03-06/1990	Cologne, FRG	Gesellschaft für Reactorsicherheit	M. Banaschik
9/07/1990	Stuttgart, FRG	Materialprüfungsanstalt	K. Kussmaul
9/08/1990	Erlangen, FRG	Siemens, A.G.	G. Barthelome

ABSTRACT: The traveler served as a member of the two-man U.S. Nuclear Regulatory Commission sponsored team who visited the Prometey Complex in Leningrad to assess the potential for expanded cooperative research concerning integrity of the primary pressure boundary in commercial light-water reactors. The emphasis was on irradiation embrittlement, structural analysis, and fracture mechanics research for reactor pressure vessels. Detailed inspections were made of the structural and fracture mechanics testing facilities as well as the hot cells used for examination of irradiated pressure vessel steels. Principal areas identified for collaborative research

included: exchange and subsequent analytical modeling of isothermal vessel failure, pressurized-thermal shock, clad beam, and dynamic wide-plate test data; comparison of alternative simplified analytical analysis methods for elastic-plastic fracture; and exchange of unirradiated or irradiated materials for comparative examination of fracture properties.

At the irradiation seminar in Cologne, presentations were made by German, French, Finnish, Russian, and U.S. delegations concerning many aspects of irradiation of pressure vessel steels. The traveler made presentations on mechanisms of irradiation embrittlement and on important aspects of the Heavy-Section Steel Irradiation Program results of irradiated fracture mechanics tests. Other topics covered included effects of irradiation flux, fluence, temperature, spectrum, and annealing, as well as materials-related parameters including chemical composition, processing history, and metallurgical structure on irradiation-induced embrittlement. During subsequent visits to MPA in Stuttgart and Siemens in Erlangen, follow-up discussions were held on detailed aspects of presentations made at the seminar.

INTRODUCTION

The original purpose of the trip was to attend the seminar on irradiation embrittlement sponsored by the Gesellschaft für Reactorsicherheit (GRS). Its original focus was on the effect of embrittlement on the overall safety of the Russian-built pressurized water reactors (PWRs) at the Greifswald site in East Germany (GDR). This was subsequently enlarged to cover a wide range of related topics, which included both Western European experience, as well as that derived in Russia, Finland, and the GDR on Russian-built reactors and their materials. The visits to the Materialprüfungsanstalt (MPA) in Stuttgart and Siemens, A.G. in Erlangen were included to provide an opportunity for detailed discussions of the material presented by their participants from those sites at the GRS seminar.

The purpose of the trip to the Prometey Complex in Leningrad was to examine the potential for additional collaborative research in the field of nuclear safety between the United States and the USSR. The trip was a follow-up to the visit of Commissioner Remick of the U.S. Nuclear Regulatory Commission (NRC) made in conjunction with the recent meetings in Moscow of the U.S.-USSR Joint Coordinating Committee on Civilian Nuclear Reactor Safety (JCCCNRS), under whose auspices current interactions in this area are conducted. In addition, it provided an excellent opportunity to discuss Russian experience on irradiation embrittlement and other topics related to light-water reactor (LWR) primary system integrity in great detail. The traveler was accompanied by C. Z. Serpan, NRC, throughout the trip.

VISIT TO THE PROMETHEY COMPLEX Leningrad, USSR

Facilities

Three sites associated with Prometey were visited: the administrative offices located within Leningrad; the Department of Structural and Mechanical Testing, located about 20 miles outside the city in the town of Gatchina; and their irradiation hot-cell testing facilities located within the grounds of the Institute of Nuclear Physics, also in Gatchina.

The visit to the hot cell facility was unusual and very important for reasons beyond its technical content. It was the first visit to that facility by any persons from any non-Soviet block country, and is clearly indicative of their interest in increasing western interactions in this area.

Structural and Mechanical Testing

Testing equipment is situated in two main areas. One area contains test frames used for normal material strength and fracture toughness testing and the other contains unusual equipment with the potential for experimental danger in numerous blast chambers. The normal mechanical and fracture testing equipment includes: approximately two screw-driven and ten servohydraulic universal testing frames with capacities up to 50 tons and one with a 1000-ton capacity, all made by Schwanke in FRG; one 8000-ton capacity, horizontal hydraulic test frame manufactured by Skoda in Czechoslovakia; a C-hammer-type, automated, instrumented Charpy impact machine with 300-J capacity; approximately six small and one large hydraulic pressurized horizontal cyclic disk bulge test frames for samples of 510 and 1300 mm diameter, respectively; and miscellaneous equipment for welding, specimen preparation, and examination. Typical material tests included impact, tensile, fracture toughness,

fatigue, and fatigue crack propagation (FCP) in air, vacuum, and various nonpressurized corrosive environments. In addition, testing of small welded structures under service conditions and large wide-plate crack arrest tests are performed. Fracture toughness testing is limited to K_{Ic} and crack opening displacement (COD) tests for materials in or below the transition. No ductile fracture toughness testing capability was observed. Overall, the type of mechanical and electronic equipment was indicative of a good quality materials testing lab but, with the exception of the 8000-ton universal testing frame, it had no unusual or unique capabilities and its technology was less than state-of-the-art. Wide-plate crack arrest tests are being conducted in the 8000-ton machine. Three have been done to date, one on a pressure vessel steel and two on off-shore structural materials. The plates are 140 mm thick and about 1.5 m wide but only 2-3 m long. Instrumentation appeared minimal, with ladder gages (and their problems) used to obtain crack-tip position during the tests. Since the pressure vessel steel test had been completed only recently, additional information on the test was not yet available. Welding of the specimens to the pull tabs caused enough difficulty that, despite Prometey's proffered expertise, they had to have them welded elsewhere.

