October 28, 1998

FOR: The Commissioners

FROM: William D. Travers /s/
Executive Director for Operations

SUBJECT: PROPOSED GENERIC LETTER 98-XX "STEAM GENERATOR TUBE INTEGRITY"

PURPOSE:

This paper informs the Commission of the staff's intent to delay the issuance for public comment of proposed Generic Letter 98-xx, "Steam Generator Tube Integrity," while the staff works with the Nuclear Energy Institute (NEI) and the nuclear power industry to resolve concerns with the industry initiative entitled NEI 97-06, "Steam Generator Program Guidelines." It is the staff's objective to avoid having to issue a generic letter and instead to endorse the industry initiative as an acceptable approach to resolving current problems associated with steam generator (SG) tube integrity. Nonetheless, the staff intends to release now for public comment the following three documents: (1) Draft Regulatory Guide DG-1074, Steam Generator Tube Integrity; (2) Differing Professional Opinion (DPO) Consideration Document; and (3) Memorandum dated September 25, 1998, to the Commission from Joram Hopenfeld, the DPO submitter.

BACKGROUND:

In COMSECY-97-013, the staff informed the Commission that it would (1) develop a generic letter containing model technical specifications (TSs) for SG tube surveillance and maintenance that requests licensees to address problems with current TSs; (2) develop guidance to support implementation of the generic letter model TSs; (3) give licensees the option to pursue alternate SG tube repair criteria supported by an appropriate risk assessment; and (4) evaluate, as part of the individual plant examination followup program, plants that appear to have a higher potential for core damage sequences that can challenge SG tubes. The staff also stated that the proposed generic letter package being issued for public comment would contain an assessment of issues raised in a differing professional opinion (DPO) concerning SG tube integrity. In the staff requirements memorandum dated June 30, 1997, the Commission approved the revised regulatory approach. The staff was committed to provide the proposed generic letter package to the Commission by September 30, 1998.

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DISCUSSION:

In support of this commitment, the staff developed a proposed generic letter that (1) informs pressurized-water reactor (PWR) licensees that plant TSs for maintaining SG tube integrity do not alone provide the needed assurance that SG tube integrity is being adequately monitored and maintained in accordance with NRC regulations and plant licensing bases; (2) advises licensees that they may request license amendments to their plant TSs to implement the model TSs attached to the generic letter for maintaining SG tube integrity, or justify alternate approaches for ensuring that SG tube integrity is monitored and maintained consistent with applicable regulatory requirements and plant licensing bases; and (3) requires that licensees submit to the NRC written responses that describe their ongoing or planned activities to monitor and maintain SG tube integrity consistent with NRC regulations and plant licensing bases, along with supporting safety bases for the plant-specific approaches. The staff also developed draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity" (Attachment 1) and the DPO Consideration Document (Attachment 2). In support of issuance of the proposed generic letter for public comment, the staff has received endorsement from the Advisory Committee on Reactor Safeguards and met with the Committee To Review Generic Requirements in June and July 1998 to discuss the package.

By letter dated December 16, 1997, the NRC staff was informed that the industry, through the NEI Nuclear Strategic Issues Advisory Committee, had voted to adopt NEI 97-06. The chief objective of the industry initiative is for PWR licensees to evaluate their existing SG programs and, where necessary, to revise or strengthen program attributes to meet the intent of the NEI 97-06 guidelines. The NEI 97-06 guidelines are intended to improve both the quality and the consistency of SG programs throughout the industry. The NEI 97-06 initiative is a higher tiered document that commits PWR licensees to a programmatic approach conceptually similar to that recommended by DG-1074. NEI 97-06 references two types of lower tiered documents for guidance on the implementation of individual programmatic features: Electric Power Research Institute (EPRI) guidelines which are directive in nature (the licensee must meet the intent of directives), and EPRI guidelines which are non-directive in nature (may be used by the licensee as general guidance). The December 1997 NEI letter indicated that the NEI initiative would be implemented no later than the first refueling outage after January 1, 1999.

