UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

May 19, 1995

NRC GENERIC LETTER 92-01, REVISION 1, SUPPLEMENT 1: REACTOR VESSEL STRUCTURAL INTEGRITY

Addressees

All holders of operating licenses (except those licenses that have been amended to possession-only status) or construction permits for nuclear power reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this supplement to Generic Letter (GL) 92-01, Revision 1, to require that all addressees identify, collect and report any new data pertinent to analysis of structural integrity of their reactor pressure vessels (RPVs) and to assess the impact of that data on their RPV integrity analyses relative to the requirements of Section 50.60 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.60), 10 CFR 50.61, Appendices G and H to 10 CFR Part 50, (which encompass pressurized thermal shock (PTS) and upper shelf energy (USE) evaluations) and any potential impact on low temperature overpressure (LTOP) limits or pressure-temperature (P-T) limits.

Background

The staff issued GL 92-01, Revision 1, "Reactor Vessel Structural Integrity," on March 6, 1992, to obtain information necessary to assess compliance with requirements regarding RPV integrity in view of certain concerns raised in its review of RPV integrity for the Yankee Nuclear Power Station. All licensees submitted the information requested by July 2, 1992. Following receipt and review of licensee supplements responding to requests for additional information, the staff completed its review of licensee responses to GL 92-01, Revision 1, in the fall of 1994. The staff issued NUREG 1511, "Reactor Vessel Status Report," summarizing key aspects of the work in December 1994 [Ref. 1].

The staff has recently reviewed data relevant to the PTS evaluations of several plants. These reviews showed that licensees may not have considered all pertinent data in their responses to GL 92-01, Revision 1, or in their RPV integrity evaluations. It has now become apparent to the staff that no single organization has all the data relevant to RPV integrity evaluations. A major complicating element in this regard is that proprietary considerations have inhibited effective sharing of information.

It has been demonstrated that some RPV integrity evaluations are very sensitive to consideration of new data. For example, under certain conditions, changing the mean copper content for the limiting vessel beltline material by a few hundredths weight percent can change the predicted date for reaching the PTS screening criteria of 10 CFR 50.61 by several years. In addition, changes in estimates of mean copper content can affect the validity of PTS evaluations based on surveillance data. The staff will be considering the impact of these findings in plant-specific evaluations and in its longer-term reassessment of 10 CFR 50.61. PTS is a concern only for pressurized water reactors (PWRs) because boiling water reactors (BWRs) operate with a large inventory of water at saturated steam conditions and, therefore, are not subject to PTS.

However, in addition to concerns regarding PTS evaluations, consideration of additional, unreviewed RPV data can also affect evaluations for USE, P-T limits, and LTOP limits. These evaluations pertain to both PWRs and BWRs, except for LTOP limits, which apply only to PWRs. The staff recognizes that addressees have previously submitted data pertinent to these evaluations as required by the regulations and in responses to GL 92-01, Revision 1, and GL 88-11.

Based on currently available information, the staff believes that the near- term focus for RPV integrity will be the Palisades RPV which is predicted to reach the PTS screening criteria by late 1999, before any other plant. However, because of the importance of RPV integrity and the potential impact of additional, unreviewed data on existing RPV evaluations, the staff believes that this issue needs to be resolved on an expedited basis. Although the issues raised in this GL supplement were highlighted by concerns pertaining to PTS analyses, licensees should consider the effect of the reexamination of RPV data on all aspects of RPV structural integrity.

Regulatory Requirements

As required by 10 CFR 50.60(a), licensees for all light water nuclear power reactors must meet fracture toughness requirements and maintain a material surveillance program for the reactor coolant pressure boundary. These requirements are set forth in Appendices G and H to 10 CFR Part 50. 10 CFR 50.60(b) provides that proposed alternatives to the requirements of Appendices G and H to 10 CFR Part 50 may be used when an exemption is granted under 10 CFR 50.12. 10 CFR 50.61 provides fracture toughness requirements for protecting PWRs against PTS events. Licensees and permit holders have also made commitments in response to GL 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," to use the methodology in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," to predict the effects of irradiation as required by Paragraph V.A of Appendix G to 10 CFR Part 50.

Discussion

The staff focused its examination of the GL 92-01, Revision 1, data and other docketed information on the two key aspects of RPV structural integrity of primary concern to the NRC: PTS and USE. With respect to USE, licensees of all plants were able to demonstrate compliance with the Appendix G requirements either through consideration of applicable data or through equivalent margins analyses. With regard to PTS, only two plants (Beaver Valley 1 and Palisades) were projected to exceed the PTS screening criteria of 10 CFR 50.61 before the end of operating life (EOL). As stated previously, based on data and analyses submitted for GL 92-01, Revision 1, and other recent reviews (e.g, Ref. 2), the staff has determined that not all licensees were aware of all the information pertinent to the analysis of the structural integrity of their RPVs. In addition, recent reviews have indicated larger-than-expected variabilities in weld chemical composition, which have, in turn, highlighted the extreme sensitivity of RPV embrittlement estimates to small changes in the chemical composition of beltline materials.

Recent NRC Staff Evaluations of RPV Structural Integrity Data for PTS Events

The staff issued a safety evaluation report to the licensee for Palisades on the variability of reactor vessel weld properties for the Palisades reactor vessel on April 12, 1995 [Ref. 2]. The staff agreed with the licensee's best-estimate analysis of the chemical composition of the reactor vessel welds and concluded that continued operation through Cycle 14 (late 1999) was acceptable. As discussed previously, while performing the evaluation, the staff noted larger variability in the chemical composition of the welds compared to that assumed for the development of the PTS rule. The staff evaluated the implications of this larger variability on the PTS rule generic margins for the Palisades vessel using the same analytic methods as those used in formulating the rule. The staff has reviewed the other PWR vessels and, based upon currently available information, believes that the Palisades vessel will reach the PTS screening criteria by late 1999, before any other PWR.

On March 27 and 28, 1995, the staff reviewed the Asea Brown Boveri-Combustion Engineering proprietary RPV data-base. The most significant information reviewed concerned the Kewaunee RPV. The particular concern was the impact of data generated subsequent to the response to GL 92-01, Revision 1, on the plant's PTS evaluation. The staff met with the licensee for Kewaunee (April 13, 1995) to discuss issues related to consideration of all appropriate chemical composition data in addition to the applicable surveillance program data. In that meeting, the licensee presented its plant-specific surveillance program results and some new information related to the chemical composition variability in the RPV welds. Based upon this information, the licensee believes that the Kewaunee vessel will not exceed the PTS screening criteria before EOL. The staff has not completed its review of the new information on the Kewaunee vessel. However, based on the new vessel specific surveillance data, chemical composition data and the greater margin to the PTS screening criteria (300°F for the limiting Kewaunee circumferential weld compared to 270°F for the limiting Palisades axial weld), the staff believes that the Kewaunee vessel will not exceed the PTS screening criteria before the Palisades vessel. A key aspect of the Kewaunee review is the determination of the need for use of the ratio procedure in accordance with the established Position 2.1 of Regulatory Guide 1.99, Revision 2, by licensees using surveillance data.

NRC Staff Generic Evaluation of RPV Structural Integrity Data for PTS Events

The staff is assessing the generic implications of chemical composition variability with regard to the current methodology for ensuring protection against PTS events for PWRs. The staff considers that the larger variability observed in recent reviews could be applicable to other reactor vessels and may, therefore, reduce the margins of safety provided by the PTS screening criteria. The staff will evaluate this concern as part of its review of plant-specific evaluations and longer-term reassessment of the PTS rule.

To provide assurance that all PWRs will maintain adequate protection against PTS events while the PTS rule is being reassessed, the staff has assessed all of the PWR RPVs using generic values of chemistry and increased margin terms to account for potentially larger chemical composition variability. It should be noted that such analyses are considered conservative evaluations, that were performed to determine whether an immediate safety concern exists for this issue and whether there is adequate time to perform a more rigorous assessment of the issue. As stated in the previous section, based upon currently available information, the staff believes that the Palisades vessel will exceed the PTS screening criteria before any other PWR. However, because of the importance of RPV integrity and the potential impact of additional, unreviewed data on RPV evaluations, the staff believes that this issue needs to be resolved on an expedited basis.

Consideration of All Data Relevant to Reactor Pressure Vessel Integrity

As described previously, another result of recent reviews was that the staff became concerned that licensees might not necessarily have all of the data pertinent to the evaluation of the structural integrity of their RPVs. This is particularly true where the RPV fabricator holds, or has held, the applicable data to be proprietary in nature. Such data include, but are not limited to: chemical composition, heat treatment, plate and forging manufacturing process records, RPV fabrication records, all mechanical property data (tensile, impact, fracture toughness), and surveillance data. Sources of data that licensees should reexamine include material test reports from the steel producer, weld wire manufacturer, RPV fabricator, independent testing laboratories, and nuclear steam supply system (NSSS) vendor. Licensees are encouraged to work closely with their respective vessel owners groups and NSSS vendor groups to ensure that all sources of information pertinent to the analysis of the structural integrity of their RPVs have been considered. The information submitted in response to this generic letter should be considered to be public information.

Required Information

Addressees are required to provide the following information:

(1) a description of those actions taken or planned to locate all data relevant to the determination of RPV integrity, or an explanation of why the existing data base is considered complete as previously submitted;

(2) an assessment of any change in best-estimate chemistry based on consideration of all relevant data;

(3) a determination of the need for use of the ratio procedure in accordance with the established Position 2.1 of Regulatory Guide 1.99, Revision 2, for those licensees that use surveillance data to provide a basis for the RPV integrity evaluation; and

(4) a written report providing any newly acquired data as specified above and (1) the results of any necessary revisions to the evaluation of RPV integrity in accordance with the requirements of 10 CFR 50.60, 10 CFR 50.61, Appendices G and H to 10 CFR Part 50, and any potential impact on the LTOP or P-T limits in the technical specifications or (2) a certification that previously submitted evaluations remain valid. Revised evaluations and certifications should include consideration of Position 2.1 of Regulatory Guide 1.99, Revision 2, as applicable, and any new data.

Required Response

All addressees are required to submit the following written responses providing the information described above:

(1) within 90 days from the date of this generic letter, a written response to part (1) of the information requirement specified above; and

(2) within 6 months from the date of this generic letter, a written response to parts (2), (3), and (4) of the information requirement above.

Address the required written reports to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, under oath or affirmation under the provisions of Section 182a, Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). In addition, submit a copy to the appropriate regional administrator.

The NRC recognizes the potential difficulties (number and types of sources, age of records, proprietary data, etc.) that licensees may encounter while ascertaining whether they have all of the data pertinent to the evaluation of their RPVs. For this reason, 90 days is allowed for the initial response.

The information obtained from the licensees as a result of Revision 1 to GL 92-01 has been entered into a computerized reactor vessel integrity database (RVID), which will be made publicly available in the third quarter of 1995. The NRC intends to hold a public meeting on this GL supplement within 30 days of its issuance and a public workshop on RPV integrity, addressing the RVID and other RPV integrity issues, in the third quarter of 1995.

Related Generic Communications

(1) NRC Generic Letter 92-01, Revision 1, "Reactor Vessel Structural Integrity," March 6, 1992.
(2) NRC Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," July 12, 1988.

Backfit Discussion

This generic letter supplement only requires information from the addressees under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). Therefore, the staff has not performed a backfit analysis. The information required will enable the NRC staff to determine whether licensees are complying with the requirements of

10 CFR 50.60, 10 CFR 50.61, Appendices G and H to 10 CFR Part 50 and any associated license conditions, and licensee commitments related to GL 88-11 and GL 92-01, Revision 1. The staff is not establishing a new position for such compliance in this generic letter supplement. Therefore, this generic letter supplement does not constitute a backfit and no documented evaluation or backfit analysis need be prepared.

Federal Register Notification

A notice of opportunity for public comment was not published in the *Federal Register* because the NRC needs to receive the responses to the generic letter in an expeditious manner. However, comments on the technical issue(s) addressed by this generic letter may be sent to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555.

