NOV 07 1990

Contract Program or Project Title: Heavy-Section Steel Technology (HSST) Program

Subject of this Document: Report of Foreign Travel of T. L. Dickson, Engineering Technology Division

Type of Document: ORNL Foreign Trip Report

Author: T. L. Dickson

Date of Document: October 22, 1990

Responsible NRC Individual and NRC Office or Division: M. E. Mayfield (FTS: 492-3844)
Division of Engineering, U.S. Nuclear Regulatory Commission

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Prepared for the
U.S. Nuclear Regulatory Commission
Washington, DC 20555
Under Interagency Agreement DOE 1886-8011-9B
NRC FIN No. B0119

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FOREIGN TRIP REPORT

ORNL/FTR-3771

DATE: October 22, 1990

SUBJECT: Report of Foreign Travel of T. L. Dickson, Engineering Technology Division

TO: Alvin W. Trivelpiece

FROM: T. L. Dickson

PURPOSE: To participate in the 16th Staatliche Materialprüfungsanstalt (MPA) Seminar on Safety and Reliability of Plant Technology with Special Emphasis on Nuclear Technology.

SITES VISITED:	10/4-5/90	16th MPA Seminar	Universität Stuttgart Stuttgart, FRG	Dr. Gillot Participants
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ABSTRACT: The traveler attended the 16th MPA Seminar on the Safety and Reliability of Plant Technology with Special Emphasis on Nuclear Technology. The objective of the trip was to gather information and data that could prove useful to the U.S. Nuclear Regulatory Commission (USNRC) sponsored Heavy-Section Steel Irradiation (HSSI) and Heavy-Section Steel Technology (HSST) Programs and to present a paper entitled, *Effects of Irradiation on Initiation and Crack-Arrest Toughness of Two High-Copper Welds and on Stainless Steel Cladding*. This paper summarizes results from the 5th, 6th, and 7th Irradiation Series of experiments performed within the HSSI Program by the Metals and Ceramics Division at Oak Ridge National Laboratory (ORNL).

INTRODUCTION

A variety of international papers were presented at the 16th MPA Seminar on Safety and Reliability of Plant Technology with Special Emphasis on Nuclear Technology. Major categories were: (1) Fracture Mechanics, (2) Nondestructive Testing, (3) Fatigue/Creep Processes, (4) Irradiation Embrittlement, (5) Thermal Loading, and (6) Integrity of Vessels and Components. A considerable amount of information was presented that is quite relevant to ongoing work within the HSSI and HSST Programs.

There were several informative papers presented in each of these major categories. Categories (1), (4), (5), and (6) are relevant to the USNRC-sponsored work at ORNL within the HSSI and HSST Programs. The conference provided an excellent format for formal as well as informal discussions between international specialists presently engaged in ongoing research programs with the primary common objective of ensuring the safe,

continued operation of nuclear power stations, primarily from a structural mechanics viewpoint.

Some of the presentations which were particularly relevant to research currently being performed within the HSST and HSSI Programs were as follows:

1. A Japanese paper was presented ("Ductile Crack Growth Under a PTS Event") which described Step B of the Japanese project, a Fundamental Study of Elastic-Plastic Fracture Mechanics. The test was conducted on a flat plate specimen simultaneously subjected to tensile, bending, and thermal-shock loads. The analytical models (3-D finite-element thermal analysis, 3-D thermo-elastic-plastic stress analysis, and 3-D fracture mechanics analysis which used the J-integral as the fracture parameter) conservatively predicted a stable crack growth of 2.5 mm at the deepest point compared to the observed value of 0.9 mm.

The particular study is part of the ongoing Japanese PTS integrity studies [initiated in 1983 as a national project by Japan Power Engineering and Inspection Corporation (JAPEIC)] which has the stated objective of gaining public acceptance for RPV integrity against PTS events and to develop a data base for plant life extension. During the period of these studies, a number of interactions between JAPEIC and the HSST staff have occurred concerning PTS experiments. In the most recent of these interchanges, JAPEIC provided the HSST Program with a problem statement on the Japanese Step B experiment for inclusion in the recently completed CSNI Project FALSIRE.

2. A Finnish paper was presented ("Experience of Irradiation Embrittlement in the Loviisa Reactor Pressure Vessels") which described that the irradiation embrittlement measured from surveillance specimens (core weld specimens) taken from the Russian designed and fabricated vessels was considerably higher than expected. The anticipated shift in the transition temperature of the specimens, based on design information, was 80°C compared to the measured 130°C. These results were the first ones obtained from a surveillance program of a VVER-440 Reactor Pressure Vessel. Based on these results, the Finnish authorities have taken measures to reduce the neutron flux and also to reduce the magnitude of thermal stresses induced in a PTS event. The Finnish authorities permitted an operating license for the Loviisa plant until 1992 based on deterministic and probabilistic fracture-mechanics analyses. These analyses were performed using the ORNL-developed OCA-P fracture-mechanics code. Current efforts to extend the life of the license of the Loviisa plant are primarily concentrated on analysis of the influence of cladding on fracture initiation and a cooperative Finnish/German/Russian materials research effort.

The Finnish authorities have concluded that thermal annealing is also a necessary measure for the older Finnish VVER-440 plants. Based on the results of thermal annealing of nine RPVs in the USSR and other Comecon countries, the Finnish authorities are optimistic that thermal annealing and flux-reduction programs can extend the life of these plants beyond the design life.