Throughout the development of the proposed generic letter and its predecessor, the draft SG rule, as well as during plant-specific, ad hoc reviews, the staff and the industry have interacted extensively on the development of SG guidance. Both the staff and the industry have incorporated portions of each other's guidance into their respective guidance documents as part of this technical interchange. The focus of the technical interchange has been to identify where additional actions (beyond the minimum required by current TSs) are needed to ensure SG tube integrity consistent with governing regulations and plant licensing bases. During this process, the staff assembled draft DG-1074, which provides one acceptable means for complying with the governing regulations and plant licensing bases to ensure tube integrity. At the same time, and in response to the staff's ongoing regulatory development effort, the industry focused its efforts on improving existing SG inspection guidance and developing new guidance. Two examples are the PWR Steam Generator Examination Guidelines now in Revision 5 and the PWR Primary-to-Secondary Leak Guidelines (both of these documents are directive guidelines as defined in NEI 97-06). The industry's efforts to improve industry guidance culminated in the NEI 97-06 initiative previously described.

Currently, technical differences still remain between the industry and the staff, as well as issues regarding the appropriate regulatory framework for implementing the NEI guidelines. However, consistent with Direction Setting Issue (DSI) 13, the staff's preferred approach is to endorse an industry initiative that addresses all staff and stakeholder concerns rather than issue a generic letter. The staff is working with industry to resolve issues associated with NEI 97-06. Accordingly, the staff proposes to delay issuance of the proposed generic letter while it meets with industry in order to try and resolve staff and any stakeholder concerns, with the objective of being able to endorse NEI 97-06 in a regulatory guide. The staff expects to determine whether the generic letter effort should be reactivated by January 1999 based on the progress in resolving remaining concerns with the industry.

The staff has concluded that this approach makes the best use of available staff resources, enables the staff to monitor the industry's effort to complete the guidance supporting NEI 97-06 and to implement the initiative, gives appropriate credit to the industry initiative, and is consistent with DSI 13. To provide a basis for the discussion of technical issues with interested stakeholders and the industry, the staff continues to support issuance of DG-1074 and the DPO consideration document for public comment.

In order to inform the Commission of his continued concerns about steam generator tube integrity, the DPO submittee prepared the document that has been included as Attachment 3. Since this document will also provide a basis for the discussion on technical issues, it will be provided for public comment. Attachment 3 was prepared after staff completion of Attachment 2. Attachment 2 has not been reviewed for any changes that might be necessary.

In COMSECY-97-013, the staff indicated that it would develop an application specific regulatory guide containing guidance for assessing the changes in risk associated with proposed relaxations to SG tube repair criteria. Accordingly, the staff is developing draft regulatory guide DG-1073 "An Approach for Plant-Specific, Risk-Informed Decision Making: Steam Generator Tube Integrity" and intends to issue it for public comment after the staff makes its final decision on the need for a generic letter on SG tube integrity and whether industry guidance supporting NEI 97-06 will adequately address risk.

RECOMMENDATION:

The staff intends to delay issuance of the proposed generic letter package while it works with industry to reach agreement on NEI 97-06. The staff will reassess the need for issuance of the proposed generic letter on the basis of the progress in resolving concerns with NEI 97-06. The staff intends to issue for public comment the attached DG-1074, the DPO consideration document, and the memorandum to the Commission from Hopenfeld to provide a basis for the discussion of technical issues with interested stakeholders and the industry.

The staff requests action within 10 days. Action will not be taken until the Staff Requirements Memorandum is received. We consider this action to be within the delegated authority of the Executive Director for Operations.

William D. Travers
Executive Director for Operations

Attachments:
1. Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity"
2. Differing Professional Opinion Consideration Document
3. Memorandum to the Commission from Joram Hopenfeld, "J. Hopenfeld's Differing Professional Opinion Concerning Voltage-based Repair Criteria for Steam Generator Tubes:Release of DPO Consideration Document for Public Comment," dated September 25, 1998