Paperwork Reduction Act Statement

The information collections contained in this request are covered by the Office of Management and Budget clearance number 3150-0011, which expires July 31, 1997. The public reporting burden for this collection of information is estimated to average 600 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needs, and completing and reviewing the collection of information. Send comments regarding this burden estimate or any other aspect of this collection of information, including suggestions for reducing this burden, to the Information and Records Management Branch (T-6 F33), U.S. Nuclear Regulatory Commission, Washington, D.C., 20555-0001, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0011), Office of Management and Budget, Washington, D.C. 20503.

Compliance with the following request for information is voluntary. The information would assist the NRC in evaluating the cost of complying with this GL supplement.

(1) the licensee staff time and costs to perform requested record reviews and developing plans for inspections;
(2) the licensee staff time and costs to prepare the requested reports and documentation;
(3) the additional short-term costs incurred as a result of the inspection findings such as the cost of the corrective actions or the costs of down time; and
(4) an estimate of the additional long-term costs that will be incurred as a result of implementing commitments such as the estimated costs of conducting future inspections and repairs.

If you have any questions about this matter, please contact the technical contacts listed below or the appropriate NRR project manager.

/s/'d by RPZimmerman

Roy P. Zimmerman
Associate Director for Projects
Office of Nuclear Reactor Regulation

Technical contacts: Edwin M. Hackett
(301) 415-2751

Keith R. Wichman
(301) 415-2757

Lead project manager: Daniel G. McDonald
(301) 415-1408

Attachments: 1. References
2. List of Recently Issued NRC Generic Letters

(NUDOCS Accession Number 9505090312)

ATTACHMENT 1

GL 92-01, Rev. 1, Supp.
May 19, 1995

References

[1] NUREG-1511, "Reactor Pressure Vessel Status Report," U.S. Nuclear Regulatory Commission, Washington, DC, December, 1994.

[2] Letter from Elinor Adensam, USNRC, to Kurt Haas, Consumers Power Company forwarding, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to the Evaluation of the Pressurized Thermal Shock Screening Criteria, Consumers Power Company, Palisades Plant, Docket No. 50-255", April 12, 1995.

TO: ALL HOLDERS OF OPERATING LICENSES OR CONSTRUCTION PERMITS FOR NUCLEAR POWER PLANTS (EXCEPT YANKEE ATOMIC ELECTRIC COMPANY, LICENSEE FOR THE YANKEE NUCLEAR POWER STATION)

SUBJECT: REACTOR VESSEL STRUCTURAL INTEGRITY, 10 CFR 50.54(f) (GENERIC LETTER 92-01, REVISION 1)

This letter replaces Generic Letter 92-01 dated February 28, 1992. The background information concerning NRC's assessment of embrittlement in the Yankee Nuclear Power Station reactor vessel was drafted by staff some months ago and has now been clarified and updated to better reflect the licensee's extensive technical efforts regarding reactor vessel integrity. The section pertaining to required information has not changed.

The U.S. Nuclear Regulatory Commission (NRC) is issuing this generic letter to obtain information needed to assess compliance with requirements and commitments regarding reactor vessel integrity in view of certain concerns raised in the staff's review of reactor vessel integrity for the Yankee Nuclear Power Station. In Section 50.60(a) of Title 10 of the Code of Federal Regulations (10 CFR 50.60(a)), the NRC requires that licensees for all light water nuclear power reactors meet fracture toughness requirements and have a material surveillance program for the reactor coolant pressure boundary. These requirements are set forth in Appendices G and H to 10 CFR Part 50. In 10 CFR 50.60(b), where the requirements of Appendices G and H to 10 CFR Part 50 cannot be met, an exemption is necessary pursuant to 10 CFR 50.12. In 10 CFR 50.61 the NRC also provided fracture toughness requirements for protecting pressurized water reactors against pressurized thermal shock events. Licensees and permit holders have also made commitments in response to Generic Letter (GL) 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations," to use the methodology in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," to predict the effects of neutron irradiation as required by Paragraph V.A of 10 CFR Part 50, Appendix G. The 10 CFR 50.60 and 10 CFR 50.61 requirements and GL 88-11 are in the overall regulatory program to maintain the structural integrity of the reactor vessel.

This generic letter is part of a program to evaluate reactor vessel integrity and take regulatory actions, if needed, to ensure that licensees and permit holders are complying with 10 CFR 50.60 and 10 CFR 50.61, and are fulfilling commitments made in response to GL 88-11. Enclosure 1 is a discussion of the applicable regulatory requirements. The NRC is requiring information on compliance under the provisions of 10 CFR 50.54(f).

- 2 -

Assessment of Embrittlement for the Yankee Nuclear Power Station Reactor Vessel

In an effort to resolve concerns regarding the neutron embrittlement of the Yankee reactor vessel, the staff performed a safety assessment of the Yankee reactor vessel. The staff found that the licensee for the Yankee Nuclear Power Station might not be in compliance with 10 CFR 50.60 and 10 CFR 50.61.

The staff found that the Charpy upper shelf energy of the Yankee reactor vessel material could be as low as 35.5 foot-pounds which is less than the 50 foot-pound value required in Appendix G to 10 CFR Part 50. However, the licensee for the Yankee Nuclear Power Station had not performed the actions required in Paragraphs IV.A.1 or V.C of Appendix G to 10 CFR Part 50. Since then, the licensee has performed an analysis in accordance with Paragraph IV.A.1 of Appendix G to 10 CFR Part 50 using criteria being developed by the American Society of Mechanical Engineers (ASME) to demonstrate margins of safety equivalent to those in the ASME Code.

The NRC expressed a concern regarding compliance with the requirements of Appendix H to 10 CFR Part 50. Section E 185 of the American Society for

Testing and Materials (ASTM) Code requires that the licensee take sample specimens from actual material used in fabricating the beltline of the reactor vessel. These surveillance materials shall include one heat of base metal, one butt weld, and one weld "heat affected zone." The licensee for the Yankee Nuclear Power Station terminated the material surveillance program in 1965. Therefore, the Yankee Nuclear Power Station had no material surveillance program on July 26, 1983, when Appendix H to 10 CFR Part 50 became effective. Further, the samples irradiated at Yankee Rowe before 1965 were comprised only of base metal.

The licensee for the Yankee Nuclear Power Station had used the methodology in draft Regulatory Guide 1.99, Revision 2, to predict the effects of neutron embrittlement. The staff raised concerns regarding the licensee's application of the methodology. The specific issues were (1) the irradiation temperature, (2) the chemistry composition of reactor vessel material, and (3) the results of the material surveillance program.

The irradiation temperature at the Yankee Nuclear Power Station is between 454 øF and 520 øF, which is below the nominal irradiation temperature of 550 øF used in developing Regulatory Guide 1.99, Revision 2. A lower irradiation temperature increases the effect of neutron embrittlement. The regulatory guide indicates that for irradiation temperatures less than 525 øF, embrittlement effects should be considered to be greater than predicted by the methods of the guide. Adjustments that were made by the licensee were insufficient to account for this effect.

The results of the surveillance program from the Yankee Nuclear Power Station indicated that the increase in the reference temperature exceeds the mean-plus-two standard deviations as predicted by the procedures in Regulatory Guide 1.99, Revision 2. The regulatory guide states that the licensee should use credible surveillance data to predict the increase in reference temperature resulting from neutron irradiation.

.

- 3 -

The staff implemented RG 1.99, Revision 2, by issuing GL 88-11. In committing to GL 88-11, licensees have committed to calculate radiation embrittlement in accordance with the procedures documented in RG 1.99, Revision 2. To meet the limitations in Section 1.3 of the regulatory guide, the licensee should consider the effects on irradiation embrittlement during core critical operation with irradiation temperatures less than 525 øF. Section 2 of the regulatory guide states that the licensees should consider the effects of the results from its surveillance capsules.

The Summer 1972 Addenda of the 1971 Edition of Section III of the ASME Boiler and Pressure Vessel Code are the earliest code requirements for testing materials to determine their unirradiated reference temperature. The Yankee reactor vessel was constructed in 1959 to ASME Code, Section VIII. Therefore, the unirradiated reference temperature could not be established in accordance with the requirements of the Summer 1972 Addenda. The licensee for the Yankee Nuclear Power Station extrapolated the available test results to determine an unirradiated reference temperature. The staff determined that the licensee's extrapolation was not conservative.

The chemical composition of the Yankee reactor vessel welds is unknown. The material's sensitivity to neutron embrittlement depends on its chemical content. The licensee assumed that the chemistry of its welds was equivalent to that of the BR-3 reactor vessel in Mol, Belgium. The heat number of the wire used to fabricate the Yankee welds was not available. The licensee was assuming a chemical composition that was not based on its plant-specific information, since the chemical composition, in particular, the amount of copper, depends upon the heat number of the weld wire.

These factors prompted the staff to find that the licensee for the Yankee Nuclear Power Station had not fully considered plant-specific information in assessing compliance with 10 CFR 50.61. When plant-specific information is considered, the Yankee reactor vessel may have exceeded the screening criteria in 10 CFR 50.61.

Upon conducting the Yankee Nuclear Power Station review, the staff became concerned about other licensee's compliance with 10 CFR 50.60 and 10 CFR 50.61 and fulfillment of commitments made in response to GL 88-11. Thus, the staff is issuing this generic letter to obtain information to assess compliance with these regulations and fulfillment of commitments. The staff is continuing to pursue this concern with the Yankee Atomic Electric Company. Therefore, the Yankee Atomic Electric Company need not respond to this generic letter.

Required Information

Portions of the following information requested are not applicable to all addressees. The responses provided should, in these cases, indicate that the requested information is not applicable and why it is not applicable.

.

1. Certain addressees are requested to provide the following information regarding Appendix H to CFR Part 50:

   Addressees who do not have a surveillance program meeting ASTM E 185-73, -79, or -82 and who do not have an integrated surveillance program approved by the NRC (see Enclosure 2), are requested to describe actions taken or to be taken to ensure compliance with Appendix H to 10 CFR Part 50. Addressees who plan to revise the surveillance program to meet Appendix H to 10 CFR Part 50 are requested to indicate when the revised program will be submitted to the NRC staff for review. If the surveillance program is not to be revised to meet Appendix H to 10 CFR Part 50, addressees are requested to indicate when they plan to request an exemption from Appendix H to 10 CFR Part 50 under 10 CFR 50.60(b).

2. Certain addressees are requested to provide the following information regarding Appendix G to 10 CFR Part 50:

   a. Addressees of plants for which the Charpy upper shelf energy is predicted to be less than 50 foot-pounds at the end of their licenses using the guidance in Paragraphs C.1.2 or C.2.2 in Regulatory Guide 1.99, Revision 2, are requested to provide to the NRC the Charpy upper shelf energy predicted for December 16, 1991, and for the end of their current license for the limiting beltline weld and the plate or forging and are requested to describe the actions taken pursuant to Paragraphs IV.A.1 or V.C of Appendix G to 10 CFR Part 50.

   b. Addressees whose reactor vessels were constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition are requested to describe the consideration given to the following material properties in their evaluations performed pursuant to 10 CFR 50.61 and Paragraph III.A of 10 CFR Part 50, Appendix G:

      (1) the results from all Charpy and drop weight tests for all unirradiated beltline materials, the unirradiated reference temperature for each beltline material, and the method of determining the unirradiated reference temperature from the Charpy and drop weight test;

      (2) the heat treatment received by all beltline and surveillance materials;

      (3) the heat number for each beltline plate or forging and the heat number of wire and flux lot number used to fabricate each beltline weld;

      (4) the heat number for each surveillance plate or forging and the heat number of wire and flux lot number used to fabricate the surveillance weld;

      (5) the chemical composition, in particular the weight in percent of copper, nickel, phosphorous, and sulfur for each beltline and surveillance material; and

      (6) the heat number of the wire used for determining the weld metal chemical composition if different than Item (3) above.