A Finnish delegation of specialists is scheduled to participate in a joint Seminar on the Structural Safety Aspects of Nuclear Power Plant Components in Oak Ridge with the HSST and HSSI staff in December 1990.

3. A German paper ("Numerical Analysis of Ductile Crack Growth in a Simplified Nozzle Model Under Pressurized Thermoshock Loading") was presented which stated that a 2-D model produces acceptable (slightly conservative) results of elastic plastic

fracture-mechanics analyses even when considering thermal stratification. This paper is of particular interest as the HSST Program is currently engaged in evaluating the effects of thermal stratification on PTS analysis of reactor pressure vessels.

The conference included a tour of experimental facilities at the University of Stuttgart (Materialprüfungsanstalt). The facilities appear to be modern, state-of-the-art, with an impressive range of experimental material testing capabilities, and very professionally staffed.

SUMMARY OF DISCUSSIONS

Several informal discussions were held with participants from Germany, England, and Japan. Many conference participants were quite interested in the results of the paper which the traveler presented, particularly with regard to the quantity and size of the fracture toughness specimens which were subjected to irradiation.

The traveler engaged in two discussions that were particularly relevant to work the HSST Program is presently actively involved in. The details of these are below.

Probabilistic Fracture-Mechanics Code Mutual Validation (Yankee Rowe Nuclear Power Station Review)

The first meeting was an extended discussion with Jack Strosnider, formerly of the USNRC and presently located in Paris, France, on an assignment with the Organization for Economic Cooperation and Development (OECD). The traveler was aware that Mr. Strosnider had been an active participant in the initial development of the probabilistic fracture-mechanics approach to the Pressurized-Thermal-Shock (PTS) issue. Mr. Strosnider was heavily involved in the development of the probabilistic portion of the VISA code in the early 1980s. The VISA code evolved, through work performed by the USNRC and Pacific Northwest Laboratory, into what is presently known as the VISA II code. The OCA-P specialized probabilistic fracture-mechanics code was independently developed at ORNL and was an instrumental tool in performing the Integrated Pressurized-Thermal-Shock (IPTS) studies. Regulatory Guide 1.154 specifies these two codes as acceptable for performing the probabilistic fracture-mechanics portion of a plant-specific safety analysis required of any nuclear plant desiring to operate beyond the PTS screening criteria.

The HSST Program has recently been heavily involved in the review of the Reactor Pressure Vessel Evaluation Report for the Yankee Rowe Nuclear Power Station. The probabilistic fracture-mechanics analyses for the Yankee Rowe Report were performed with the VISA II code. Part of the Yankee Rowe Nuclear Power Station Review performed by the HSST Program has been an attempt to mutually validate these two independently developed codes. While the two codes were successfully validated for deterministic solutions, initial attempts at mutual validation of the probabilistic solutions were not successful. After performing a thorough examination of the two codes, three modifications, coordinated with Pacific Northwest Laboratory personnel, were made to the VISA II code. Two of the code modifications were relatively minor; however, the third was particularly significant and focuses on the philosophy used to perform probabilistic fracture-mechanics analyses. After making the code modifications to VISA II, the probabilistic solutions of VISA II and OCA-P appear to mutually validate one another.

The traveler and Mr. Strosnider discussed these code modifications in a considerable amount of detail. Mr. Strosnider was quite interested in the code comparisons and seemed to be in basic agreement with the code modifications proposed for VISA II which appear to bring the probabilistic solutions of VISA II into relatively close agreement with those of OCA-P.

Impact of Crack-Arrest Prediction on PTS Analysis

The second particularly relevant discussion was with Dr. E. Smith of the University of Manchester/UK regarding a paper which he presented entitled "Predicting Crack Arrest in Reactor Pressure Vessels." The work was sponsored by EPRI and coauthored by T. J. Griesbach. The thesis of this paper is that the ASME procedure (and the procedure utilized in the OCA-P code) for predicting crack arrest, which is based on "static" linear-elastic fracture-mechanics (LEFM) principles, is overly conservative when it is applied to the arrest of a propagating crack in a reactor pressure vessel. The paper claims that the application of a more recent (and less conservative) method of predicting crack arrest, i.e., the Combustion Engineering constrained static procedure which considers dynamic effects, could potentially impact the screening criteria limits in the PTS rule. The traveler informed Mr. Smith that a recent study performed at ORNL had concluded that enhanced crack-arrest potential (based on crack-arrest data derived from HSST large-specimen fracture-mechanics experiments) does not appear to have the potential to significantly impact the results of PTS analyses. The results of this study will soon be published/presented at the 18th Water Reactor Safety Information Meeting. It should be noted that HSST personnel are expecting to meet with specialists from Combustion Engineering in the near future to discuss this "less conservative" method of predicting crack arrest and its potential impact on PTS analyses.

APPENDIX A**Itinerary**

October 2-3	Travel	Oak Ridge, TN to Stuttgart, FRG
October 4-5	Meeting	16th MPA Seminar, Stuttgart, FRG
October 6	Travel	Stuttgart, FRG to Oak Ridge, TN

APPENDIX B

Literature Acquired

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13. D. Möller, K. Maile, and W. Eckert, "Studies on Finite-Element-Calculations of Elbows in the Creep Regime," pp. 14.1–14.23 in *Proceedings of 16th Staatliche Materialprüfungsanstalt (MPA) Seminar*, Vol. 1, October 4-5, 1990.*
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*Requires German to English translation.

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