U.S. Nuclear Regulatory Commission Research Conference 2003

Panel Discussion on

Materials Degradation- Present Status and Future Directions

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Presentation by Paul Gunter, Director

Reactor Watchdog Project

Nuclear Information and Resource Service

1424 16th Street NW Suite 404

Washington, DC 20036

Tel. 202 328 0002

http://www.nirs.org

NRC Objective: To describe the results characterizing degradation mechanism in reactor coolant pressure boundary materials, methodologies for mitigation and repair, and actions planned for evaluating and managing such degradation.

NRC areas topical to discussion:

-Analysis of structural integrity of exposed clad in Davis Besse corrosion cavity

-Development of a susceptibility model for predicting CRDM degradation

-Development of technologies for inspecting replacement heads

-NRC Industry collaboration on problems of degradation

Thank you for providing this opportunity to present some aspects of the public's perceptions on the current approach to the evaluation and management of the aging and deterioration of the pressure boundary at U.S nuclear power stations.

A number of recent surprises and unanticipated deterioration impacting the all important primary pressure boundary have shocked industry, regulator and public alike. Moreover, let us remember that the industry the regulatory are measured by the weakest link as the potential consequences of failure are unacceptable.

Industry noncompliance with nuclear power plant licensing requirements and the failure to provide complete and accurate information for maintaining the pressure boundary combined with the lack of NRC oversight and enforcement as most recently revealed in yesterday's Office of the Inspector General Report present a significant threat to public health and safety, particularly in an aging and bottom line focused nuclear power industry faced with an increasingly competitive electricity market.

Davis-Besse has set the current low water mark for the disregard of licensing agreements where production margins were allowed to override safety margins.

As worrisome is an apparent lack of consensus regarding analysis of the remaining structural integrity of the exposed vessel cladding material in the Davis Besse corrosion cavity and the associated risk in Core Damage Frequency.

According to reports by Structural Integrity Associates (SIA) and BWX Technologies both commissioned by FirstEnergy, there was no real concern as "significant margins" existed against vessel rupture even in the thinnest part of stainless steel inner liner of Davis-Besse's reactor pressure vessel despite the evidence of its deformation and cracking. According to SIA, the maximum pressure for the cladding without loss of pressure boundary function was 5600 psi and vessel rupture was therefore considered not a credible event. FENOC management went so far as to tell NRC that "a revision to the safety significance assessment is not warranted."

We now understand that a significantly lower cladding rupture point of 1900 psi obtained by a government laboratory mock up burst test is anticipated to be reported in the upcoming Accident Sequence Precursor Events Report to Congress.

The disparity between these two values is alarming.

With Davis-Besse's operational pressure of 2230 psi, clearly, the lower value of 1900 psi represents a significantly higher risk in terms of Davis Besse's Core Damage Frequency for the April 2000 restart from its 12th refueling through the subsequent NRC decision more than a year later to allow Davis-Besse to operate beyond the advised December 31, 2001 shutdown and inspection date per Bulletin 2001-01 to its shutdown in February 2002.

The lower value raises the significant and ongoing question regarding FirstEnergy's compliance to 10 CFR 50.9 to provide information that is "complete and accurate in all material aspects" in characterizing the degradation mechanism and the associated risk of failure of the Davis-Besse reactor coolant pressure boundary material.

While the focus of this discussion is on reactor pressure vessel materials degradation I want to address some concerns regarding the evaluation and management of the deterioration of the equally important steam generators. Steam generator tubing in service today is by and large fabricated of the same susceptible material as of concern in the vessel penetrations sleeves and nozzles, Alloy 600.

Steam generator tubing constitutes <u>more</u> <u>than</u> 50% of the high pressure boundary at 69 U.S. reactors and must serve not only as the effective heat transfer system for electricity production but more vitally as a barrier to protect the environment and the public health from the radioactivity contained within the steam generators and the reactor's primary pressure system loop. This portion of the pressure boundary has no backup. If breached it would allow direct radioactive releases to the environment. Originally designed to last the forty-year license, the structural integrity of steam generators has fallen far short with entire replacements occurring as early as 10 years.

More worrisome is NRC's adoption of an alternative to allow pressurized water reactors to operate with cracked tubes irrespective of crack penetration depths. Permission to operate with defective tubes relies on two increasingly dubious assumptions:

1) There is a valid correlation between eddy current voltage readings and potential tube leakage and break. It is our understanding gained from Differing Professional Opinion insights that a large number of surface cracks will produce a large voltage reading while a single through-wall crack will produce a small voltage reading. The single through wall crack is allowed to remain in service even though it is more prone to leak and break than the high voltage surface cracks;

2) Voltage readings at the beginning of a fuel cycle together with statistical data on voltage growth rates from previous cycles can be used to predict tube integrity at the end of the next operating cycle. Despite many years of industrial experience with stress corrosion cracking, the degradation mechanism remains a mystery and unpredictable from one inspection cycle to the next. It was Consolidated Edison reliance upon plant historical data to defer steam generator tube inspections the summer of 1999 and NRC relaxation of the licensing agreement that resulted in the February 15, 2000 steam tube rupture at Indian Point just over seven months later.

In spite of the fact that adequate safety standards for operating steam generators with defective tubes do not exist, the NRC allows pressurized water reactors to continue to operate with cracking steam generator tubes and emerging NRC guidelines to allow operation with through wall cracks. It is increasingly dangerous for NRC and the nuclear industry to place production agendas over public safety considerations by allowing operation with cracked tubes.

The lack of NRC staff and industry understanding of crack growth rates is extremely disturbing given that they also do not have a technically defensible analysis of how flawed steam generator tubes will behave under severe accident conditions where the reactor coolant remains pressurized.

The public health and safety interest community remains concerned that economics continues to drive the inspection and maintenance of the pressure boundary system, particularly the miles and miles of steam generator tubing contained within a typical Pressurized Water Reactor.

As pressure boundaries in PWRs continue to age and deteriorate there is an increasing need for more intrusive inspection, more conservative guidelines and more rigorous regulation and enforcement that favor public safety over industry economic and regulatory burden considerations. Not the reverse. The methodology for inspection and mitigation should clearly include the increased use of the best available technology for detection as a more conservative approach to mitigation.

In our view and in conclusion, such a conservative safety effort would require that:

-All steam generator tubes would be examined with both the bobbin coil and the rotating coil throughout the entire length of the tubes. Currently, the rotating coil is used at selected locations only.

-All tubes with crack indications would be plugged.

-The number of degraded tube removals would be increased for independent laboratory examination.

-Susceptible areas in steam generators would be examined more frequently with the rotating coil by way of additional outages to the mid-cycle and refueling outages.

-All loose parts would be closely monitored and removed with each outage.