3. Addressees are requested to provide the following information regarding commitments made to respond to GL 88-11:

   a. How the embrittlement effects of operating at an irradiation temperature (cold leg or recirculation suction temperature) below 525 ØF were considered. In particular licensees are requested to describe consideration given to determining the effect of lower irradiation temperature on the reference temperature and on the Charpy upper shelf energy.

   b. How their surveillance results on the predicted amount of embrittlement were considered.

   c. If a measured increase in reference temperature exceeds the mean-plus-two standard deviations predicted by Regulatory Guide 1.99, Revision 2, or if a measured decrease in Charpy upper shelf energy exceeds the value predicted using the guidance in Paragraph C.1.2 in Regulatory Guide 1.99, Revision 2, the licensee is requested to report the information and describe the effect of the surveillance results on the adjusted reference temperature and

Charpy upper shelf energy for each beltline material as predicted for December 16, 1991, and for the end of its current license.

Reporting Requirements

Pursuant to Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f), each addressee shall submit a letter within 120 days of the date of this generic letter providing the information described under "Required Information." The letter shall be addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555, under oath or affirmation. *A copy shall also be submitted to the* appropriate Regional Administrator. This generic letter requests information that will enable the NRC to verify that the licensee is complying with its current licensing basis regarding reactor vessel fracture toughness and material surveillance for the reactor coolant pressure boundary. Accordingly, an evaluation justifying this information request is not necessary under 10 CFR 50.54(f).

- 6 -

Backfit Discussion

This generic letter requests information that will enable the NRC staff to determine whether licensees are complying with their prior commitments and any license conditions regarding 10 CFR 50.60, 10 CFR 50.61, and GL 88-11. The staff is not establishing a new position for such compliance in this generic letter. The staff is requesting information to verify that the licensee is complying with its previously established commitments and is not establishing any new position. Therefore, this generic letter does not constitute a backfit and no documented evaluation or backfit analysis need be prepared.

Request for Voluntary Submittal of Impact Data

This request is covered by Office of Management and Budget Clearance Number 3150-0011, which expires May 31, 1994. The estimated average number of burden hours is 200 person hours for each addressee's response, including the time required to assess the requirements, search data sources, gather and analyze the data, and prepare the required letters. This estimated average number of burden hours pertains only to the identified response-related matters and does not include the time to implement the actions required by the regulations. Comments on the accuracy of this estimate and suggestions to reduce the burden may be directed to Ronald Minsk, Office of Information and Regulatory Affairs (3150-0011), NEOB-3019, Office of Management and Budget, Washington, DC 20503, and to the U.S. Nuclear Regulatory Commission, Information and Records Management Branch, Division of Information Support Services, Office of Information and Resources Management, Washington, DC 20555.

Although no specific request or requirement is intended, the following information would assist the NRC in evaluating the cost of complying with this generic letter:

(1) the licensee staff's time and costs to perform requested inspections, corrective actions, and associated testing;

(2) the licensee staff's time and costs to prepare the requested reports and documentation;

(3) the additional short-term costs incurred to address the inspection findings such as the costs of the corrective actions or the costs of down time; and

(4) an estimate of the additional long-term costs that will be incurred as a result of implementing commitments such as the estimated costs of conducting future inspections or increased maintenance.

- 7 -

If you have any questions about this matter, please contact one of the NRC technical contacts or the lead project manager listed below.

Sincerely,

James G. Partlow
Associate Director for Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Applicable Regulatory Requirements

2. Plants with Integrated Programs
3. List of Recently Issued Generic Letters

Technical Contacts:
Barry J. Elliot, NRR
(301) 504-2709

Keith R. Wichman, NRR
(301) 504-2757

Lead Project Manager:
Daniel G. McDonald, NRR
(301) 504-1408

.

Enclosure 1

Regulatory Requirements Applicable to

Reactor Vessel Structural Integrity

10 CFR 50.60

Pursuant to 10 CFR 50.60, all light water nuclear power reactors must meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in Appendices G and H to 10 CFR Part 50.

The fracture toughness of the reactor coolant pressure boundary required by 10 CFR 50.60 is necessary to provide adequate margins of safety during any condition of normal plant operation, including anticipated operational occurrences and system hydrostatic tests. The material surveillance program required by 10 CFR 50.60 monitors changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors resulting from exposure of these materials to neutron irradiation and the thermal environment. Under the program, fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel.

Appendix G to 10 CFR Part 50 requires that the reactor vessel beltline materials must have Charpy upper shelf energy of no less than 50 ft-lb throughout the life of the vessel. Otherwise, licensees are required to provide demonstration of equivalent margins of safety in accordance with Paragraph IV.A.1 of Appendix G to 10 CFR Part 50 or perform actions in accordance with Paragraph V.C of Appendix G to 10 CFR Part 50.

Appendix H to 10 CFR Part 50 requires the surveillance program to meet the American Society for Testing and Materials (ASTM) Standard E 185, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels." Further, Appendix H to 10 CFR Part 50 specifies the applicable edition of ASTM E 185. Appendix H to 10 CFR Part 50, as amended on July 26, 1983, requires that the part of the surveillance program conducted before the first capsule is withdrawn must meet the requirements of the 1973, the 1979, or the 1982 edition of ASTM E 185 that is current on the issue date of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code under which the reactor vessel was purchased. The licensee may also use later editions of ASTM E 185 which have been endorsed by the NRC. The test procedures and reporting requirements for each capsule withdrawal after July 26, 1983 must meet the requirements of the 1982 edition of ASTM E 185 to the extent practical for the configuration of the specimens in the capsule. The licensee may use either the 1973, the 1979, or the 1982 edition of ASTM E 185 for each capsule withdrawal before July 26, 1983.

.

- 2 -

Licensees, especially those with reactor vessels purchased before ASTM issued the 1973 edition of ASTM E 185, may have surveillance programs that do not meet the requirements of Appendix H to 10 CFR Part 50 but may have alternative surveillance programs. The licensee may use these alternative surveillance programs in accordance with 10 CFR 50.60(b) if the licensee has been granted an exemption by the Commission under 10 CFR 50.12.

The licensee must monitor the test results from the material surveillance program. According to Paragraph III.C of Appendix H to 10 CFR Part 50, the results of the surveillance program may indicate that a technical specifications change is required, either in the pressure-temperature limits or in the operating procedures required to meet the limits.

10 CFR 50.61

Pursuant to 10 CFR 50.61, there are fracture toughness requirements for protection against pressurized thermal shock events for pressurized water reactors. Licensees are required to perform an assessment of the projected values of reference temperature. If the projected reference temperature exceeds the screening criteria established in 10 CFR 50.61, licensees are required to submit an analysis and schedule for such flux reduction programs as are reasonably practicable to avoid exceeding the screening criteria. If no reasonably practicable flux reduction program will avoid exceeding the screening criteria, licensees shall submit a safety analysis to determine what actions are necessary to prevent potential failure of the reactor vessel if continued operation beyond the screening criteria is allowed. In 10 CFR 50.61(b)(1), as amended effective June 14, 1991 (56 Fed Reg 22300 et. seq., May 15, 1991), licensees are required to submit their assessment by December 16, 1991, if the projected reference temperature will exceed the screening criteria before the expiration of the operating license.

Plant-specific information is required to be considered in assessing the level of neutron embrittlement as specified in 10 CFR 50.61(b)(3). This information includes but is not limited to the reactor vessel operating temperature and surveillance results.

Prediction of Irradiation Embrittlement

Paragraph V.A of Appendix G to 10 CFR Part 50 requires the prediction of the effects of neutron irradiation on reactor vessel materials. The extent of neutron embrittlement depends on the material properties, thermal environment, and results of the material surveillance program. In Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations," the staff stated that it will use the guidance in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," in estimating the embrittlement of the materials in the reactor vessel beltline. All licensees and permittees have responded to Generic Letter 88-11 committing to use the methodology in Regulatory Guide 1.99,

.

- 3 -

Revision 2, in predicting the effects of neutron irradiation as required by Paragraph V.A of 10 CFR Part 50, Appendix G. The methodology in Regulatory Guide 1.99, Revision 2, is also the basis in 10 CFR 50.61 in projecting the reference temperature.

.

Enclosure 2

Plants With Integrated Surveillance Programs Approved By The NRC

Oconee Units 1, 2, and 3
Arkansas Nuclear One Unit 1
Rancho Seco
Three Mile Island Unit 1
Davis-Besse
Ginna
Point Beach Units 1 and 2
Surry Units 1 and 2
Turkey Point Units 3 and 4
Zion Units 1 and 2

Screen Blockage + passage to core

Discovery of Sump Performance Issue

The NRC first published regulatory guidance on the performance of pressurized-water reactor (PWR) containment sumps and boiling-water reactor (BWR) suction strainers in 1974 with the issuance of revision 0 of Regulatory Guide (RG) 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident." BWR suction strainers perform the same function as PWR containment sump screens.

As part of its responsibility to ensure public health and safety, NRC continually assesses the design and operation of nuclear power plants to determine whether its regulations, its guidanc or nuclear power plant design or operations need to be modified.

Late 1970s

Because of internal questions by the NRC staff, the NRC first sponsored research to study the accumulation of debris on PWR containment sump screens and BWR suction strainers in the lat 1970s (approximately 1979). With the information and engineering tools available in the late 1970s and early 1980s, the NRC concluded that its regulatory guidance needed to be revised and issued in 1985 revision 1 of RG 1.82. As documented in Generic Letter 85-22, "Potential fo Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage," the NRC concluded that no additional regulatory action was warranted for operating nuclear power plant but that new nuclear power plants would need to satisfy the guidance in the revised RG 1.82, and that operating nuclear power plants should consider the guidance in the revised RG 1.82 when making plant modifications, namely changing thermal insulation.

Early 1990s

In the early 1990s, because of an event at a BWR in Sweden and several events at BWRs in th United States, the NRC sponsored new research to study the accumulation of debris on BWR suction strainers. Based on the information learned from operational experience and the new research using more sophisticated engineering tools than those readily available in the late 1970s, the NRC concluded that its guidance needed additional revision for BWRs. In 1996, the NRC issued revision 2 of RG 1.82. Using new research data and better engineering tools, the NRC concluded that additional regulatory action was needed to ensure safety is maintained at BWR power plants. As documented in Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors," the NRC concluded that additional regulatory action was warranted for operating BWRs and asked BWRs to conduct plant-specific evaluations of their suction strainer performance and, if necessary, modify their plant design and/or operation.

Late 1990s

Because of the information the NRC learned during the assessment of BWR suction strainers ar oversight of BWR plant-specific evaluations and modifications, the NRC sponsored a new research effort to study the accumulation of debris on PWR containment sump screens. Based the most recent research study, "GSI-191 Technical Assessment: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance," the NRC concluded that its guidance needed additional revision for PWRs. In November 2003, the NRC issued revision 3 o RG 1.82. The NRC has concluded that additional regulatory action is warranted. Currently, the NRC is implementing its plan to have all PWR licensees perform a plant-specific evaluation for the potential for excessive head loss across the containment sump screen because of the accumulation of debris on the containment sump screen. The NRC also expects licensees to evaluate effects of debris that might pass through the sump screens.

Based on the information available to date, continued operation of PWRs is justified until plant-specific evaluations are completed. To provide additional assurance regarding the continued operation of PWRs, the NRC asked the licensees of PWRs to implement compensatory measure This was done through the issuance of Bulletin 2003-01, "Potential Impact of Debris Blockage c Emergency Sump Recirculation at Pressurized-Water Reactors." If the results of ongoing NRC inspections and reviews or ongoing and planned studies indicate that unsafe conditions exist at any operating PWR, the NRC will take immediate actions to ensure the continued health and safety of the public. Also, if a licensee discovers that it is not in compliance with the NRC regulations during the implementation of the requested actions in Bulletin 2003-01, it is requir to take prompt corrective actions.