ATTACHMENT 2

DIFFERING PROFESSIONAL OPINION
CONSIDERATION DOCUMENT

I. INTRODUCTION AND EXECUTIVE SUMMARY

The staff has considered the differing professional opinion (DPO) views, as they have been presented, during past regulatory activities such as development of Generic Letter (GL) 95-05 and in more recent development of the regulatory framework as described in SECY-97-013 and proposed in draft GL 98-xx "Steam Generator Tube Integrity" (currently being delayed for three months while the staff works with industry on resolving concerns associated with NEI 97-06). The purpose of this document is to provide an integrated report on how the DPO issues have been considered in ongoing regulatory activities and how they have been addressed in development of the proposed regulatory framework for steam generators (SG). In preparing this document, the staff reviewed all available documentation discussing the DPO (see section IV list of references) and grouped the concerns into five broad issues[1]: (1) limitations of nondestructive examination (NDE) methods, (2) primary-to-secondary SG tube leakage during postulated main steam line break (MSLB) conditions, (3) increased risk due to SG tube degradation and implementation of alternate repair criteria, (4) iodine spiking assumptions for radiological analyses and (5) SG tube integrity under severe accident conditions. By letter dated April 9, 1998 from the deputy EDO to the DPO author, the DPO author was requested to provide any additional concerns that he may have regarding the DPO consideration document (this document), or the proposed GL package. In a memo dated April 21, 1998, the DPO author indicated that he did not have any new concerns regarding this document. Some specific comments were provided on the proposed GL and DG-1074. The additional comments will be addressed as part of the public comment process.

It should also be noted that the staff is currently working with industry to resolve issues associated with the industry initiative entitled NEI 97-06 "Steam Generator Program Guidelines." As a result, the staff may not issue a generic letter on SG tube integrity if sufficient progress is achieved with industry in resolving staff concerns. Regardless of whether the staff issues a generic letter on SG tube integrity or chooses to endorse the industry initiative, the revised regulatory framework will still address the significant issues associated with maintaining and monitoring SG tube integrity consistent with governing regulations and plant licensing bases. In the mean time, the staff continues its ad hoc approach to monitoring licensees actions in this regard.

The staff's assessment of each of the above issues is discussed in the following sections. It should be noted that some elements of the issues raised have merit and the staff was, in fact, addressing many of these issues before they were raised in various stages of the DPO.

The subject of limitations in NDE was specifically evaluated in NUREG-1477 and in the methodology developed in GL 95-05. Further, a major component of any acceptable revised regulatory approach (whether that ultimately is a generic letter or the NEI initiative) will be on providing the necessary guidance on how to assure that NDE methods are properly qualified and NDE uncertainties accounted for in assessing tube integrity.

The issue of primary-to-secondary SG tube leakage under postulated design basis conditions (e.g., MSLB conditions) was (1) specifically evaluated in NUREG-1477, (2) is addressed by the analysis methods developed in GL 95-05, and (3) is specifically controlled by one of the performance criteria established under the new regulatory framework (note that the performance criteria may be implemented via the NEI 97-06 initiative instead of being implemented via a generic letter).

The risk assessment performed by the staff in support of the effort to revise the regulatory framework considered the issues raised in the DPO. A major conclusion of the risk assessment and associated regulatory impact analysis was that a generic backfit requiring licensees to take action to reduce risk associated with SG tube degradation could not be supported per the criteria of 10 CFR 50.109. However, the risk evaluation also concluded that (1) the conclusion regarding inability to support a generic backfit was based on licensees taking action beyond those required by the current technical specifications, (2) further assessments are necessary to determine if plant-specific vulnerabilities require additional action, and (3) certain forms of alternate repair criteria could potentially have an adverse impact on risk. It is the staff's intent to include assessment of plant-specific vulnerabilities as part of the individual plant examination (IPE) followup program. Also, since some forms of alternate repair criteria could introduce new vulnerabilities and contributions to risk, under any acceptable regulatory approach (whether a generic letter or the NEI 97-06 initiative) the staff will encourage licensees to follow risk-informed approaches when proposing new ARC as an alternative to compliance with existing deterministic requirements. The staff will not approve new ARC that unacceptably increase the risk associated with SG tube integrity .

As part of assessing the issue of iodine spiking, the staff has reviewed the industry models and performed independent analyses of available data and models. Based on these assessments, the staff has concluded that although there are no data corresponding directly to the rapid depressurization conditions associated with a postulated MSLB, the spiking factors assumed in the existing dose assessment methodologies provide a reasonable level of conservatism for calculation of doses against the 10 CFR Part 100 guideline limits.

Regarding severe accidents, the staff has concluded that core damage conditions, particularly those associated with high primary pressure, dry steam generator secondary side events, can introduce vulnerabilities that have not been previously considered. These vulnerabilities are the result of challenges to tube integrity from the high reactor coolant system (RCS) temperatures predicted during these events. The staff considered high temperature effects in its risk assessment and also assessed the potential for plant-specific vulnerabilities due to particular forms of degradation. As a result of these assessments, the staff has concluded that certain vulnerabilities will be considered along with results of the previously mentioned IPE followup program.