The blast chambers contained experimental equipment with the potential for mechanical, hydraulic, or environmental danger to the researchers. Included in the chambers were: facilities for isothermal and pressurized thermal-shock (PTS) vessel experiments; very high-temperature tensile test systems, autoclaves for FCP and stress corrosion cracking (SCC) in LWR water chemistries, temperatures, and pressures; environmental creep testing frames for vacuum and simulated, high-temperature gas-cooled reactor (HTGR) coolant; a planar, multiaxial creep loading frame for testing turbine discs; isothermal fatigue and FCP testing for pressured components including nozzle corners; thermal-fatigue testing of mechanically loaded plates; and a lead/lithium liquid metal corrosion loop. The isothermal vessel experiments are being conducted on intermediate size test vessels (ITVs) which are 3.5 m long, 1.55 m I.D., and have a wall thickness of 82 mm. All ITVs are tested in the mid-to-lower transition temperature range. Cracks are located on the inside and outside of the ITVs and are fatigue sharpened prior to testing. The PTS experiments are on vessels which are about 1.2 m in diameter and have a 140-mm wall thickness. The vessel is internally pressurized up to 300 bar and simultaneously cooled on the outside by gravity flow of water into an open container surrounding the vessel through twin 250-mm pipes from two 5000-L tanks located 5 m above the vessel. Flaws in the first vessel now being examined are located on the outside and include a long shallow flaw (500 mm) and a deeper semielliptical flaw at 180° from the shallow flaw. In the first three transients, the flaws were only sharp machined flaws with no additional precracking. Following the apparent lack of success of crack initiation during those transients, precracks were introduced with the spontaneous centerline cracking of a high carbon weld pass placed in each flaw. The technique is used successfully by the Finnish at VTT, but cracks have not yet been verified on the Russian PTS vessel. Instrumentation on both the ITV and PTS vessels is only from the outside and includes one crack mouth opening displacement (CMOD) gage per flaw, strain gages, and thermocouples (TCs). For the PTS test, the TCs are located on the outside surface and at depths of 10, 25, and 50 mm, apparently at only one location. All instrumentation for the experiments is provided by the Finnish as a joint venture, and its absence made further examination impossible during the visit. The overall impression of the blast chamber equipment was that it covered a wide range of disciplines and included some innovative ways of resolving technical problems. However, the breadth of the capability was not matched by the state-of-art instrumentation and measurement techniques needed to gain the maximum, or in some cases even the minimum, amount of information required to assess the results of the tests. It was flattering to see direct copies of many of the tests performed by the NRC-sponsored Heavy-Section Steel Technology (HSST) program (e.g., ITV, PTS, and wide-plate tests) and it is apparent the Russians read a great deal of western literature. However, the lack of attention to detail, needed for meaningful and successful results such as the lack of sharp flaws and

adequate instrumentation in their PTS test, raises concerns about the value of any joint structural experiments.

The irradiation hot cells consisted of two back-to-back banks of six cells each, and one receiving and distribution cell located between the banks at one end. Cells are well laid out with usually only one task per cell. A centralized remote distribution system allows "hot" material to be unloaded in the receiving cell and then moved from any cell to any other cell without further violation of the containment barrier. The main activity of the overall facility is the examination of irradiated structural materials, primarily reactor pressure vessel (RPV) steels. Capabilities are designed for the mechanical, metallurgical, and fracture properties assessment of irradiated RPV steel. Facilities contained in the cells include those needed for: capsule disassembly; sawing of moderately large pieces of irradiated steel, such as RPV trepan; conventional machining operations including a mill and a lathe; specimen fabrication using electrodischarge machining (EDM); metallurgical specimen preparation; optical microscopy; density measurements for swelling assessments; tensile testing of small specimens; high and very high temperature fatigue testing in air and vacuum; FCP testing in air; full-size, non-instrumented Charpy specimen impact testing; and K_{Ic} testing of specimens up to 25 mm thick. In addition, a small impact machine with a maximum energy of 15 J, a full-size 300-J instrumented impact machine, a modern scanning electron microscope (SEM), and a transmission electron microscope (TEM) were also available adjacent to the hot-cell area.

Irradiation capsules typically used for test reactor irradiation are long (~1 m) with a relatively small diameter (~75 mm OD). They normally hold full and subsize Charpy V-notch (CVN) impact, round miniature tensile (~2.5 mm diam), and round 12.5-mm-thick compact tension (CT) specimens. Irradiations are typically performed at one of the test reactors also located at the Institute for Nuclear Physics. One of those reactors is currently operating and is an 18-MW water-cooled, water-moderated, swimming pool type. No information was obtained on the second shutdown reactor. During irradiation, the temperature of the specimens is controlled solely by altering the absolute pressure of helium which fills an annular gap on the outside of the capsule. Estimates of temperature uniformity within the capsule were stated to be $\pm 5^\circ\text{C}$, but this would appear to be too good for the capsule construction and control methods. In subsequent conversations with personnel from the Kurchatov Institute in Moscow, substantial doubts were raised concerning the irradiation temperature of the Prometey capsules.

Overall, the quality of the irradiation testing facilities seemed to be quite good, with exceptionally good handling, machining, and specimen preparation capabilities. The only noticeable limitations they seem to have are for testing irradiated subsize CVN impact and large (>25 mm thick) fracture toughness specimens and for electron microscopy of irradiated materials. In addition, no ductile fracture toughness testing is performed.