TOP

Tuesday, February 13, 2007

GSI-191

Long-Term Core Cooling

Core Inlet Flow Blockage

February 6, 2007

Topics

- Problem Statement
- PWR Vessel Design Comparisons
  - Selection of plant to be modelled
- WCOBRA/TRAC Modelling Approach
- Preliminary Results

Problem Statement

- For a Double-Ended Guillotine Break:
  - RWST can be Depleted, and,
  - Sump Recirculation Begun Within ~ 20 Minutes
- Fibrous Debris and Particulates Can Pass Through Sump Screen
- Potential for Build-up at Core Inlet
  - Fuel assembly bottom nozzle, debris filter, grids
  - 1/8 inch of uniform matting with trapped particulates sufficient to cause high head loss

Selection of Limiting Break

Double-Ended Cold Leg

- Spilling of ECCS to containment
- Gravity head to loop level only
  - True for no single failure also
- Lower flow results in slower debris build-up

Double-Ended Hot Leg

- No spilling of ECCS
- Additional driving head from ECCS pumps
  - more for no single failure
- Higher flow results in faster build-up

Use Double-Ended Cold Leg Break With No Delay in Debris Build-up

Vessel Design Considerations (W OEM)

- Designed Upflow is Least Limiting
  - Numerous large pressure relief holes in baffle wall allow flow to bypass core inlet if blocked
- Converted Upflow is More Limiting
  - No pressure relief holes, limited flow to top of core (if any)
- Downflow is Most Limiting
  - Flow must enter core through lower core plate

Other PWR Vessel Designs

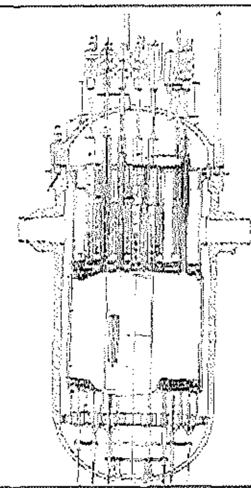
- B&W Design Similar to W Designed Upflow
  - Numerous large pressure relief holes in baffle wall allow flow to bypass core inlet if blocked
  - Barrel vent valves located above loop level
    - No impact on this issue
- CE Design Similar to W Converted Upflow No pressure relief holes, limited flow to top of core (if any)

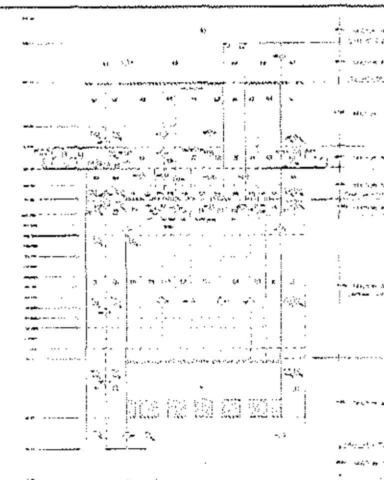
W Downflow Design is Bounding

Plant Selection

- Downflow Most Limiting Configuration
- Core Power Density Also Important for Heat Removal
  - Use available 3-loop downflow model for plant rated at 2900 MWt

Vessel Sketch and Noding Diagram

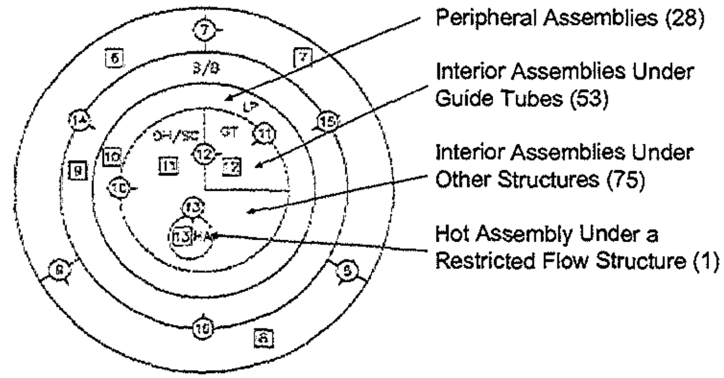




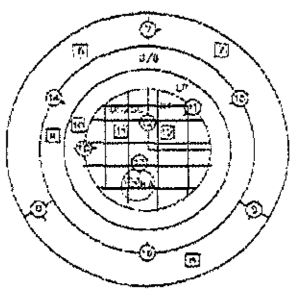
WCOBRA/TRAC Modeling Approach

- Run Problem from Break Initiation
  - Create Single Use code version which ramps in high resistance as specified by User
  - Ramp in large increase in resistance at core inlet of PWR model
  - Investigate effect of radial distribution of resistances (conceptual at this point)

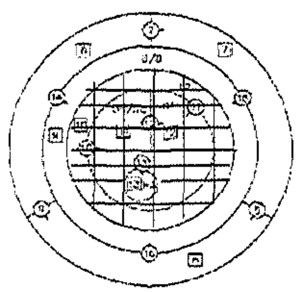
Standard Core Modelling



Blockage Approaches (Conceptual)



Block All Except Peripheral (82%)

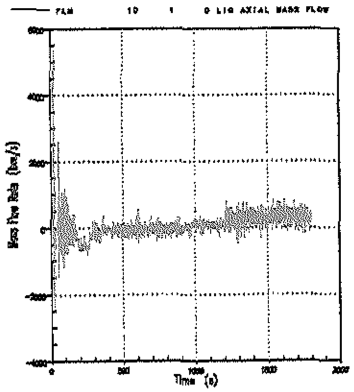


Block All Except Hot Assembly (99.4%)

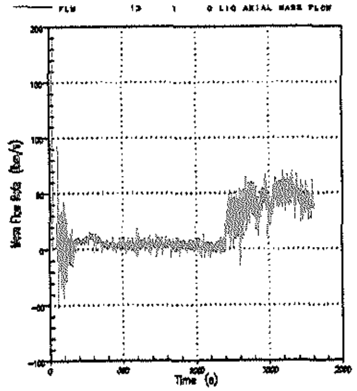
Results to Date

- Both Blockage Cases Run to 30 Minutes
  - Blockage ramped in from 20 to 20.5 minutes
    - *K = 1000 used to simulate blockage*
- Evaluation of Initial Results
  - Need larger K to adequately block channels
    - Case with 99.4% blockage has hot assembly flow approaching that required to replace core boil-off due to decay heat
    - Case with 82% blockage has total peripheral assembly flow well in excess of boil-off

Flow Through Unblocked Channel



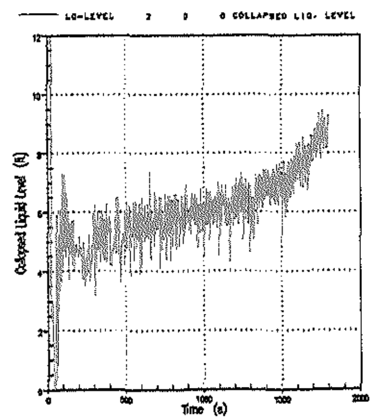
82% Blockage



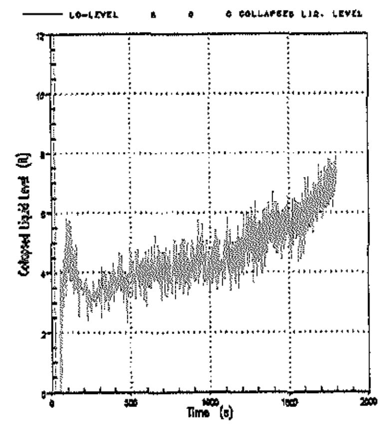
99.4% Blockage

Need < 60 lb/sec to replace boil-off at 20 min.

Collapsed Liquid Level in Core

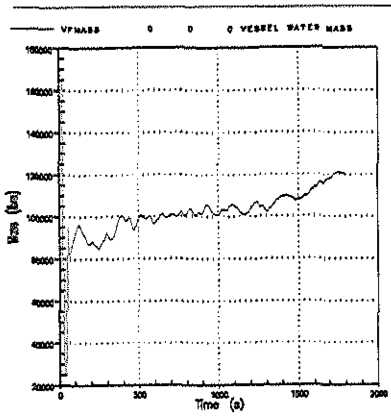


82% Blockage

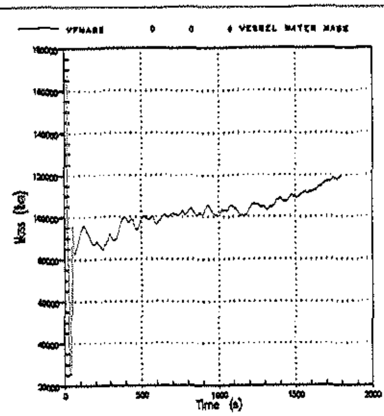


99.4% Blockage

Total Vessel Mass

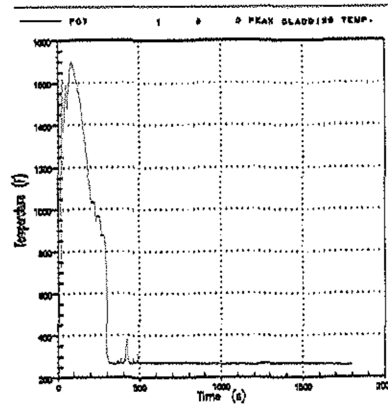


82% Blockage

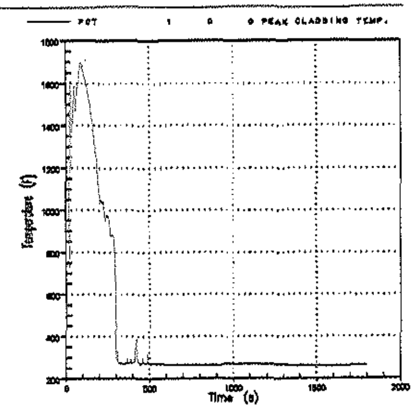


99.4% Blockage

Peak Cladding Temperature in Core



82% Blockage



99.4% Blockage

Summary

- Initial Results
  - Indicate Blockages > 90% Can Be Accommodated
  - Are consistent with NRC calculations performed as part of an audit
- Further Investigations to be Performed
  - Improved modelling of blockage
  - Longer transients
  - Further study of flow vs. blockage vs. core boil-off rates

Overview of Reactor Pressure Boundary Integrity Issues

On this page:

Generic Activities on Alloy-600 Cracking

Alloy 600 is used to fabricate various parts in nuclear power plants, including reactor vessel to head penetrations for control rod drive mechanism (CRDMs), control drive element mechanism (CEDMs), in-core instruments (ICIs) and thermocouples, reactor vessel bottom head bottom mounted instruments (BMIs), pressurizer heater sleeves, and various other instrumentation ports. Related weld materials Alloy 82 and Alloy 182 are used to join these Alloy 600 parts to the ferritic steel components and also as a bi-metallic weld joining ferritic base materials to austenitic stainless steel base materials. Alloy 600 and its associated weld filler metals were originally used because of expectations of resistance to service-induced cracking. However, pa fabricated from these materials have demonstrated a susceptibility to primary water stress corrosion cracking (PWSCC), also referred to as low potential stress corrosion cracking (LPSCC

In the United States, PWSCC of Alloy 600 became an issue following a leakage event of a pressurizer heater sleeve nozzle at Calvert Cliffs Unit 2 in 1989. Other instances of leakage in pressurizer instrument nozzles were identified in both domestic and foreign PWRs, as describe in Information Notice 90-10.

The first indication of cracking in upper head Alloy 600 penetrations was identified in France at Bugey Unit 3 in 1991 during the ten-year primary system hydrostatic test. The leakage was from an axial flaw that had initiated on the nozzle inside surface near the elevation of the J-groove weld. Several other partial depth axial cracks were identified at a similar elevation in th nozzle. Failure analysis confirmed that the cracking was due to PWSCC.