II. SUMMARY OF ISSUES RAISED

Based on a review of all available documentation discussing the DPO issues (the documentation reviewed is referenced at the end of the main body of this report), the concerns raised are summarized into the following five issues:

1. **NDE Issue:** The concern was identified that nondestructive examination (NDE) techniques are not capable of adequately detecting and sizing intergranular stress corrosion cracking (IGSCC) and that correlations of leakage versus an NDE parameter such as bobbin coil voltage cannot be reliably used to calculate leakage for design basis events such as a main steam line break (MSLB). Additionally, it was stated that the complex morphology of IGSCC cracks and the limitations of current NDE technology make it impossible to construct such correlations.

2. **MSLB Leakage Issue:** The concern is that elevated tube differential pressure caused by design basis secondary depressurization transients (including MSLB) can cause primary-to-secondary leakage that could be greater than the leakage from a steam generator tube rupture (SGTR). The leakage may be sufficient to deplete the refueling water storage tank (RWST) inventory via ECCS injection lost to the secondary side of the SGs (and therefore not available for recirculation from the containment sump) thereby leading to core damage.

3. **Risk Increase Issue:** The concern is that the frequency of core damage with containment bypass may be approximately 3.4 X 10-4 per reactor year. This risk value was initially attributed to the increased risk resulting from the increased potential for RWST depletion as a result of large postulated primary-to-secondary tube leakages as discussed in issue 2 above. More recently, this concern has included references to station blackout sequences, and it has been implied that severe accident leakage and failures of degraded tubes under such conditions could lead to higher risk.

4. **Iodine Spiking Issue:** The concern is that the iodine spike (i.e., the release of radioactive fission products into the RCS from the fuel through

cladding perforations) following a large depressurization transient such as a MSLB, may be greater than the value of 500 assumed in the standard review plan (SRP) dose assessment methodology. Additionally, the concern is raised that the iodine spike (i.e., 500) could increase to higher multipliers as initial RCS coolant activity is decreased so that simply reducing initial RCS activity (via a technical specification change) may not result in a one-for-one reduction in calculated accident doses.

5. **Severe Accident Issues:** The concern is that the SG tubes may fail prior to other portions of the reactor coolant pressure boundary (RCPB) due to (1) inadequate NDE characterization (leaving in service potentially large numbers of through-wall flaws or flaws that grow through-wall during the operating cycle and which subsequently fail under severe accident conditions prior to other portions of the RCPB), (2) increased flow through the tube cracks (as small as pin-hole leaks) resulting in increased heat transfer to the tubes and a change in the thermal-hydraulic regime analyzed during this portion of the severe accident, (3) the cracks in tubes opening and unplugging due to increased pressure, and (4) the potential for jets from the cracks to erode and fail adjacent tubes leading to a large release.

III. CONSIDERATION OF DPO ISSUES IN SG GENERIC LETTER DEVELOPMENT

The staff considered each of the preceding issues during the effort to revise the regulatory approach on SG tube integrity including the development of supporting regulatory guidance and performance of a supporting risk assessment. The following sections discuss the staff's consideration of each issue.

Response to Issue 1. "NDE Issue":

Regarding whether current NDE techniques are capable of detecting and sizing IGSCC, it is recognized that current NDE systems (equipment, procedures, and personnel) have limitations with respect to their capabilities to accurately detect and size flaws, particularly IGSCC flaws. Industry and the NRC staff have been aware of these limitations and have taken action as necessary to ensure tube integrity is maintained.

Technology improvements have significantly increased eddy current testing (ECT) sensitivity to IGSCC. Industry guidelines have been developed which have gained widespread acceptance among licensees and which have led to improved procedures and improved qualification of the equipment and procedures and improved training and qualification of personnel with respect to IGSCC detection. The improved ECT technology, procedures, and qualification exceed minimum technical specification requirements, but have received widespread application throughout the industry.