Large-Scale Testing Related to Pressurized Thermal Shock

Prometey has conducted three series of what they call PTS tests, starting with flat plates about 140 mm thick by about 280 mm wide. The plates had a series of thumbnail cracks within and through a cladding layer. They were heated, loaded in three-point bending, and sprayed with cold water. These tests were unsuccessful. The second series used similar specimens, although 1 m wide and unclad, which were loaded in tension. They contained a 15-mm deep mechanical notch artificially embrittled but not fatigue sharpened. Cooling was applied to both sides and crack extension of about 5 mm occurred. Finally, the third series used a model vessel with exterior cracks, interior pressure, and exterior cooling by cold water flooding as described above. They have run three transients on the same vessel, but with no apparent crack extension. Additional weld embrittlement of the crack

tip may allow successful testing. The report on their second series of tension-bending loaded flat plates with two-sided quench and thumbnail cracks was requested for our calculations and comparison. Also requested were reports on their third series of PTS tests, when available, in exchange for our pressurized thermal-shock experiment (PTSE) reports.

The analysis efforts occurring through CSNI-PWG3 Fracture Analysis Group which will examine worldwide, state-of-the-art predictions of large-scale verification tests were described to Prometey. They have not seen the prediction data. It was agreed that ORNL will send, if approved by NRC, prediction data to Prometey. They will calculate, and either send results to ORNL or present at future meetings. (This will be an exceptionally valuable way to assess the level of their analytical fracture mechanics capability.) Serpan, NRC, however, must first assure that such actions with the Soviets will be acceptable to OECD-NEA.

Prometey has apparently developed some alternatives for elastic-plastic fracture mechanics analysis methods, and wishes to exchange such information with Merkle of ORNL for mutual examination and comment. Prometey plans to send the work of Karzov, Margolin, and Shvetryova in exchange for work by Merkle. Considering the apparent lack of numerical, computer-based analytical capabilities, this in concert with the above CSNI-FAG exchange, is likely to provide the most valuable information available concerning their analytical abilities.

Thermal and Irradiation-Induced Embrittlement

While the great majority of embrittlement information from Soviet studies was presented the following week in Cologne, FRG, two additional items were addressed at Prometey. They have measured shifts of 25 to 40°C in 15x2VMΦA steel (the VVER-1000 material) after thermal aging for 10 years at 350°C. While perhaps not too surprising for that steel (>2% Ni), time, and temperature, they go on to predict up to an 80°C shift for this steel in 70 years of normal service.

In a separate study, several years ago, Karzov found a severe flattening of the slope of the K_{Ic} vs temperature behavior for a high-fluence, low-temperature irradiation of 1×10^{21} n/cm² ($E > 0.5$ MeV) at -80°C. Similar experiments to lower fluences in the 10^{19} range showed only a shift but no slope change. He will try to obtain a report detailing the results to send to us.

During the course of the discussions, an odd pattern seemed to emerge. Karzov would at first describe the extensive and comprehensive work on a given topic, but when pressed for details, he would retreat, until he finally described a much lesser scope of actual work. When we explored his interest in their sending an expert to a U.S. lab for up to six months, he immediately declined saying that anyone worthwhile would be too valuable to be gone so long, and no one knows enough English to properly interact. For whatever reason, Karzov seemed much less interested in actual cooperation than was expected from the recent letters to NRC Commissioner Remick. If anything, the traveler got the impression that they were interested in performing experiments under contract to the United States rather than in collaborative research.

**PARTICIPATION IN THE INTERNATIONAL SEMINAR ON
"IRRADIATION EMBRITTLEMENT OF REACTOR PRESSURE VESSEL"**
Cologne, FRG, and at satellite meetings at the MPA, Stuttgart, FRG,
and Siemens, A.G., Erlangen, FRG

Overview

The original purpose of this seminar was to provide the soon-to-be united East and West German nuclear regulators with additional information upon which to base the decision for continued operation of the early models (230-type) of the Russian-built 440-MW PWRs (VVER 440's) located at the Greifswald site in the GDR. However, with those decisions apparently made (the units will probably not be restarted), the scope of the seminar was enlarged to include Russian, West European, and U.S. experience on RPV radiation embrittlement and related topics germane to PWR primary system integrity.

A total of about 30 delegates from the FRG, the GDR, France, Finland, the USSR, and the United States were invited to participate. The seminar included four days of presentations at the GRS in Cologne and a one-day tour of the MPA in Stuttgart. Principal topics covered in the seminar were: mechanisms of irradiation embrittlement; effects of irradiation flux and spectrum and material chemistry on embrittlement; postirradiation annealing; thermal embrittlement; relative shifts of fracture toughness and CVN tests; and a comparison of regulatory approaches. Following the seminar, the traveler participated in related discussions with personnel from Siemens, A.G. in Erlangen. Since the information obtained at GRS, MPA, and Siemens is closely related, it is summarized together below.

Much of the information presented by the Russian delegation was however, previously presented in the meeting of the JCCCNRS in Moscow in June of this year and is well documented in the foreign trip report of R. K. Nanstad, ORNL/FTR-3667. Only new information will be reported here.

Embrittlement Mechanisms and Modeling

Information on embrittlement mechanisms was presented by the Russians and was focused on the dislocation structures produced by irradiation, with some additional items concerning apparent chemistry effects in general and independence of nickel and copper in particular, as well as the lack of effect of grain size. Gurovitch, from Kurchatov, described extensive TEM work on dislocation structure, very similar to that reported at the recent JCCCNRS WG-3 meeting.

Kryukov, from Kurchatov, described a study in which the effects of a number of irradiation variables were deduced from an extensive nonlinear regression analysis of their surveillance data. His conclusions were that copper itself did not cause significant embrittlement for their materials, but that its interaction with phosphorus was important as was the effect of phosphorus by itself. Interestingly, he also stated that the principal embrittlement contribution of phosphorus occurred at low fluences, with subsequent embrittlement due to copper-phosphorus interactions and an as-of-yet unknown "matrix" defect. Kryukov also described a systematic study of the effect of grain size on the embrittlement rate. Steels with intentionally varied combinations of phosphorus, copper, and silicon were given a variety of heat treatments producing microstructures in which average grain size varied from about 11 to 500 μm . While the initial values of Charpy transition temperature, T_k (temperature at 48 J), was quite strongly influenced by grain size as expected, very little difference in shifts in T_k was observed as a function of grain size.