In the United States, the NRC and the industry initiated activities to assess the safety significance of VHP nozzle cracking. An action plan was implemented by the NRC staff in 1991 address PWSCC of Alloy 600 VHPs at all U.S. PWRs. This action plan included a review of safet assessments submitted by the PWR Owners Groups, the development of VHP mock-ups by the Electric Power Research Institute (EPRI), the qualification of inspectors on the VHP mock-ups b EPRI, the review of proposed generic acceptance criteria from the Nuclear Utility Management and Resource Council (NUMARC) [now the Nuclear Energy Institute (NEI)], and VHP inspection As part of this action plan, the NRC staff met with each of the owners groups separately and with the entire industry throgh NUMARC/NEI. After reviewing the industry's safety assessment and examining the overseas inspection findings, the NRC staff concluded in a safety evaluation dated November 19, 1993, [3] that VHP nozzle cracking was not an immediate safety concern. The bases for this conclusion were that if PWSCC occurred at VHP nozzles (1) the cracks would be predominately axial in orientation, (2) the cracks would result in detectable leakage before catastrophic failure, and (3) the leakage would be detected during visual examinations performed as part of surveillance walkdown inspections before significant damage to the reactc vessel closure head would occur.

The first U.S. inspection of VHPs took place in the spring of 1994 at the Point Beach Nuclear Generating Station, and no indications were detected in any of its 49 CRDM penetrations. The eddy current inspection at the Oconee Nuclear Station (ONS) in the fall of 1994 revealed 20 indications in one penetration. Ultrasonic testing (UT) did not reveal the depth of these indications because they were shallow. UT cannot accurately size defects that are less than one mil deep (0.03 mm). These indications may be associated with the original fabrication and may not grow; however, they will be reexamined during the next refueling outage. A limited examination of eight in-core instrumentation penetrations conducted at the Palisades plant found no cracking. An examination of the CRDM penetrations at the D. C. Cook plant in the fall of 1994 revealed three clustered indications in one penetration. The indications were 46 mm (1.81 in.), 16 mm (0.63 in.), and 6 to 8 mm (0.24 to 0.31 in.) in length, and the deepest flaw was 6.8 mm (0.27 in.) deep. The tip of the 46-mm (1.81 in.) flaw was just below the J-groove weld. Virginia Electric and Power Company inspected North Anna Unit 1 during its spring 1996 refueling outage. Some high-stress areas (e.g., upper and lower hillsides) were examined on each outer ring CRDM penetrations and no indications were observed using eddy current testin During this time, each of the vendors was developing a susceptibility model for VHP nozzles based on a number of factors, including operating temperature, years of power operation, method of fabrication of the VHP, microstructure of the VHP, and the location of the VHP on the head. Each time a plant's VHPs are inspected, the inspection results are incorporated into the model.

TOP

Generic Letter 97-01

On April 1, 1997, the NRC issued Generic Letter 97-01 to request PWR licensees submit descriptions of their programs for inspecting CRDM and other VHP nozzle penetrations. The industry used a histogram grouping of plants, in combination with completed inspections and planned inspections as its approach for managing this issue. The plant grouping used probabilistic crack initiation and growth models to estimate the amount of time remaining (in effective full power years, EFPYs) until the plant reached a limiting condition for a reference plant. This limiting condition was the time for the plant to reach the same probability of having crack 75% through-wall as D.C. Cook Unit 2 had at the time a 6.5 mm deep crack was identifie in 1994. These models included differences in operating time and temperature, water chemistr environment, surface stress, component geometry, material yield strength and microstructure, and fabrication practices (amount of cold work during machining) between the subject plant an the reference plant in determining a plant's susceptibility.

Inspections continued into the fall of 2000, with no significant adverse results. The most significant crack identified in these inspections was a 6.8 mm (0.27 in.) deep crack found at D. Cook Unit 2. This flaw was repaired by a process that involved partial removal (by grinding) an overlay weld to isolate the remnant of the original flaw from the environment. Three plants identified small "craze cracks," generally found as cluster of shallow, less than 0.2 mm deep (0.008 in.) and axially oriented. At Millstone Unit 2, one nozzle with seven such indications in a single nozzle were removed by flapper wheel grinding to a depth of 0.8 mm (0.032 in.).

Worldwide, inspection activities were finding PWSCC in VHP nozzles, and in some cases RPV heads were being replaced. Common characteristics of these findings were the flaws originatin in the nozzle base material and located on the inside surface of the nozzles.

TOP

Hot Leg Axial Cracking at the V.C. Summer Nuclear Station

The hot leg weld axial cracking was identified at V.C. Summer on October 7, 2000, during a *normally scheduled inspection* of the V.C. Summer containment after entering a refueling outage. The licensee found over 200 pounds of boric acid crystals on the containment floor anc protruding from the air boot around the "A" loop reactor coolant system (RCS) hot leg pipe. Examinations showed a short through-wall axial crack in the hot leg nozzle safe end weld, approximately 3 feet from the reactor vessel. *Additional examinations* of the other 5 nozzle saf end welds found crack indications but no through-wall cracks. A 12-inch long section of the hot leg pipe containing the leaking weld was replaced with a new section of stainless steel pipe anc Alloy 52/152 welds. The remaining weld indications were analyzed and found to be safe for another cycle of operation.

In a letter dated December 14, 2000, Mr. David J. Modeen of the Nuclear Energy Institute (NEI informed Dr. Brian W. Sheron of the NRC *that the EPRI Materials Reliability Project* (MRP) intended to lead the industry's actions to address the generic implications of the cracking seen at V.C. Summer. The MRP is a utility-directed oversight organization of the PWR Owners Group whose purpose is to address and resolve, on a consistent industry-wide basis, PWR material-related issues.

TOP

Impetus for Bulletin 2001-01

In the fall of 2000, the inspection findings in RPV head penetrations became more significant. / ONS-1 that fall, boron deposits were identified on the RPV head at one CRDM nozzle and at fiv (of the eight) thermocouple nozzles (one of only two plants with small diameter thermocouple nozzles). Contrary to expectations, the boron deposits were *very small (less than 1 in. total* volume). Analysis of the CRDM nozzle identified an axial-radial PWSCC crack that initiated in tł J-groove weld and propagated part way into the outer diameter surface of the nozzle. The crac in the J-groove weld was arrested when it encountered the RPV head base material, consistent with expectations.

In February 2001, ONS-3 identified nine nozzles with leaks (again small deposits). Additional inspections, including ultrasonic, eddy current and *liquid penetrant examinations, identified* numerous part- and through-wall axial cracks, generally initiated on the outer diameter surface of the nozzles below the J-groove weld. During the repair of these nozzles, two of the nozzles were found to have through-wall circumferential crack extending 165 around the nozzle, although the cracks were not through-wall for their entire circumferential extent. These cracks were identified as having initiated on the nozzle outer diameter surface. The findings at ONS-3 were the subject of NRC Information Notice 2001-05, issued on April 30, 2001.

In March 2001, Arkansas Nuclear One Unit 1 (ANO-1) identified boron deposits on a single CRD nozzle. Examination of this nozzle identified an axial part-thoguh wall crack that initiated on th nozzle outer diameter surface below the J-groove weld and propagated to a distance 33 mm (1.3 in.) above the J-groove weld.

In April 2001, ONS-2 identified boron deposits on four CRDM nozzles. Eddy current examinatio of these nozzles identified cluster of shallow axial indications on the *nozzle inside surfaces*, ranging in depth from 0.35 to 0.8 mm (0.014 to 0.032 in.) and in length from 23 to 79 mm (0. to 3.1 in.). Ultrasonic examination of these nozzles identified numerous axial flaws on the nozz outer diameter surfaces, including one circumferential crack above the J-groove weld. The latte was reported as 32 mm (1.25 in.) long and 1.8 mm (0.07 in.) deep. Leakage from these nozzl was identified as originating from the outer diameter surface cracks that propagated along the weld to nozzle interface from below the J-groove weld to above the weld.

TOP

Bulletin 2001-01

The Nuclear Regulatory Commission issued Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," dated August 3, 2001, seeking information from all PWR nuclear power licensees regarding the structural integrity of reactor pressure vessel head penetrations.

The focus of this Bulletin was the safety issue of circumferential cracking in VHP nozzles, with a goal of providing assurance that no such issues existed in plants. Because of the time frame involved in the development of a circumferential crack that could be subject to nozzle ejection, visual inspections of the RPV head outer surface, where the nozzle intersected the RPV head, were considered at that point to be an adequate inspection.

TOP

Bulletin 2002-01

In response to the findings at Davis-Besse, the NRC issued Bulletin 2002-01 on March 18, 2002. The focus of this Bulletin was to assess licensee inspections and other information that could provide a basis for conclusions on the condition of the RPV head. The Bulletin also addressed boric acid corrosion of other parts of the reactor coolant system.

Following issuance of this Bulletin, the spring 2002 inspection findings were relatively quiet. The exception was an inspection at Millstone Unit 2, which identified three nozzles (no leaks) requiring repair, with axial outer diameter surface cracks that extended from below the J-groove into the weld zone. This finding is significant because this plant had the lowest susceptibility of any plant that had identified cracking.

TOP

Bulletin 2002-02

Since the initial findings of circumferential cracking at ONS-3, the nuclear industry was working to develop inspection recommendations (and justification for the recommendations) that would provide effective management of the issue. This effort was continually challenged by new findings, e.g., Davis-Besse upper head wastage, and the industry did not have a proposal available for consideration by the summer of 2002. To address cracking and wastage on the upper RPV head, the NRC issued Bulletin 2002-02 in August 2002. This Bulletin provided a description of a comprehensive inspection program that addressed a combination of visual and non-visual examinations on a graded approach consistent with a variety of plant susceptibilities to PWSCC. This Bulletin used a parameter referred to as effective degradation years (EDY) to characterize plant susceptibility to PWSCC. Calculation of this parameter requires information on the RPV head operating temperature(s) and the operating time (i.e., effective full power years, EFPY) at each operating temperature. These data are used to integrate the effects of operating temperature, normalized to 316 C (600 F).

Notable inspection findings were prevalent during the fall 2002 outages. North Anna Unit 2 identified two leaking nozzles. One of these leaks was from a nozzle that had received a weld over-lay repair at the previous outage. Failure of the of the repair was attributed to the weld over-lay repair not completely covering the original Alloy 182 weld butter, with cracking then occurring in the original weld at the periphery of the repair weld. Surface examinations of the J-groove welds identified more than half of the welds with cracks. Ultrasonic testing of the nozzle base material identified twenty nozzles with axial indications. Several nozzles were identified with circumferential cracks on the nozzle outer diameter surface within the zone of the J-groove weld, just below the root of the weld. With the myriad of repairs necessary due to these findings, this plant became the first U.S. plant to install a new RPV head using Alloy 690 nozzle base material and Alloy 52 and 152 welds.

At ANO-1, a leak was identified from the nozzle that had been repaired in the spring of 2001. The failure of the repair was attributed to the weld over-lay repair not completely covering the original J-groove weld, similar to the North Anna Unit 2 finding.

At Sequoyah Unit 2, minor head corrosion was identified from a boron leak located above the RPV head. In particular, the licensee identified a leak from a valve in the reactor vessel level instrument system (RVLIS). Leaking coolant impacted the RPV head insulation below the valve, fell through a seam in the insulation and onto the RPV head. After the RPV head was cleaned up, a corrosion area was identified with dimensions 127 mm (5 in.) long and 8 mm (5/16 in.) wide with a maximum depth of 3 mm (1/8 in.).

TOP

Davis-Besse

The Davis-Besse refueling outage began on February 16, 2002. The licensee planned to perforr a visual inspection of the outer surface RPV head looking for signs of boron deposits, and ultrasonic inspection of all CRDM nozzles. The inspection identified five nozzles with indications including three with through-wall cracks, and the licensee decided to repair all five nozzles. During machining to facilitate repair of nozzle #3, the equipment rotated and was removed fro the head. Upon removal, the licensee found that the nozzle had tipped, with the CRDM flange (located above the head) contacting the flange of an adjacent CRDM. The licensee cleaned the surface of the RPV head and found a large cavity adjacent to nozzle #3, where the RPV head base material had been corroded down to the stainless steel cladding. Subsequent investigatio revealed an additional much smaller degraded area near nozzle #2, located within the wall thickness (no cladding was exposed).