Licensees have performed extensive metallographic examinations of IGSCC degraded tube specimens removed from the field and in-situ pressure tests to verify that ECT detection performance in conjunction with other licensee actions is adequate to ensure that tube integrity is being maintained. ECT capabilities to accurately measure IGSCC depth continues to be poor and, thus, it is general industry practice to conservatively assume that all IGSCC indications fail to satisfy the 40% depth-based plugging limit, where applicable, and to plug all such indications (i.e., "plug on detection"). In addition, licensees have implemented more frequent inspections, reduced hot leg temperatures, more effective monitoring of operational leakage, and more stringent operational leakage limits, beyond minimum TS requirements, as necessary to ensure tube integrity is maintained between scheduled inspections.

The accumulated evidence from operating experience, including metallographic examinations of removed tube specimens from the field and in situ pressure tests, support the conclusion that ECT detection capability for IGSCC in conjunction with implementation of other measures (discussed above), when and as appropriate, is adequate to ensure that tube integrity is maintained. Licensee actions to manage IGSCC to this effect have been somewhat ad-hoc, in view of shortcomings in existing TS requirements. The effort to revise the regulatory approach on SG tube integrity has as its key objective identifying those additional actions that are needed (in addition to the current minimum TS requirements) to maintain and monitor SG tube integrity consistent with governing regulations and plant licensing bases. In the meantime, the staff monitors licensee actions to ensure that tube integrity is being maintained. In addition, the staff issues generic communications as warranted in response to problems experienced in the field. Recent examples of generic letters addressing problems in the field include GL 95-03 "Circumferential Cracking of Steam Generator Tubes", GL 97-05 "Steam Generator Tube Inspection Techniques", and GL 97-06 "Degradation of Steam Generator Internals".

Regarding the development and use of empirical correlations for the calculation of leakage during design basis events, the staff has approved the use of one such approach to date (i.e., empirical correlations for the calculation of both accident-induced leakage and the potential for tube failure under design basis conditions) through the issuance of GL 95-05 (voltage-based repair criteria for outside diameter stress corrosion cracking (ODSCC) at tube support plates (TSPs) in Westinghouse-designed SGs). It was GL 95-05 which led the DPO author to file the DPO in July 1994. The DPO issues are related to issues identified earlier in a differing professional view (December 1991). For the voltage-based repair criteria, the DPO issues were considered by the staff during the development of the GL 95-05 guidance, as well as being considered by Advisory Committee for Reactor Safeguards (ACRS) and the Committee to Review Generic Requirements (CRGR) during the review process for the GL. The staff, ACRS, and CRGR all concluded that GL 95-05 was acceptable in light of the issues raised and as a result the GL was issued. As a result, to date, this issue has been addressed appropriately. It should be noted that GL 95-05 explicitly addressed the need to conservatively consider, through rigorous statistical analyses, the uncertainties associated with voltage-based correlations and required licensees to explicitly incorporate into the assessment model consideration of the probability of nondetection.

Ten plants currently (March 1998) implement a voltage-based steam generator (SG) tube repair criteria per Generic Letter (GL) 95-05. At this time, the staff has completed eight reviews of the 90-day reports submitted by licensees (five of the reviews were performed by Pacific Northwest National Laboratories (PNNL); the remaining three were performed by EMCB staff). Copies of the evaluations that document these reviews have been provided to the author of the DPO. Several other 90-day reports are currently being reviewed. Table 1 summarizes the status of the 90-day report reviews for the plants implementing voltage-based tube repair criteria.

Based on the reviews of 90-day reports, the GL 95-05 methodology for most of the plants has reasonably predicted the end-of-cycle (EOC) voltage distribution of outside diameter stress corrosion cracking (ODSCC) indications located at the tube support plates (TSPs). Predictions of EOC voltage distributions that are reasonably comparable to actual inspection results have led to conservative predictions of leakage during a postulated main steam line break (MSLB) event. However, recent experiences at Braidwood and Farley warrant further staff review and assessment.

Braidwood Unit 1 (Braidwood-1)