Nikolaev, from Prometey, described annealing studies of Fe-Ni and Fe-Cu binary alloys irradiated at low temperatures ($\sim 140^\circ\text{C}$) which indicated independent mechanisms of irradiation damage. This conclusion was based on the failure of the irradiation-induced increase in the electrical resistivity of the nickel-bearing alloy to anneal out, in contrast to that of the copper-bearing alloy. He interprets this to mean that the copper is going back into solid solution whereas the nickel agglomerates. This interpreted independence of copper and nickel does not generally agree with the U.S. experience. Nikolaev also described test reactor results of irradiation on the higher nickel steels (1.0-1.8 wt%) used for the new generation of VVER 1000 PWRs. Embrittlement rates were substantially greater than predicted using the copper- and phosphorus-based predictive equations generated from the much lower nickel containing VVER 440 steels (0.1-0.3 wt %). Indeed, the rate of embrittlement is great enough to exceed the Russian Normative Curve (equivalent to Reg. Guide 1.99) at high fluences [e.g., above $\sim 1 \times 10^{20} \text{ n/cm}^2$ ($>0.5 \text{ MeV}$)].

The traveler presented U.S. information on mechanisms, concentrating on the ORNL work on atom probe field ion microscopy (APFIM) by Miller and theoretical studies of Stoller and Mansur on effects of nonsteady state concentrations of point defects during the initial irradiation transient. The information appeared new to most of the participants, including the Russians. Of particular interest to the Russians were the ultrafine scale studies of irradiation-induced clusters and precipitates, since their highest resolution techniques are still too coarse to allow such examinations.

Effects of Flux and Spectrum

Kryukov described the results of test and power reactor irradiations of high phosphorus, moderate copper welds which indicated a strong inverse relationship between flux levels of 0.4 to $7 \times 10^{12} \text{ n/cm}^2\text{s}$ ($>0.5 \text{ MeV}$) and the measured shift in the T_k . Ahlstrand, of Imatron Voima (IVO), Finland, stated that for comparable steel they had observed no effect of flux between 0.4 and $4 \times 10^{12} \text{ n/cm}^2\text{s}$ ($>0.5 \text{ MeV}$). The fluences in both sets of experiments ranged from 0.2 to $5 \times 10^{20} \text{ n/cm}^2$ ($>0.5 \text{ MeV}$). Kryukov's regression analysis of surveillance data described above also revealed a flux dependence of various components of the total observed embrittlement. He concluded that the matrix and small copper contributions to embrittlement are greater at fluxes below about $5 \times 10^{11} \text{ n/cm}^2\text{s}$ ($>0.5 \text{ MeV}$) than at higher fluxes for all fluences up to about $5 \times 10^{20} \text{ n/cm}^2$ ($>0.5 \text{ MeV}$). The contributions of phosphorus and the phosphorus/copper interaction as a function of flux, however, depend on the fluence level with phosphorus effects greater, relatively speaking, at low fluences than at high ones, exactly the opposite of the interaction contribution. In the traveler's opinion, however, the amount of detail claimed to be derived from the regression analyses is exceedingly large and may well surpass the amount which can be meaningful or reliable.

In the study of the trepan from the Gundremmingen reactor, Föhl, MPA, stated that the apparent gradient in the irradiated Charpy transition temperature through the thickness of the vessel forging, originally attributed in part to effects of the flux, has now been shown to agree with a microstructural gradient. There is metallographic and fractographic evidence of a significant increase in the size and effect of MnS stringers as the inner surface of the RPV is approached. Föhl has tentatively concluded that there was no significant effect of flux or spectrum on the RPV forging for the conditions examined.

Two sets of experiments were described which both indicated that there is little or no effect of thermal neutrons on irradiation at RPV operating temperatures. Serpan described the British irradiations of the Gundremmingen material in spectra with and without a large thermal neutron component in which no change in embrittlement was observed. Kryukov stated that specimens

exposed to typical LWR spectrum with and without a cadmium coating to absorb any thermal neutrons exhibited similar levels of embrittlement.

In light of the apparently subtle or contradictory effects of flux on embrittlement for LWR operating conditions and the need for long-term research to resolve the details, Schultz, of GRS, suggested that it would be beneficial to pool all available surveillance results from different countries. This would provide a larger and wider ranging data base from which to analyze flux effects at the low levels applicable to power reactor operation. It appeared the French and German delegations, as well as Serpan all supported the idea and will check, as needed upon return, to see if pooling of data from their respective countries will be possible.

Postirradiation Annealing

Amaev, Kurchatov, described the extensive Russian experience with annealing. To date they have annealed seven unclad and two clad VVER 440 vessels. All have been annealed at 460°C and their plans are to continue to perform annealing at this temperature. The temperatures of the anneals are based on a large experimental data base which includes effects of annealing time and temperature, irradiation temperature, material chemistry, and, recently, irradiation rate. Their results have shown that they can obtain as much recovery as possible at an annealing temperature which is ~190°C above the irradiation temperature. Most of the recovery occurs within the first 10 h, but current plans use 150 h to assure the maximum recovery obtainable. For the majority of the VVER 440s which require annealing, the temperature of operation is 270°C. Whether full recovery of unirradiated properties is possible depends on the chemistry of the material, with the level of phosphorus impurities being the most important factor in determining residual embrittlement. They are unwilling to anneal at too high a temperature since some of their studies at an annealing temperature of 500°C have shown a tendency toward intergranular fracture caused by excessive segregation to the grain boundaries. The intergranular fracture has only been observed in anneals lasting over 1500 h, but the possibility of repeated anneals for the same vessel has been enough to cause the Russians to go no higher than 460°C. In studies of repeated irradiation and annealing cycles, the data for several materials tend to show both a slight reduction of embrittlement rate upon reirradiation and a tendency to saturate the total amount of embrittlement as measured by the CVN shift. Nonetheless, to be sure they are being conservative, the Russians assume the reembrittlement rate of the shift following annealing is the same as the original embrittlement rate and add to it any residual phosphorus-related shift expected after the anneal to predict subsequent values of T_L .