After the initial finding of the cavity at Davis-Besse, the NRC issued Information Notice 2002-1 "Recent Experience with Degradation of Reactor Pressure Vessel Head," on March 12, 2002. After some of the evidence began to be accumulated regarding secondary indications of a serious ferritic corrosion event, the NRC issued Information Notice 2002-13, "Possible Indicato of Ongoing Reactor Pressure Vessel Head Degradation," on April 4, 2002.

TOP

Order EA-03-009

The NRC issued Order EA-03-009 to all PWR licensees on February 11, 2003. This Order provided specific inspection requirements for all PWR plants. The Order required that plants evaluate their susceptibility to PWSCC using a formula for effective degradation years, EDY. Th Order then provided specific inspection requirements based upon the EDY level of the plant. Th Order provided requirements for plants with EDY greater than 12 or have experienced PWSCC. These plants were required to perform a bare metal visual examination and a non-visual examination every refueling outage. Moderate susceptibility plants (those with EDY from 8 to 12) were required to perform either bare metal visual or non-visual examination every outage, alternating the two methods each RFO. Low susceptibility plants (with EDY less than 8) were required to perform a bare metal visual examination by their second refueling outage after issuance of the Order and every third refueling outage or five years thereafter. In addition, low susceptibility plants were required to perform non-visual examination by February 11, 2008, a then repeat every fourth refueling outage or seven years thereafter.

The non-visual examinations described in the Order were ultrasonic examination or surface examination. The ultrasonic examination covered from the bottom of the nozzle to 2 inches above the J-groove weld, and included an assessment to determine if leakage has occurred in the interference fit zone of the nozzles. The scope of the surface examination included the surface of the J-groove weld, the outer diameter surface of the VHP nozzle base metal, and th inside surface of the VHP nozzle to a point 2 inches above the J-groove weld.

The Order provided explicit inspection requirements for repaired nozzles and welds, and makes no distinction for heads fabricated from Alloy 600 or Alloy 690.

In addition to the susceptibility based inspections of the RPV head surface and VHP nozzles, th Order required that all licensees perform visual inspections to identify boric acid leaks from components above the RPV head, with follow-up actions including inspection of potentially-affected RPV head areas and VHP nozzles should any leaks be identified.

The Order also provided means for licensees to request relaxation from its requirements upon demonstration of good cause. As of January 2004, twenty-four plants had made specific requests for relaxation. These requests related to limitations in inspection accessability and technology.

The NRC revised certain inspection aspects of the original NRC Order EA-03-009 with respect t bare metal visual inspections, penetration nozzle inspection coverage, flexibility in combination of non-destructive examination methods, flaw evaluation, and requirements for plants which have replaced their reactor pressure vessel head. The First Revised NRC Order EA-03-009 Rev. 01 was isssued on February 13, 2004.

TOP

Bottom-Mounted Instrument Cracking

With the focus of attention on PWSCC of Alloy 600 on the upper RPV head and possible boric acid corrosion of ferritic components throughout the reactor coolant system, visual examinatioı of other applications of Alloy 600 have increased in their thoroughness and effectiveness. One area that was not anticipated to provide short-term PWSCC concerns was the RPV lower head BMIs, due to the cold-leg operating temperature of the RPV lower head. However, in the sprinç of 2003, the licensee for the South Texas Project Unit 1 (STP-1) identified apparent boron deposits on the lower RPV head near two BMIs. Characterization of all of the BMI nozzles at ST 1 identified PWSCC in these two nozzles, and no PWSCC in any other nozzle. The operating temperature of the STP-1 lower head was ~ 294 C (561 F), and the calculated EDY was less than three (3). The NRC issued Bulletin 2003-02 to obtain information on licensee inspection activities and inspection plans for the RPV lower head. Thus far other plants have identified white residue on the lower head, frequently boron traced to refueling seal leakage or other sources above the RPV lower head, and no other plant has identified PWSCC in the BMIs.

TOP

Pressurizer

Operating experience, both domestic and foreign, has demonstrated that Alloy 82/182/600 materials connected to a PWR's pressurizer may be particularly susceptible to PWSCC. Since th late 1980's, approximately 50 Alloy 600 pressurizer heater sleeves at Combustion Engineering-designed (CE-designed) facilities in the United States have shown evidence of RCPB leakage which has been attributed to PWSCC. The most recent events of this type occurred at Millstone Unit 2, and Waterford, Unit 3, in October 2003. All available evidence from finite element modeling studies and limited nondestructive evaluation (NDE) has suggested that these leakag events were the result of axially-oriented PWSCC of the pressure boundary portion of these heater sleeves. However, recent NDE results from Palo Verde, Unit 2, on heater sleeves which had not shown evidence of leakage have demonstrated that circumferentially-oriented PWSCC can occur in the non-pressure boundary portion (i.e., above the J-groove attachment weld) of these components.

Degradation attributed to PWSCC has also been observed in the pressurizer heater bundles usє in B&W-designed PWRs. The B&W-designed heater bundle employs a diaphragm plate manufactured from Alloy 600 and seal welded with Alloy 82/182, with structural support for th diaphragm plate being provided by a low alloy steel strongback which is bolted to the pressuriz shell. Most recently, in October 2003, pressure boundary leakage through a cracked diaphragr plate was observed at Three Mile Island, Unit 1 (TMI-1). The cracking in the TMI-1 diaphragm plate was attributed to PWSCC in the heat affected zone of the seal weld. Boric acid corrosion (the low alloy steel strongback was also observed to have resulted from the leakage.

Small diameter Alloy 82/182 instrument line penetrations have also shown evidence of PWSCC at many PWR facilities since the 1980's. For example, in October 2003, the Crystal River, Unit . licensee reported RCPB leakage from three pressurizer upper level instrument tap nozzles, whi are exposed to the steam space in the pressurizer. The leakage was attributed to PWSCC of Alloy 82/182/600 material from which the connections were constructed.

Finally, inspections conducted in September 2003 at Tsuruga, Unit 2, in Japan demonstrated that larger diameter, butt welded lines connected to the steam space of the pressurizer may al be susceptible to PWSCC. Evidence of boron deposits on the surface of a pressurizer relief valv nozzle (inside diameter 130 mm, or approximately 5 inches) lead to the discovery of five axiall oriented flaws in the Inconel alloy weld material used in the fabrication of the nozzle-to-safe er weld. Subsequent NDE performed on a safety valve nozzle of similar diameter resulted in the discovery of two additional flaws in its nozzle-to-safe end weld. Fractographic analysis of the flaw surfaces confirmed PWSCC as the mechanism for flaw initiation and growth.

Extensive operational experience with PWSCC in Alloy 82/182/600 materials used in the fabrication of pressurizer penetrations and steam space piping connections is not surprising. Tł initiation and growth of PWSCC flaws is known to be strongly dependent on the temperature oí the primary system water to which the Alloy 82/182/600 materials are exposed. Given the fact that at the pressurizer the reactor coolant system environment attains a temperature of about 650 F (343 C), PWSCC should be expected to occur in these materials and an effective degradation management program is warranted.

TOP

Outlook

The long-term goal for RPV Upper Head Issues is for the NRC to incorporate inspection requirements into 10 CFR 50.55a to ensure the integrity of the RPV head and VHP nozzles. It is preferred that the American Society for Mechanical Engineers (ASME Code) adopt acceptable requirements in Section XI of the Code. The NRC could then endorse the new Code requirements.

Regarding the future for other applications of Alloy 600 in PWRs, it is reasonable to expect that all parts, components and joints fabricated from Alloy 600 and weld filler metals Alloys 82 and 182 will continue to crack during operation. The longer term solution for many plants has been to seek replacement using Alloy 690 base metals and Alloy 52 and 152 for weld filler metals. However, the critical aspect of preventing this cracking from leading to challenges to plant safety systems will be the implementation of materials ageing management programs, includin effective inspection activities, to identify and remediate the cracking.

TOP

Tuesday, February 13, 2007

Fire Barriers

On this page:

Thermo-Lag Fire Barriers

Generic Letter (GL) 92-08, "Thermo-Lag 330-1 Fire Barriers," expressed the staff's concerns with Thermo-Lag 330-1 whicl included the lack of adequate reviews of the fire test results to determine if the tests were valid and the lack of adequate reviews to determine if the test results applied to plant designs. The NRC issued Supplement 1 to Generic Letter 86-10, "Fire Endurance Test Acceptance Criteria for Fire Barrier Systems Used to Separate Redundant Safe Shutdown Trains With the Same Fire Area," on March 25, 1994. Supplement 1 to Generic Letter 86-10, provided licensees with guidance for acceptance testing and equivalency evaluations for future tests of fire barrier systems.

Bulletins

Document Number	Description
BL92-01	Failure of Thermo-Lag 330 Fire barrier System to Maintain Cabling in Wide Cable Trays and Small Conduits Free From Damage, dated June 24, 1992.
BL92-01s1	Sup. 1 Failure of Thermo-Lag 330 Fire barrier System to Perform its Specified Fire Endurance Function dated August 28, 1992.

Generic Letters

Document Number	Description
GL86-10, Sup. 1	Fire Endurance Test Acceptance Criteria for Fire Barrier Systems Used to Separate Redundant Safe Shutdown Trains Within the Same Fire Area, dated March 25, 1994
GL92-08	Thermo-Lag 330-1 Fire Barriers, dated December 17, 1992

Information Notices

Document Number	Description

IN-91-47	Failure of Thermo-Lag Fire Barrier Material to Pass Fire Endurance Test, dated August 6, 1991.
IN-91-79	Deficiencies Found in Thermo-Lag Fire barrier Installation, dated December 6, 1991.
IN-91-79,Sup 1	Deficiencies Found in Thermo-Lag Fire barrier Installation, dated August 4, 1994.
IN-92-46	Thermo-Lag Fire Barrier Review Team Findings, Current Fire Endurance Tests, and Ampacity Calculation Errors, dated June 23, 1992.
IN-92-55	Current Fire Endurance Test Results for Thermo-Lag Fire Barrier Material, dated July 27, 1992.
IN-92-82	Results of Thermo-Lag 330-1 Combustibility Testing, dated December 15, 1992.
IN-94-22	*Fire Endurance and Ampacity Derating* Test Results for 3-Hour Fire Rated Thermo-Lag 330-1 Fire barriers, dated March 16, 1994.
IN-94-34	Thermo-Lag 330-660 Flexi-Blanket Ampacity Derating Concerns, dated May 13, 1994.
IN-94-86	Legal Actions Against Thermal Science, Inc., Manufacturer of Thermo-Lag, dated December 22, 1994.
IN-94-86, Sup 1	Legal Actions Against Thermal Science, Inc., Manufacturer of Thermo-Lag, dated November 15, 1995.
IN-95-27	NRC Review of Nuclear Energy Institute, Thermo-Lag *330-1 Combustibility Evaluation Methodology* Plant Screening Guide, dated May 31, 1995.
IN-95-32	Thermo-Lag 330-1 Flame Spread Test Results, dated August 10, 1995.
IN-95-49	Seismic Adequacy of Thermo-Lag Panels, dated October 27, 1995.
IN-95-49, Sup 1	Seismic Adequacy of Thermo-Lag Panels," dated December 10, 1997.

Hemyc/MT Fire Barriers

On September 20, 1995, the NRC staff documented its conclusion to the Commission that a broader scope of inspections would be needed to close out the Thermo-Lag Action Plan. Rather than developing a stand-alone Thermo-Lag fire barrier inspection program, a more robust program - the Fire Protection Functional Inspection (FPFI) was proposed. SECY 96-267 provides details of the proposed *FPFI. The FPFI included a review of safe shutdown design and licensing bases. It was thes* inspections, started in 1999, that identified the findings related to the Hemyc and MT fire barrier systems.