Braidwood Unit 1 and Byron Unit 1 have plant-specific license amendments that allow higher voltage-based repair criteria than allowed under the 95-05 methodology. These higher voltage repair limits were based on stabilization of the tube support plates by expansion of selected tubes at the support plate intersections so that the tube support plates could be credited as staying in place under postulated main steam line break conditions (see safety evaluation in letter from M.D. Lynch (NRC) to D.L. Farrar (Commonwealth Edison) dated November 9, 1995). At the EOC-6 at Braidwood-1, the licensee compared the actual bobbin coil eddy current voltage distribution of its ODSCC TSP indications with the projected voltage distribution obtained during the previous outage. The licensee found the predicted voltage distribution was nonconservative with respect to the actual voltage distribution. As a result, the predicted leakage during a postulated MSLB event significantly underestimated the leakage calculated based on actual EOC conditions (6.99 gpm versus 11.5 gpm) although the latter value was still well within the site allowable leakage limit (19.0 gpm). The licensee attributed the nonconservative prediction of the EOC voltage distribution to a voltage-dependent growth rate that appears to have occurred, in part, due to the higher voltage repair criteria in effect at Braidwood-1. For the current operating cycle (cycle 7), the licensee applied a voltage-dependent growth rate to predict the EOC-7 conditions. The licensee also reassessed the EOC-8 predictions for Byron-1 since a similar voltage-dependent growth rate may exist (Byron-1 also implements higher voltage repair criteria). Based on EOC predictions that incorporated voltage-dependent growth rates, both plants concluded the reactor coolant system dose equivalent iodine (DEI) levels should be reduced. The staff will review the license amendment request to reduce DEI in the coming weeks.

The staff is reviewing the most recent Braidwood-1 90-day report. The review will focus on the possible generic implications of the voltage-dependent growth rate phenomena as well as the plant-specific actions taken at Braidwood-1 and Byron-1 with respect to current operating cycle predictions. Our assessment at this time is that the modification to include a voltage-dependent growth rate is reasonable and appropriate.

Farley Unit 1 (Farley-1)

Both Farley units have been under predicting the maximum ODSCC voltage indications. However, the overall distribution of predicted voltages has been conservative enough to result in generally conservative predictions of the limiting leakage. For example, at the most recent outage at the EOC-14, Farley-1 compared actual ODSCC indication voltages with predictions made during the previous outage. In all three SGs, the total number of indications of any size, the number of indications greater than 1.0 volt, and the number of indications greater than 2.0 volt was usually significantly over predicted. In terms of maximum voltages, SG "A" was predicted to have a maximum voltage as high as 6.9 volt; the actual maximum voltage was 6.4 volt. SG "B" was predicted to have a maximum voltage as high as 6.7 volt; the actual maximum voltage was 3.1 volt. SG "C" was predicted to have a maximum voltage of 7.6 volt; the actual maximum voltage was 13.7 volt. The limiting MSLB leakage was predicted for SG "C" to be 10.2 gpm. Using the actual voltage distributions from the EOC-14 inspection, the limiting MSLB leakage was calculated for SG "C" at 7.6 gpm. Thus, the MSLB leakage was conservatively predicted and remained within the Farley-1 licensing basis despite the significant under prediction of the maximum voltage in SG "C." Based on EOC-15 predictions that incorporated the high voltage growth found in SG "C," Farley-1 concluded the plant cannot complete its current operating cycle without taking additional action to remain within its site allowable leakage limit. At this time, the licensee has not yet determined what specific action will be taken (e.g., mid-cycle inspection or reduced RCS dose equivalent iodine levels).

The staff is reviewing the most recent Farley 90-day reports. The methodology has continued to conservatively predict leakage at Farley. The lack of conservatively predicted maximum voltage indications in all cases was not unexpected at the time the staff approved the voltage-based repair criteria. However, it may indicate a need for Farley to reassess and possibly adjust the calculational approach for the predictions. Staff conclusions on this matter will be provided when we complete our review of the Farley 90-day reports.

The NRC staff has been informed that a potential discrepancy exists in the assessment of the radiological consequences of a MSLB due to primary-to-secondary leakage from SG tube indications (see meeting summary issued August 25, 1997). In correspondence dated July 30, 1997, Nuclear Energy Institute (NEI) stated all pressurized water reactor licensees would be informed of the potential error associated with performing leak rate and dose consequence calculations.

In summary, the staff has been monitoring and continues to monitor the performance of the voltage-based repair criteria. Based on reviews to date (up to March 1998), although the GL 95-05 methodologies may not predict the maximum voltage amplitude, the predicted EOC leakage rates under postulated accident conditions have been conservative when compared to the actual EOC conditions. (The Braidwood and Byron experience is related to plant-specific criteria and is being pursued on a plant-specific basis.) Furthermore, application of the voltage-based criteria (including Braidwood and Byron) has not resulted in any plant being outside its licensing basis. It also should be noted that several conservatism exist in the methodology used to calculate leakage and radiological doses under the voltage-based methodology. In particular, the leakage rates are calculated so as to be conservative 95 out of 100 times and the radiological doses associated with the projected leakage rate are conservatively calculated consistent with Standard Review Plan Section 15.1.5.