Recent studies of annealing of the trepans removed from the vessel of Novovoronezh Unit 1 which operated at only 250°C to a maximum fluence of 2×10^{19} n/cm² (>0.5 MeV) were described. Annealing of these was done at 360, 420, 460, and 500°C as well as some of the outer layers at 650°C to simulate unirradiated material. Annealing at 460 to 500°C of material from the inner wall of the vessel showed essentially full recovery of unirradiated properties measured by both CVN and 0.5TCT specimens (12.5 mm thick) when compared to the material from the outside of the vessel annealed at 650°C. A substantial controversy arose between Prometey and Kurchatov personnel regarding the amount of recovery which occurred at 420°C. Nikolaev, Prometey, described data showing that the recovery of Novovoronezh material annealed at 420°C was appreciable less than the trend previously established in the bulk of the Russian data. He attributed this to a flux effect in that the Novovoronezh vessel material had been subjected to a flux of only $\sim 1 \times 10^{10}$ n/cm²s (>0.5 MeV) whereas most of the rest of the data came from accelerated surveillance or test reactor irradiations with flux levels of 0.1 to 1.0×10^{12} n/cm²s (>0.5 MeV). Kryukov, Kurchatov, violently disagreed with both Nikolaev's data and conclusions. It became apparent that there is a very substantial amount of jealousy and competition between the two organizations and this needs to be

borne in mind when examining their data. In general, the Kurchatov Institute, from which Amaev has the lead for JCCCNRS WG-3, performs most of the surveillance testing for the Soviet reactors and RPV irradiation research using accelerated surveillance-type capsules in power reactors. On the other hand, Prometey Complex performs most of the irradiation experiments for RPV materials in test reactors and also does the bulk of the structural mechanics and large-scale fracture mechanics experiments.

Results of the annealing on the Greifswald Unit were presented jointly by Valo, VTT, Finland, and Leitz, Siemens, FRG. The vessel was one of the unclad, old type VVER 440s which had been in service for 15 years. The CVN transition, T_k , had been estimated to be between 170 and 200°C. Following annealing, small boat samples about 28 × 67 × 3-5 mm deep were removed using EDM techniques to allow an assessment of post-annealing residual embrittlement to be made. Subsize CVN specimens 3 × 4 × 27 mm were fabricated by VTT from these boats using the stud welding reconstitution method. The specimens contained a 1-mm-deep 60° notch with a 0.1-mm radius at the notch tip. Subsequent testing and correlation of the results at Siemens resulted in a T_k of 35°C which was lower than the 66°C which had been estimated using Russian predictive methods. The correlation of the subsize to full-size specimens was based on a large body of German data on similar sized specimens. The correlation showed a 1:1 slope of T_k for the full and subsize CVN specimens, measured at 48 and 2.2 J, respectively, but with a constant offset of 65°C, with the larger specimens yielding the higher transition temperature.

In subsequent private discussions, Amdev stated that Novovoronezh units two and three are the next vessels scheduled to be annealed, contrary the information obtained at the JCCCNRS meetings. Both will be annealed within the next year.

Thermally Induced Embrittlement

New information was obtained from the Russians regarding their previous claims that thermal and irradiation-induced embrittlement were additive. Kryukov described studies of steel containing 0.02 and 0.2 wt % silicon in which heating to 650°C followed by slow cooling produced increases in T_k of 30 and 60°C, respectively. When these alloys, with and without the special heat treatment, were subsequently irradiated, the total shift in T_k measured after irradiation was larger for the heat treated materials by amounts similar to that caused by the heat-treatment alone. Without disputing their results, the traveler would question the relevance of the thermal treatment used and its resulting effects on microstructure and subsequent irradiation response for RPV applications. Föhl noted that the thermal aging of base metal control specimens from the Gundremmingen reactor on the top of the steam drum for more than 30,000 h resulted in a CVN shift of about 30°C and stated that additional exposures of the archival material have begun.

Relative Shifts of K_{Ic} and CVN

One of the most interesting and important items discussed was the relationship of the irradiation-induced shifts of K_{Ic} and CVN. The traveler presented data from the Fifth and Sixth HSSI Series, concerned with shifts in K_{Ic} and K_{Ia} , respectively, for high-copper welds. The data from the Fifth Series have shown that the shift in the lower bound of the K_{Ic} data exceeds that predicted by the CVN shift. This discrepancy was somewhat greater than but, in qualitative agreement with, the correlation of an extensive amount of CVN vs fracture toughness data from the literature recently performed by Hiser, NRC. Data were presented by Nikolaev on similar studies of the weld metal removed from the Novovoronezh RPV in which shifts in fracture toughness and Charpy impact energy were compared. In contrast to the U.S. experience, the Russian study showed an appreciably

smaller shift ($\sim 90^\circ\text{C}$) in the cleavage fracture toughness, measured at $60\text{ MPa}\sqrt{\text{m}}$ with 0.5TCT specimens, than was seen for CVN specimens at 48 J from the same trepan ($\sim 160^\circ\text{C}$). This was discussed within the group but no explanations were forthcoming as to why such radically different behavior should exist. The relative temperature relationship between the CVN and fracture toughness for the Russian weld, after the 650°C anneal designed to return the steel to its unirradiated condition, appeared to be quite similar to that for unirradiated U.S. steels. For example, the difference in temperature between 48 J of CVN energy and $60\text{ MPa}\sqrt{\text{m}}$ in fracture toughness was about 45 to 50°C in the condition simulating unirradiated steel. By contrast the difference in temperature between the same levels of CVN energy and fracture toughness for the irradiated steel was more than 120°C . Comments by both the Germans and the French indicated that their data show that the shift in the Charpy curves generally bounds the shift in fracture toughness data, if they are both indexed at the "proper" levels, similar to the Russian experience. Neither the French or the Germans had, however, seen such radical differences as the Russians presented, nor could they comment meaningfully on them.

