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Summary

This conference is the third in a series of conferences held approximately every three years on the subject of radiation embrittlement of reactor pressure vessel steels. As with prior meetings, emphasis was on the understanding of radiation embrittlement of steels and its interpretation to better assure the structural integrity of the pressure vessels in operating nuclear power reactors. The present meeting built upon the status of the technology at the time of the previous meetings and reflected changes that are taking place in the nuclear power industry. In particular, there is a growing emphasis on plant-life extension, fundamental mechanisms of radiation damage, as well as vessel aging aspects. These emphases were evident in the contents of the meeting's formal papers and related discussion.[1]

Thirty-nine specialists from 13 countries presented 21 papers in the two-day meeting. Discussion was encouraged and became a critical part of the meeting's results. Besides a special discussion that contributed to the recommendations, there were four technical sessions: (1) Overview of National Programs; (2) Surveillance Program Results; (3) Research to Support Vessel Integrity Assurance; and (4) Fundamental Damage Mechanisms. The four formal sessions are reviewed briefly below and general conclusions are given.

Overview of National Programs

National programs aimed at assessing irradiation effects on pressure vessel steels were summarized by specialists in Switzerland, the Federal Republic of Germany, the United States, and the United Kingdom. The research priorities, regulatory procedures and changes, and data generation were described.

Program priorities are reasonably consistent among the reporting countries, but the level of funding and the number of issues being addressed differ substantially. The priorities are, of course, driven most of all by the need to assure vessel integrity in operating and planned plants. One of the most prominent areas is the generation of extended ASTM-valid fracture-toughness data bases that include the effects of irradiation on fracture toughness at higher temperatures. Activities in this area were cited by the Swiss, FRG, and U.S. programs. An adjunct to this is the continued interest in relating irradiation effects observed in surveillance specimens to effects on fracture-toughness properties; this activity is most important inasmuch as surveillance specimens are most often Charpy-impact specimens and valid fracture-toughness determinations for temperatures much above the RT_{NDT} require large compact or bend specimens. Continued priority is also on validation of dosimetry techniques through benchmark experiments and standardization of methods. Spectrum and dose-rate effects continue to be concerns in developing correlations for high-fluence regions; however, increased interest is also developing for understanding irradiation effects on materials in regions of low exposure and low temperatures. While considerable progress has been made in recent years in understanding the sensitivity of radiation

[1] This summary reflects papers presented at the conference. One paper presented was omitted in the book and one paper not presented was included.

age to copper and nickel contents, the influence of other compositional elements continues to be pursued. One additional area, where priority is increasing, is in seeking an understanding of thermal-annealing effects in irradiation-anneal-reirradiation scenarios, because of the increased overall priority that is developing for plant-life extension criteria, especially in the United States.

The approach to applying the observed irradiation effects in vessel safety assessments remains mostly linked to the K_{IR} curve concept that prescribes the RT_{NDT} as a function of exposure and compositional factors. The USNRC is continuing to examine trend curves in their *Regulatory Guide 1.99,* Revision 2. The considerations include the impacts that the revisions have on probabilistic risk assessments (PRA) and pressure-temperature limits for operating reactors. Of course, revised correlations impact the results from the application of regulatory rules such as the USNRC screening rule for pressurized thermal shock. However, through the PRA analyses, they can also have impacts on the prescription of the rules themselves. Therefore, the total evaluation of *Regulatory Guide 1.99,* Revision 2, is continuing. Of course, plans for new plants are reaping the benefits of the tremendous progress made over the past several years. For example, the chemistry limits incorporated in the United Kingdom's Sizewell B pressure vessel material specifications reflect the current knowledge of compositional influence.

Generation of properties data continues on two levels. The first level is to extend the data base for improved correlation development. This includes the validation of the applicability of research reactor data to correlations based on surveillance data, as well as extending the range of fracture-toughness measurements obtained from research reactors to higher temperatures (larger irradiated specimen testing). The second level deals with research data aimed at generating more fundamental understanding to better correlate observations made from various types of tests (for example, temperature shifts and shape changes observed from Charpy impact tests and fracture-toughness tests) and to modeling mechanisms that allow the transfer of data and criteria obtained from relatively short-term high-dose-rate research reactor circumstances to long-term operating reactor conditions. Among other reasons, understanding dose-rate effects is very important to bring about the inclusion of high-flux research data into the formulation of irradiation-damage correlations. Additionally, mechanism studies are an important part of the annealing studies to enhance predictions of reembrittlement rates. The IAEA Coordinated Research Program (CRP) is providing continuity to these efforts through the use of reference test materials, as well as through the sharing of data and conclusions.