As a result of the FPFIs, and also the triennial fire protection inspections that followed the FPFIs, unresolved items (URIs) were opened at some nuclear power stations due to questions raised regarding the fire rating of the Hemyc and MT fire barrier materials. NRR concluded that the original testing (the Spanish Hemyc tests and the Southwest Research MT tests) was insufficient to qualify Hemyc for cable trays or conduits or MT for conduits as rated fire barriers. NRR documented this conclusion in Task Interface Agreement (TIA) response dated August 1, 2000, titled, "NRR Response to Task Interface Agreement (TIA) 99-028, Shearon Harris Nuclear Power Plant, Unit 1 -Resolution of Pilot Fire Protection Inspection Fire Barrier Qualification Issues (TAC No. MA7235)".

Based on the NRC's conclusion that the existing body of testing did not provide sufficient basis to qualify these fire barrier: the NRC chose to perform confirmatory tests on these materials using the criteria provided in Generic Letter 86-10, *Supplement 1. The purpose of these tests* was to determine Hemyc and MT's actual fire ratings to meet 10 CFR 50, Appendix R, "Fire Protection Program for Nuclear Facilities Operating Prior to January 1, 1979," Section III.G requirements 10 CFR 50.48, "Fire Protection, " requirements, and other regulatory commitments.

The Office of Research completed the testing in March 2005. The Office of Nuclear Reactor Regulation shared the test results with the licensees using Information Notice 2005-07 "Results of HEMYC Electrical Raceway Fire Barrier System Full Scale Fire Testing." Subsequently, the NRC issued GL 2006-03 in April 2006. This Generic Letter requested licensees to evaluate their *facilities to confirm compliance with the existing applicable regulatory requirements in light of the informati* provided in the GL and, if appropriate, to take additional actions.

All licensees have provided the information requested by GL 2006-03 about the adequacy of their Hemyc and MT fire barriers, as well as other fire barriers installed at their plants. The staff has reviewed the responses and closed out the generic letter for a large number of plants. The staff has issued requests for additional information and is in the process of closing out the generic letter for the remaining plants.

Some links on this page are to documents in our Agencywide Documents Access and Management System (ADAMS), and others are to documents in Adobe Portable Document Format (PDF). ADAMS documents are provided in either PDF or Tagged Image File Format (TIFF). To obtain free viewers for displaying these formats, see our Plugins, Viewers, and Other Tools page. If you have questions about search techniques or problems with viewing or printing documents from ADAMS, please contact the Public Document Room staff.

Generic Communications

Document Number	Description
GL06-03	Potentially Nonconforming Hemyc and MT Fire Barrier Configurations
IN-05-07	Results of HEMYC Electrical Raceway Fire Barrier System Full Scale Fire Testing

Other Documents

Date	Description
05/14/05	M. T. (3-Hour) Electrical Raceway Fire Barrier System Performance Testing Final Report.
05/03/05	Hemyc Test # 2 Post-Test Photographs
05/03/05	Hemyc Test # 1 Post-Test Photographs
05/03/05	Hemyc Photographs Used at Public Meeting
05/03/05	Memo - Public Availability of Post-Test Photographs MT Omega Point Laboratory (OPL) for Hemyc 1-Hour Fire Rated Electrical Raceway Fire Barrier System (ERFBS) Testing
04/27/05	Preliminary Pass/Fail Test Results for MT 3-Hour Rated Electrical Raceway Fire Barrier Systems.
04/21/05	Sandia National Laboratories Hemyc (1-Hour) ERFBS Test #2 Final Report Transmittal Letter
04/18/05	Performance Testing Cable Tray Cable Air Drop and Junction Raceways Final Report
04/15/05	Sandia National Laboratories Hemyc (1-Hour) ERFBS Performance Testing Final Report Transmittal Letter - Tes # 1
04/14/05	Pre-test Photographs by OPL #1
04/14/05	Pre-test Photographs by OPL #2
04/14/05	Public Availability of Pre-test Photographs by OMEGA Point Lab. (OPL) for HEMY 1-Hour Fire Rated Electrical Raceway Fire Barrier System (ERFBS) Testing
04/13/05	Materials Characterization for Siltemp and Refrasil.
04/11/05	Hemyc 1-Hour ERFBS Performance Testing Final Report
04/08/05	Plan for Hemyc (1-Hour) and M.T. (3-Hour) Electrical Raceway Fire Barrier System Testing Revision M
04/06/05	Notice of Public Meeting: Forthcoming Public meeting With Stakeholders Regarding Hemyc and MT Electrical Raceway Fire barrier System Performance
04/06/05	Test #1 Hemyc, Direct Attachment, Thermocouple Location Drawings and Test Data
04/01/05	Information Notice 2005-07: Results of Hemyc Electrical Raceway Fire Barrier System Full Scale Fire Testing
03/28/05	Preliminary Pass/Fail Test Results for Hemyc 1-Hour Rated Electrical Raceway Fire Barrier Systems
03/25/05	Guide to Pre-test Photographs by OPL #2
03/11/05	Guide to Pre-test Photographs by OPL #1
01/25/05	Letter from D. Lew to F. Emerson: NRC's response to NEI's comments on the Hemyc and M.T. test plan
12/21/04	Letter from F. Emerson of NEI to S. Weerakkody: Additional Comments on Testing the Hemyc and M.T. Fire Wrap Material
11/18/04	Letter from John Hannon to A. Marion of NEI: Hemyc (1-Hour) and MT (3-Hour) Fire Protection Wrap Performance Testing Fire Test Plan
1/16/03	Program Plan For Hemyc (1-Hour) and M.T. (3-Hour) Fire Protective Wrap Performance Testing, Final, dated January 16, 2003
12/06/02	Letter from A. Marion of NEI to John Hannon: NEI Comments on NRC Hemyc Test Plan
11/22/02	Summary of the October 31, 2002 Public Meeting on the Proposed Plan to Perform Fire Testing of Hemyc (1-Hour) and MT (3-Hour) Fire Protection Wrap
10/09/02	Notice of the Public Meeting on the Proposed Plan to Perform Fire Testing of Hemyc (1-Hour) and MT (3-Hour) Fire Protection Wrap
12/28/01	Letter from F. Emerson to D. Frumkin: NEI Provides Description of Installed Hemyc and MT Configurations

06/20/01	Promatec Hemyc and MT Electrical Raceway Fire Barrier Systems
04/25/01	NEI Document: Promatec 1-Hour and MT 3-Hour Fire Barrier Systems
04/17/01	Shearon Harris Nuclear Power Plant Unit No.1 - Docket No.50-4000: Licensing Basis of Promatec Hemyc Fire Barrier Systems
12/15/00	McGuire Fire Protection Inspection Report
11/29/00	Waterford Steam Electric Station, Unit 3; Fire Protection Report; Fire Protection
08/01/00	Harris Hemyc TIA - Letter from S. Black to L. Plisco
02/02/00	Shearon Harris Fire Protection Inspection Report

Kaowool and FP-60 Fire Barriers

Generic Communications

Document Number	Description
IN 93-41	"One Hour Fire Endurance Test Results for Thermal Ceramics Kaowool, 3M Company FS-195 and 3r Company Interam E-50 Fire Barrier Systems," dated May 28, 1993
IN 93-40	"Fire Endurance Test Results for Thermal Ceramics FP-60 Fire Barrier Material," dated May 26, 199:

Other Documents

Document Number	Description
SECY-99- 204	"Kaowool and FP-60 Fire Barriers," dated August 4, 1999

Other Fire Barriers

Document Number	Description
IN 97-59	"Fire Endurance Test Results of Versawrap Fire Barriers," dated August 1, 1997.
IN 95-52 Supp. 1	"Fire Endurance Test Results for Electrical Raceway Fire Barrier Systems Constructed From 3M Company Interam Fire Barrier Materials," dated March 17, 1998.
IN 95-52	"Fire Endurance Test Results for Electrical Raceway Fire Barrier Systems Constructed From 3M Company Interam Fire Barrier Materials," dated November 14, 1995.
IN 93-41	"One Hour Fire Endurance Test Results for Thermal Ceramics Kaowool, 3M Company FS-195 and 3m Company Interam E-50 Fire Barrier Systems," dated May 28, 1993.
GL86-10, Supp. 1	"Fire Endurance Test Acceptance Criteria for Fire Barrier Systems Used to Separate Redundant Safe Shutdown Trains Within the Same Fire Area," dated March 25, 1994.

Reactor Coolant System Weld Issues

The discovery, in October 2006, of five circumferential indications in three dissimilar metal (DM welds on the pressurizer at the Wolf Creek Generating Station (Wolf Creek) raised safety concerns based on the size and location of the indications. At Wolf Creek, three indications wer in the pressurizer surge nozzle-to-safe end weld, and two separate indications were in the safe and relief nozzle-to-safe end welds. These findings also indicated that significant concerns migl exist with the current inspection schedules and plans for addressing these pressurizer weld concerns.

The Nuclear Regulatory Commission (NRC) is concerned about the pressurizer surge nozzle-to-safe end weld indications, as this is the first time that multiple circumferential primary water stress-corrosion cracking (PWSCC) indications have been identified in a weld. This condition ca into question the degree of safety margin present in past structural integrity evaluations for DM welds susceptible to PWSCC, since multiple stress-corrosion cracking flaws may grow independently and ultimately grow together, significantly reducing the time from flaw initiation to leakage or rupture. The length of the relief nozzle-to-safe end flaw is also of concern, as this flaw was much longer than those analyzed previously in a small diameter nozzle.

The NRC communicated the need for near-term enhancements to the industry through public meetings held on November 30, 2006, December 20, 2006, and February 2, 2007. Licensees submitted letters voluntarily committing to enhanced inspection and leakage monitoring requirements. After teleconferences with specific licensees held between February 12 through February 23, 2007, the licensees submitted supplemental commitment letters addressing the NRC staff's concerns regarding inspection, compensatory actions, and reporting. The NRC is confirming these licensee actions and commitments through Confirmatory Action Letters, know as CALs.

In all, the NRC issued CALs to licensees of 40 Pressurized Water Reactor (PWR) plants. For 31 plants, the CALs confirmed their commitment to inspect the pressurizer surge, spray, safety, and relief nozzle welds by December 31, 2007, to implement enhanced reactor coolant system leakage monitoring until the inspections are complete, to repeat butt weld examinations every years until the welds are either removed from service or mitigated, and to report inspection results to NRR. Nine plants desired to perform the inspections during their scheduled Spring 2008 outages, as they had no other outages scheduled between the time the CALs were issued and the end of 2007. For these nine plants, the CAL confirmed that these plants would accelerate outages into 2007 if the finite element analyses (FEA) being developed by industry did not demonstrate an adequate level of safety to the NRC. The industry has provided the FEA and documented it as Materials Reliability Program document MRP-216, Rev. 1, "Materials Reliability Program: Advanced FEA Evaluation of Growth of Postulated Circumferential PWSCC Flaws in Pressurizer Nozzle Dissimilar Metal Welds." The staff has reviewed the FEA and concluded in a safety assessment that the results are acceptable. Licensees for the nine plants have provided letters to the NRC staff confirming that the analyses bound their plant and their desire to inspect the welds during their scheduled Spring 2008 refueling outages. Based on the CALs, the NRC staff is issuing evaluation letters informing the nine licensees of its conclusion regarding continued operation of the plants. (Both the licensee letters and the NRC staff's evaluation letters can be accessed via the "Confirmatory Action Letters" link above.)

On October 7, 2000, during a containment inspection after entering a refueling outage, the licensee for V.C Summer Nuclear Power Station identified a circumferential indication in the firs weld between the reactor vessel nozzle and the "A" loop hot leg piping, approximately 3 feet from the reactor vessel. The NRC formed a Special Inspection Team to determine the adequacy of the licensee's previous inspection, confirm that the licensee had completed an analysis and examination to determine the root cause, and to review the overall corrective action plan and the extent of conditions.