One substantial difference in the U.S. vs foreign data is that only in the HSSI study were thick specimens (up to 100 mm) used. The detailed evaluation of size effects within that study have shown that the temperatures at which the irradiated fracture toughness of 1TCT specimens (25 mm thick) exhibit a toughness of $100\text{ MPa}\sqrt{\text{m}}$ is about 19°C lower than for 4TCT specimens (100 mm thick). It may be worth examining the results of the foreign studies in detail to see if a similar specimen size effect could be leading to nonconservative interpretations of their results. An examination of the size effect question in combination with a detailed comparison of the CVN energy index levels used by different countries could lead to a much better understanding of the apparent discrepancies in K_{Ic} vs CVN relationship.

In subsequent discussions with Nikolaev it was suggested that one useful way to examine the differences between their and our results would be to exchange either fracture mechanics and CVN specimens or material from which to make such specimens and perform comparative interlaboratory testing. To minimize the costs and maximize the comparative information so generated, it was suggested that materials for which good data bases already exist be utilized. Serpan was to check upon his return as to the acceptability of such an exchange.

Regulatory Approaches

Presentations were made which described the German, French, and Russian approaches to vessel protection against brittle fracture. The German approach is very similar to that of the U.S. and relies upon shifting the ASME K_{Ic} or K_{IR} vs temperature curves according to the shift measured in the surveillance Charpy specimens. The fracture resistance curves so obtained are compared against the loadings anticipated to be caused by internal pressure, thermal, and in accident conditions residual stresses. Operational loadings use the standard one-fourth thickness ASME flaw. Accident conditions examine various sizes of flaws with 2:1 aspect ratios. The only significant deviation in the German approach is the use of asymmetric thermal loading from low flow conditions during high pressure emergency core cooling system injections. This has been shown by their experiments and analysis to significantly increase the K_I values for greater times for flaws which could exist in certain locations in the vessel.

The French described their approach of using different sizes of flaws and margins of safety for anticipated accident conditions depending on the likelihood of such a transient occurring. The margins of safety (K_{Ic}/K_I) ranged from 2.5 for transients with anticipated probabilities of occurrence of one per reactor year to 1.25 for probabilities of 10^{-4} . When this is combined with the direct

approach they use for establishing the relationship between K_{Ic} and CVN, the result is a rather simplified screening criteria for their plants. If RT_{NDT} at end of life for a reactor is below 85°C, no mitigation action is required. It falls between 85 and 100°C then rigorous nondestructive examination (NDE) to assure maximum flaw sizes of 6 mm must be performed. If RT_{NDT} will exceed 100°C, which is currently anticipated for only two of their plants, then special actions are required including fluence reduction schemes and heating of safety injection water.

The Russian approach was much harder to quantify. Much was said concerning the quality assurance aspects of vessel manufacture and both initial and in-service flaw detection. The development and use of trend curves to assess radiation damage was described as was fracture determinations from small specimens and engineering-scale structures with and without cladding. Loading conditions evaluated include pressure and thermal shock type loadings and varying safety factors are used in assessing the resistance of the vessel to brittle failure depending on the type of transient being considered. Little, if any, concern was apparent regarding low energy ductile failure, though some of the irradiated Charpy data presented during the meeting exhibited some very low upper-shelf energy levels. Unfortunately, the quantitative details of the approach were not apparent from the presentations. Hopefully, some of the Russian language reports, which we were told did address this question, will provide additional insight into this area when translated.

Additional Areas Covered

A number of additional points were raised during the presentations at the seminar and in their subsequent discussions. The more interesting or important of those points are described below briefly below.

Timofeev, Prometey, described research on FCP behavior of their steels both in simulated PWR water chemistry and after irradiation in air. The Russian data agree well with those of the United States with regard to the effect of a high sulfur content increasing the crack growth rate. His postirradiation tests in air showed no effects on the growth rate when compared to unirradiated material.

Timofeev also briefly described Charpy tests of cladding which had been irradiated at normal operating temperatures to a fluence of 1.3×10^{20} n/cm² ($E > 0.5$ MeV). He stated that they had observed shifts in the transition temperature of the cladding of up to 100°C, substantially greater than ORNL HSSI results. This information was not included in any of their written material and details of it would be worth trying to obtain. Also regarding irradiation effects of cladding, Föhl, described tests of subsize CVN specimens, similar in size to those used for the Greifswald investigations, removed from the Gundremmingen trepans. The cladding had been exposed to a fluence of about 4×10^{18} n/cm² ($E > 1$ MeV). Using correlations obtained for very low upper-shelf ferritic steel (code KS22) from the Integrity of Components Program conducted at MPA, he deduced that the cladding had a postirradiation upper-shelf energy of about 40 J and an unirradiated upper shelf around 90 J. These values are lower and higher, respectively, than those determined in the ORNL HSSI study but agree qualitatively with our results. Moreover, it is very difficult to accurately predict upper shelf energies from subsize specimens even with correlations from comparable materials. To use a correlation derived from ferritic steel to predict behavior of a duplex austenitic/ferritic stainless steel weld is still more difficult.