Surveillance Program Results

This session of the meeting addressed research studies and surveillance programs associated with Yugoslavia's Krsko reactor, the United States High Flux Isotope (HFIR) Reactor, and Finland's contribution to the IAEA-CRP. Both the Krsko and HFIR studies showed irradiation-induced degradation that exceeded earlier expectations for what were expected to be low property-shift conditions. These studies employed A533 grade B class 1 and A212 grade B steels, respectively. The Yugoslavian study was for a low-copper material with a relatively low exposure. The HFIR study was for long-term (17.5 EFPY) low-flux exposure of vessel materials at low temperatures (~50°C). In addition to the HFIR vessel materials showing more degradation than expected at the time of vessel design, the Charpy-energy shifts of surveillance specimens occurred with a neutron fluence about ten times lower than that required to achieve the same shift when exposed in the Oak Ridge Research Reactor (ORR) at a much higher dose rate. These observations support the increased interest in developing data and validating the applicability of trend curves, for example, USNRC *Regulatory Guide 1.99,* Revision 2, to relative low-damage regimes. The

United Kingdom has not adopted *Regulatory Guide 1.99*, Revision 2, because they believe its applicability to modern steels (for example, those with low-copper contents) has not been fully established. This is an area recommended for further coordinated research. Maximum attention has historically been on the beltline regions of operating reactors. These observations are now drawing attention to other components as well, including core structures and vessel supports.

Finland's work in the IAEA-CRP involves using a variety of specimens to examine specimen/size effects in determining fracture-toughness (K_{Ic}) values at temperatures into the Charpy transition range. The specimen sizes range from Charpy impact to 4TCT specimens and are made of IAEA reference materials. Statistical size-effect adjustments are reported to be appropriate. Though the activities were not reported in detail in this session, the USNRC research program is also irradiating two high-copper welds with specimens that range from Charpy to 4TCT (control unirradiated specimens to 8TCT). It is recommended that the results of these studies be coordinated and interpreted in terms of size and loading-rate adjustment factors that have been explored for results from fracture tests of unirradiated materials.

Research to Support Vessel Integrity Assurance

Papers presented in this session covered a wide range of topics that included methods of design and material specification to minimize irradiation effects (for example, design measures, material specification measures, and flux reduction measures), properties trend curves, and evaluations of thermal annealing on irradiation effects in pressure vessel steels. The papers individually address the specific issues and details. Efforts to provide maximum assurance of reactor pressure vessel integrity need to include all these considerations. That is to say, use of the most improved dosimetry and properties models for representing irradiation effects will of course contribute to the best available prediction of allowable pressure vessel lifetimes. Core configuration and other factors influencing the neutron flux at the reactor wall and careful material specification are two principal contributors to assuring long allowable lifetimes for new reactors. Of course, flux abatement is a crucial step to extending the remaining life of the pressure vessels in operating reactors. Finally, given that a vessel is nearing the end of allowable life due to the level of neutron exposure, there is great economic incentive to explore methods for life extension. Thermal annealing is one area that holds much promise. Over the past decade, considerable understanding has been developed through the use of small specimens under irradiation/thermal-anneal/reirradiation studies. From an engineering point of view at least two major questions remain. First, do the observations on annealing effects made from small specimens carry over to thick-section specimens exposed to the same conditions? Second, what path needs to be followed to validate in-situ thermal annealing influences and to demonstrate the engineering feasibility of in-situ thermally annealing of full-scale reactor pressure vessels? These assurances and demonstrations will require a considerable amount of time and effort to complete; therefore, it is recommended that each be pursued to have methods available in a timely manner. Another detailed recommendation made at the meeting was that the IAEA include crack-arrest specimens in the CRP.

Fundamental Damage Mechanisms

The three papers in this session addressed mechanistic aspects of annealing and irradiation damage interactions, small angle neutron scattering analyses, and positron annihilation analyses. These analytical papers suggest that irradiation and annealing can be qualitatively explained by the precipitation behavior of copper in the ferritic-iron matrix. They have shown that irradiation hardening occurs because of formation of small copper precipitates and that annealing of irradiated material leads to coarsening of larger particles at the expense of smaller particles. Reirradiation hardening occurs by further precipitation and formation of additional small particles. Positron lifetime measures have been found to correlate well with alloy composition. Similarly, the positron annihilation parameters have been correlated with irradiation-induced hardening variations in forged materials. These types of correlations are promising building blocks for future methods of transferring observations of laboratory and research reactor studies to long-time operating reactor vessel conditions. Additionally, these correlations hold a great deal of promise for more effective utilization of very scarce surveillance material. That is, nondestructive measurements may be possible for material that can be reinserted for further irradiation or measurements may be made from remaining pieces of material specimens tested for other purposes.

General Conclusions

The enthusiasm and extent of ongoing work reported at the meeting speaks well for the future expansion of our understanding of radiation embrittlement of steels. Further, it reflects a global view of active positivism for the material sciences and technologies that support nuclear power. The recommendations cited above are believed to be a reflection of consensus opinions on the topics. Strong support continues for research and development in these subjects, and the rate of progress is gratifying. A further consensus is that the IAEA provides a very valuable international outlet for information on the specific topic of irradiation embrittlement of reactor pressure vessel steels and, accordingly, that the sponsoring International Working Group should continue to serve this function and to support the area through the Coordinated Research Program. It was generally recommended that the frequency of these international meetings should be maintained.

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