The NRC team identified potentially generic issues involving limitations of required non-destructive examinations to detect certain small inside diameter stress corrosion cracks, and th potential for multiple weld repairs to result in high residual stresses which can contribute to stress corrosion cracking. The following links include information on the initial finding and the subsequent NRC and MRP activities on reactor coolant pressure boundary butt welds.

- News and Correspondence
- Public Meetings

Backgrounder on Nuclear Power Plant Fire Protection

Some links on this page are to documents in our Agencywide Documents Access and Management System (ADAMS), and others are to documents in Adobe Portable Document Format (PDF). ADAMS documents are provided in either PDF or Tagged Image File Format (TIFF). To obtain free viewers for displaying these formats, see our Plugins, Viewers, and Other Tools page. If you have questions about search techniques or problems with viewing or printing documents from ADAMS, please contact the Public Document Room staff.

Background

On March 22, 1975, a fire at the Browns Ferry Nuclear Power Plant fundamentally changed the concept of fire protection and associated regulatory requirements for U.S. nuclear power plants. Plant workers were fixing leaks in the cable spreading room outside the reactor building. The workers used a candle to test seals for air leaks into the reactor building. The polyurethane foam seal, however, was not fire-rated. The flame from the candle ignited both the seal and the electric cables that passed through it.

By the time firefighters extinguished the fire, it had burned for almost 7 hours. More than 1600 electrical cables were affected, 628 of which were important to plant safety. The fire damaged electrical power, control systems, and instrumentation cables and impaired cooling systems for the reactor. Operators could not monitor the plant normally and had to perform emergency repairs on systems needed to shut the reactor down safely.

Investigations after the fire revealed deficiencies in the design of fire protection features at nuclear power plants and in th plant procedures for responding to a fire. Fire insurance companies, normally concerned with occupant safety and propert protection, did not sufficiently consider nuclear safety issues. A fire in certain locations at a nuclear plant could cause redundant safety systems and components to fail, making it difficult to shut the reactor down safely.

Since the Browns Ferry incident, no fire at a U.S. commercial nuclear power plants has affected the safe operation of a reactor.

Fire Protection Regulations

After the Browns Ferry fire, the Nuclear Regulatory Commission revised its fire protection regulations to reduce the chance of a fire starting and the consequences should a fire occur. The regulations' bottom line is that each licensee must maintai the ability to shut down the reactor safely in the event of a fire. The objectives of NRC's fire protection regulations ensure this ability by:

1. minimizing the potential for fires and explosions;
2. rapidly detecting, controlling, and extinguishing fires that do occur; and
3. ensuring that operators can shut down the reactor safely despite a fire, and minimize the risk of significant radioactive releases to the environment.

Nuclear power plants today use redundant methods of fire protection to keep fires from damaging plant safety systems. Some of these methods include fire barriers such as insulation, fire detection systems, and fire suppression systems (such as sprinklers). If a required element of fire protection is not available, the licensee must compensate for it, often by placin dedicated personnel on a continuous fire watch. The NRC regularly inspects licensees' means of achieving and maintaining the safe shutdown of the reactor in the event of a fire.

Deterministic fire protection requirements help keep nuclear power plants safe by ensuring that systems for shutting the reactor down safely will survive a fire. These requirements, based on a set of possible serious fires, were developed before the staff or the industry had experience with probabilistic risk assessments (PRAs) for fires. Deterministic requirements als do not take advantage of recent advances in performance-based analysis methods such as fire modeling.

Risk-informed regulations consider the safety significance of requirements and ensure the requirements' burden on licensees is appropriate to the safety level they provide. Performance-based regulations rely on a required outcome rather than requiring a specific process or technique.

The NRC approved one such risk-informed performance-based alternative in July 2004 that allows licensees to focus their fire protection activities on the areas of greatest risk. The agency enacted rule 50.48(c), which endorsed National Fire Protection Association (NFPA) Standard 805, "Performance-Based Standard for Fire Protection for Light-Water Reactor Electric Generating Plants, 2001 Edition", with some exceptions. To help licensees implement NFPA 805, NRC staff issued Regulatory Guide 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants The guide also endorsed the related Nuclear Energy Institute document NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48(c)"

Discussion

The NRC continues to oversee fire protection at nuclear power plants through inspection and oversight. The NRC also work with international codes and standards organizations and nuclear industry representatives to resolve the generic fire protection concerns discussed below.

Operator Manual Actions

Licensees are required to protect plant equipment necessary for safe shutdown using a combination of physical separation barriers, and methods to detect and control or extinguish fires. In certain cases at specific plants, the NRC has also reviewed and approved operator manual actions as another acceptable method to safely shut down the plant in the event a fire. An example would be manually opening a valve to prevent it from closing improperly during a fire.

However, some licensees rely on operator manual actions that have not been reviewed and approved by the NRC to mitigate fires in fire areas with redundant safety trains (commonly referred to as III.G.2 areas since Section III.G.2 of Appendix R to 10 CFR 50 provides the requirements) , which could have led to a very large number of requests for exemptions to the regulations. NRC staff proposed a rule that would enable the licensee to demonstrate the acceptability (any manual actions used to safely shut down a plant in the event of a fire. The key concept of that rule was that plant personnel should not face significant hazards in carrying out feasible, reliable manual actions to ensure safe shutdown of the plant. The rule's primary objective was improving efficiency by minimizing the number of exemption requests.

The public comment period for the proposed rule ended in May 2005, and industry comments indicated the NRC would receive a very large number of exemption requests under the proposed rule, primarily because the staff emphasized that fire detection and suppression requirements are essential when licensees rely on operator manual actions in III.G.2 fire areas. The staff concluded this would defeat the rule's objective, and therefore requested the proposed rule be withdrawn. The Commission approved that request, and the staff continues to enforce existing requirements through the Reactor Oversight Process. The expectations endorsed by the Commission with respect to operator manual actions are documente in Regulatory Issue Summary 2006-10.

Implementing Risk-Informed, Performance-Based Fire Protection

The Commission approved the 50.48(c) rule in May 2004, and published the rule in June. It took effect in July. The Commission also allowed the staff to use its discretion in enforcing certain fire protection issues for plants transitioning to the new rule. The enforcement discretion provided an incentive for licensees to adopt NFPA 805. Subsequently, by the enc of February 2006, operators of 42 reactors had sent letters of intent indicating their commitment to adopt the voluntary standard.

Two nuclear stations, Oconee and Shearon Harris, volunteered to be pilot plants for the transition to NFPA 805. Consequently, the staff kicked off the pilot implementation in August 2005. The staff has conducted Observation Visits at Duke Power and Progress Energy in November 2005, March 2006, October 2006, November 2006, March 2007, and May 2007. More observations are planned at these plant sites. To aid plants in their transition to the new rule, NRC staff and industry developed a Frequently Asked Questions program.

Risk Insights for Electrical Circuit Inspections

In the past, inspectors discovered electrical circuits at particular plants that, if damaged by fire, could prevent critical equipment from working properly. In 2004, the NRC issued guidelines to allow inspectors to take relative risk into account when inspecting electrical circuits that are needed to shut the reactor down safely. In addition, the staff proposed a draft Generic Letter to clarify compliance expectations with respect to the issue of multiple false actuations in October 2005. In SRM-SECY-06-0196, the Commission disapproved the issuance of the proposed Generic Letter and directed the staff to work with stakeholders to develop a solution to address the circuit issue. The NRC continues to work with regional inspectors and industry representatives to clarify and apply risk insights to regulatory requirements for analyzing circuits needed to shut down a reactor after a fire.

Significance Determination Process (SDP)

The NRC has revised its process for evaluating the significance of fire protection deficiencies found during inspections. This process is based on a simplified fire probabilistic risk assessment (PRA). NRC inspectors can easily use the SDP to obtain a assessment of the risk significance of a finding. The NRC has also developed quantitative fire hazards analysis methods, in NUREG-1805, for the fire protection significance determination process.

Hemyc Fire Barrier Qualification

Full-scale fire tests recently performed by the NRC revealed that Hemyc, a fire barrier system used to protect cables in electrical raceways in nuclear power plants, does not perform as designed. The outer covering of the barrier can shrink during a fire, opening joints in the material and potentially allowing the fire to damage cables inside. These results show that Hemyc does not serve as a fire barrier for the full hour required.

The agency has contacted the licensees of those 11 plants using Hemyc to inform them about the test results so that appropriate compensatory actions can be taken. Since the material is designed for use in areas with both fire detection an automatic fire suppression systems, the NRC remains assured that those plants' overall fire protection scheme will ensure safe shutdown of a reactor in case of fire. The NRC issued Generic Letter 2006-03 in April 2006 to ensure that the affectec licensees take appropriate corrective actions.

All licensees have provided the information requested by GL 2006-03 about the adequacy of their Hemyc and MT fire barriers, as well as other fire barriers installed at their plants. The staff has reviewed the responses and closed out the generic letter for several plants. The staff has issued requests for additional information for the remaining plants.

Source Documents

Documents related to fire protection are available on the NRC's Web site pertaining to Regulations, Guidance, and Communications. The NRC's regulations for nuclear power plants can be found in Title 10 of the Code of Federal Regulatio (10 CFR), available on the NRC's Web site. Fire protection regulations are detailed in Part 50.48 of 10 CFR and Appendices A and R to Part 50.

Safety and Security Improvements at Nuclear Plants

Post 9-11 Actions

The Nuclear Regulatory Commission (NRC) - responsible for protecting public health and the environment from potential hazards involved in using nuclear materials - took prompt action to enhance safety and security, and has comprehensively re-evaluated security at nuclear power plants and other facilities it regulates.

Since September 11, 2001, NRC has strengthened security at nuclear facilities by working with national experts using state-of-the-art structural and fire analyses to realistically predict the consequences of terrorist acts. These studies confirm that, given robust plant designs and the additional enhancements to safety, security, and emergency preparedness and response, it is unlikely that significant radiological consequences would result from a wide range of terrorist attacks, including one from a large commercial aircraft.

Actions taken by Federal aviation safety and security agencies - Federal Air Marshals, reinforced cockpit doors, airport passenger and baggage screening, improved ability to detect deviation from planned flight paths and greater military aircraft intercept capability - have reduced the likelihood that large commercial aircraft could be used to attack critical infrastructure, including a nuclear facility. Other actions, such as improved communication between military surveillance authorities, NRC, and its licensees, would allow plant operators to prepare the plant for safe shutdown should it be necessary. These actions, coupled with those taken by the NRC and the nuclear industry, are an integral part of the government's overall strategy for protecting the nation's critical infrastructure.

NRC has strengthened requirements at nuclear power plants and enhanced coordination with Federal, State and local organizations since 9-11

NRC major actions include:

- Ordered plant owners to sharply increase physical security programs to defend against a more challenging adversarial threat;
- Required more restrictive site access controls for all personnel;
- Enhanced communication and liaison with the Intelligence Community;
- Ordered plant owners to improve their capability to respond to events involving explosions or fires;

- With mitigation strategies and measures in place, the probability of damaging the reactor core and releasing radioactivity that could affect public health and safety is low;
- Significant releases due to a terrorist attack on a spent fuel pool are very unlikely;
- It is highly unlikely that a significant release of radioactivity would occur from a dry spent fuel storage cask; and
- No release of radioactive material is expected from an aircraft attack on a transportation cask.

Time is available to protect the public in unlikely event of a radiation release

- If a radiation release did occur, there would be time to implement mitigating actions and offsite emergency plans at power plants, spent fuel pools, and dry-cask storage installations; and
- Safety and security studies confirm that NRC's emergency planning basis remains valid.

NRC has taken action to strengthen security and safety
Increased aviation security and aggressive NRC action provide enhanced protection against terrorist attacks

Other Security Information can be found at:

- Dirty Bombs
- Force-on-Force Security Exercises
- Force-on-Force Exercises at Nuclear Power Plants
- Safety and Security Improvements at Nuclear Plants

February 2005