An animated discussion occurred concerning the use of different weld consumables in the circumferential fabrication welds in the Russian vessels. Apparently the root pass is made with the manual metal arc process and then left in place during the remainder of the welding. The location

of the root was described as being near the inner surface, the middle of the wall, and near the outer surface. In any case the quality of the root pass and its radiation resistance is good, according to Ahlstrand, who has examined Russian welds. In addition, an anticorrosive outer layer, described as a "decorative layer," is also applied to these welds, but it seems to have little structural importance.

In discussions concerning Russian NDE techniques, a comparison of their capabilities with those assessed by the various PISC programs was made. The 100% likelihood of detecting a flaw in a Russian vessel (presumably during in-service inspections) was only approached for flaws with a depth of about 28 mm or greater.

Brillaud from EDF, during discussions on the various methods of correlating K_{Ic} with impact testing results, warned against assuming that all impact tests would produce comparable results. He gave an example that differences in the shape of the tup which actually impacts the specimen can cause significant variations in results. The French tup can produce impact energies which vary by as much as 15 J in the mid-transition region from those determined according to ASTM standards.

MPA had just run their first wide-plate type crack arrest test. The specimen is made of what they call their "limiting material." It is a vanadium-containing rotor steel which has been specially heat treated to give a very high transition temperature ($>200^{\circ}\text{C}$). In the traveler's opinion it does not, however, represent materials of relevance to RPVs, since it is the only steel in which an upper shelf in crack arrest properties has been observed. The specimen was 1500 mm wide, 175 mm thick, and 6.4 m in length including the pull tabs. A temperature gradient from 200°C on the cool side to 300°C on the hot side was established across the width of the specimen, with the flaw in the cooler side. The flaw initiated upon loading, arrested and then reinitiated for a total of five to seven (this is a disputed number) run-arrest events, similar to the HSST WP-2 test series on low upper-shelf material. Crack tip position during the test was obtained using strain gages positioned along the crack path as in the HSST wide plate tests. No brittle crack extension was observed at temperatures in excess of the onset of the CVN upper shelf, contrary to the HSST observations. Analyses of the results was under way during our visit.

Appendix A**Itinerary****1990**

August 28-29	Travel from Knoxville, TN, to Leningrad, USSR
August 30-31	Visit at Prometey Complex, Leningrad, USSR
September 1-2	Weekend and travel to Cologne, FRG
September 3-6	Visit at Gesellschaft für Reactorsicherheit, FRG
September 7	Visit at Materialprüfungsanstalt, Stuttgart, FRG
September 8	Visit at Siemens, A.G., Erlangen, FRG
September 9	Travel from Frankfurt, FRG, to Knoxville, TN

Appendix B

Persons Contacted

From the Prometey Complex in Leningrad, USSR:

Karzov, G. Head of Structural Mechanics Department
 Nikolaev, V. Head of Irradiation Research Department
 Kunilovski, V. Environmental Effects Specialist
 Blumin Irradiation Effects Specialist
 Alekseenko, N. Head of Structural Testing Lab at Gatchina
 Timofeev, B. Strength of Materials Specialist

From the Gesellschaft für Reactorshicherheit (GRS) in Cologne, FRG:

Banaschik, M. Department Head of Research Management
 Schmülling, W. Accidents and Components Research
 Schultz, H. Structural Mechanics Research

From the Materialprüfungsanstalt (MPA), Stuttgart, FRG:

Kussmaul, K. Director, MPA
 Föhl, J. Irradiation Effects Specialist

From Siemens A.G. in Erlangen, FRG:

Tenckhoff, E. Department Head of Structural Integrity Research
 Bartelome, G. Irradiation Effects Specialist
 Erve, M. Structural Integrity Specialist
 Klausnitzer, E. Irradiation Effects Specialist
 Leitz, C. Irradiation Effects Specialist

Other significant personnel contacts at the GRS Seminar in Cologne, FRG:

Pachur, D.	KFA, Jülich, FRG	Lloret, R.	CEA, Grenoble, France
Gerngross, M.	TÜV, Nord, GDR	Amaev, A.	Kurchatov, USSR
Hellwig, K.	SAAS, East Berlin, GDR	Gurovitch, B.	Kurchatov, USSR
Popp, K.	ZFK, Rossendorf, GDR	Kryukov, A.	Kurchatov, USSR
Ahlstrand, R.	IVO, Helsinki, Finland	Tutnov, A.	Kurchatov, USSR
Valo, M.	VTT, Helsinki, Finland		
Brillaud, C.	EDF, France		

Appendix C

Literature Acquired

B. A. Gurovich, "Investigation of the Microstructure and Mechanism of Radiation Embrittlement of Materials for VVER-440 Reactors."

J. Sievers, "Ductile Fracture Mechanical Analyses of Large Scale Thermal Shock Experiments," Submitted to *Nuclear Engineering and Design* (June 1989), revised version (May 1990).

Karl Kussmaul, "West German Research Programs on Irradiation Effects on Reactor Pressure Vessels," *Radiation Embrittlement and Surveillance of Nuclear Reactor Pressure Vessels: An International Study*, ASTM STP 819, L. E. Steele, ed., American Society for Testing and Materials, 1984.

K. Kussmaul, "Radiation Embrittlement and Surveillance of Nuclear Reactor Pressure Vessels: An International Study," *Specific Problems of Reactor Pressure Vessels Related to Irradiation Effects*, ASTM Special Technical Publication 819.

K. Kussmaul, E. Roos, and J. Föhl, "Principles of the German Approach to the Analysis of Flawed Structures," *Int. J. Pres. Ves. & Piping* 25 (1986) 185-215.

K. Jussmaul, J. Föhl, and E. Roos, "Some Conclusions With Regard to the Safety Assessment of Cracked Components Drawn From the Research Programme Integrity of Components (FKS II) at the Present State," *Nuclear Engineering and Design* 102 (1987) 397-418, North-Holland, Amsterdam.

Eberhard Roos and Wolfgang Guth, "Investigations Concerning the Crack Resistance Behavior of Thick-Walled Components Under Thermal Shock Loading," *Materials Technology* 61 (4) (1990).

Eberhard Roos, Thomas Demler, Ulrich Eisele, and Rainer Gillot, "Fracture Mechanics Safety Assessment Based on Mechanics of Materials," *Materials Technology* 61 (4) (1990).

G. Bartholomé, M. Erve, and R. Hertlein, *Safety Analysis of Irradiation Influenced Components*, SIEMENS AG-KWU, Erlangen, FRG.

C. Leitz, D. Pastor, and M. Valo, "Recovery Annealing and Subsize Specimen Testing for Units I and II of NPP Nord (Greifswald)," Seminar über Strahlenbeeinflussung von Reaktordruckbehältern GRS Köln, 3. 7. 9. 1990.

K. Kussmaul, J. Föhl, and T. Weissenberg, "Investigation of Materials from Decommissioned Reactor Pressure Vessel - a Contribution to the Understanding of Irradiation Embrittlement."

C. Brillaud (E.D.F.-S.P.T.-Groupe des Laboratoires) and F. Hédin (E.D.F.-SEPTEN), Evaluation Du Comportement Sous Irradiation Des Aciers De Cuve," *Colloque International, International Symposium FONTEVRAUD II, September 10-14, 1990*.

R. Lloret, A. Bévilacqua, J. J. Bonnin, J. C. Nimal, L. Bourdet, F. Hédin, and J. C. Guilleret, "Determination of the Doses Absorbed by the Chooz A PWR Vessel. Summary and Conclusions of Investigations."

F. F. Giginyak, Yu. I. Zvezdin, A. A. Lebedev, and B. T. Timofeev, "Mechanical Strength of Large Pressure-Vessel Materials in Corrosive Medium."

V. A. Nikolaev, "Radiation Embrittlement of Reactor Vessel Steels Irradiated in Research Reactors."

V. V. Rybin, V. A. Nikolaev, A. M. Morozov, L. A. Veiner, and S. A. Karpi, "Fracture Strength of Reactor Vessel Material of Unit No. 1 of NOVOVORONEZHSKAYA NPP."

B. A. Ignatov, V. A. Nikolaev, V. I. Badanin, and L. G. Egorov, "Radiation Damage Annealing in Reactor Vessel Material of Unit No. 1 of NOVOVORONEZHSKAYA NPP."

Victor A. Ignatov, Georg P. Karzov, and Boris T. Timofeev, "The Main Principles and Research Ways to Base Strength of Light Water Reactor Vessels in the USSR."

K. Wallin, K. Törrönen, Technical Research Centre of Finland VTT, Espoo, Finland, R. Ahlstrand, Imatra Power Co., Vantaa, Finland, B. T. Timofeev, V. Rybin, V. Nikolaev, and A. Morozov, Metallurgical Institute AEE, Leningrad, USSR, "Theory Based Statistical Interpretation of Fracture Toughness of Reactor Pressure Vessel Steel 15x2MφA and its Welds."

Christian Brillaud and Francois Hedin, "In-Service Evaluation of French Pressurized Water Reactor Vessel Steel."

J. C. Nimal, L. Bourdet, T. Vergnaud, R. Lloret, A. Bevilacqua, F. Hedin, and J. C. Lefebvre, "Determination of Neutronic Characteristics for the French Nuclear Plant Unit Surveillance Programme 900 MWe PWR."

In addition, a number of Russian language reports were obtained which will need to be translated to be of significant use.

Appendix D

Distribution

1. John J. Easton, Jr., Assistant Secretary for International Affairs and Energy Emergencies (IE-1), DOE, Washington, DC 20585
2. J. M. Taylor, Executive Director for Operations, NRC, Washington, DC 20555
3. E. S. Beckjord, Director, Office of Nuclear Regulatory Research, NRC, Washington, DC 20555
4. L. C. Shao, Director, Division of Engineering, NRC, Washington, DC 20555
5. E. J. McCallum, Director, Division of Safeguards and Security, DOE, Washington, DC 20585
6. A. Bryan Siebert, Director, Office of Classification and Technology Policy, DOE, Washington, DC 20585
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- 11-12. A. Taboada, Division of Engineering, NRC, Washington, DC 20555
- 13-14. C. Z. Serpan, Jr., Division of Engineering, NRC, Washington, DC 20555
- 15-16. P. N. Randall, Division of Engineering, NRC, Washington, DC 20555
- 17-18. R. J. Bosnak, Deputy Director, Division of Engineering, NRC, Washington, DC 20555
- 19-20. D. J. Cook, Director, Safeguards and Security Division, DOE/ORO
21. T. E. Murley, Director, Office of Nuclear Reactor Regulation, NRC, Washington, DC 20555
22. T. P. Spies, Deputy Director, Office of Nuclear Regulatory Research, NRC, Washington, DC 20555
23. E. C. Shomaker, International Programs Office, NRC, Washington, DC 20555
24. G. Fowler, International Programs Office, NRC, Washington, DC 20555
25. Executive Secretary, Advisory Committee on Reactor Safeguards, NRC, Washington, DC 20555
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29. G. R. Odette, Department of Chemical and Nuclear Engineering, University of California, Ward Memorial Drive, Santa Barbara, CA 93106
30. L. Greenwood, Battelle-Pacific Northwest Laboratories, MS E7-22, P.O. Box 999, Richland, WA 99352
31. S. T. Rosinski, Sandia National Laboratories, Division 6521, Albuquerque, NM 87185
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33. D. J. McGoff, Associate Deputy Assistant Secretary for Reactor Deployment, DOE, NE40, A439 1GTN, Washington, DC 20585
34. E. Purvis, DOE, Washington, DC 20585
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- 36-37. Office of Scientific and Technical Information, P.O. Box 62, Oak Ridge, TN 37831

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