Table 1:

Plants Implementing Voltage-Based Steam Generator Tube Repair Criteria
per the Guidance of Generic Letter 95-05

PLANT	STATUS OF 90-DAY REPORTS AND ASSOCIATED REVIEWS
Beaver Valley 1	90-day report: EOC-11/BOC-12 (Reviewed by EMCB)

Braidwood 1	90-day report: EOC-5A/BOC-6 (Reviewed by PNNL) 90-day report: EOC-5B/BOC-6 (Reviewed by EMCB) 90-day report: EOC-6/BOC-7 (under review)
Byron 1	90-day report: EOC-6/BOC-7 (PNNL reviewed) 90-day report: EOC-7B/BOC-8 (under review)
Cook 1	90-day report: EOC-13/BOC-14 (PNNL reviewed) 90-day report: EOC-15/BOC-16 (EMCB preliminary review complete)
Farley 1	90-day report: EOC-13/BOC-14 (PNNL reviewed) 90-day report: EOC-14/BOC-15 (under review)
Farley 2	90-day report: EOC-10/BOC-11 (PNNL reviewed) 90-day report: EOC-11/BOC-12 (under review)
Kewaunee	90-day report: EOC-21/BOC-22 (EMCB preliminary review complete)
Sequoyah 1	90-day report: EOC-8/BOC-9 (under review)
Sequoyah 2	90-day report: EOC-7/BOC-8 (EMCB preliminary review complete)
South Texas 1	90-day report: EOC-6/BOC-7 (EMCB reviewed)

Regarding how NDE issues will be addressed in the future, the staff's regulatory initiative with respect to steam generator tube integrity is intended to ensure that a systematic approach for ensuring that tube integrity is maintained rather than continuing to rely on an ad-hoc approach. This regulatory initiative has as its key objective identifying those additional actions that are needed (in addition to the current minimum TS requirements) to maintain and monitor SG tube integrity consistent with governing regulations and plant licensing bases. The staff developed draft Regulatory Guide DG-1074 to provide guidance on one acceptable means for ensuring SG tube integrity consistent with governing regulations and plant licensing bases.
Inservice inspection NDE issues are among the key issues addressed in the DG-1074 guidance (refer to section C.1 of DG-1074). The guidelines identify measures that the staff finds acceptable for the nondestructive examination of SG tubing and specifically states that NDE systems should undergo a formal qualification process for each degradation mechanism in accordance with the Electric Power Research Institute (EPRI) Steam Generator Guidelines. This guidance states that only NDE systems satisfying flaw detection performance criteria specified in the EPRI guidelines may be used for inservice inspection. In addition, flaw size measurement performance should be established for each degradation mechanism through performance demonstration and must be accounted for in the applicable repair limit. NDE systems whose measurement performance has been characterized in this manner are termed "validated for sizing." Alternatively, if validated NDE techniques are not available for a given degradation mechanism, all indications associated with that mechanism should be plugged or repaired, irrespective of the measured flaw size.

The guidance developed by the staff to support the new regulatory initiative (i.e., DG-1074) also addresses leakage induced by postulated accidents such as a main steam line break (MSLB), the use of empirical correlations between leakage and an eddy current response parameter for calculating this potential leakage, and the treatment of eddy current limitations as they affect the leakage calculation. The guidelines provide that the tubes should be monitored with respect to performance criteria and that actions be taken as necessary to ensure that the performance criteria will continue to be met. The performance criteria are defined in DG-1074 so as to ensure compliance with plant licensing bases. Monitoring consists of inspection and/or test methods and analysis methods that provide high confidence in the assessment of the tubing relative to the performance criteria. DG-1074 states that it is acceptable to utilize flaw measurements from the NDE inspections as part of the assessment provided NDE measurement error/variability has been quantified by an acceptable performance demonstration. Where NDE measurement performance has not been quantified, this guidance provides that alternative methods (e.g. in situ pressure testing) should be employed in lieu of the NDE flaw measurements to monitor the condition of the tubing.

NDE flaw measurements, where appropriate in accordance with the guidelines accompanying the generic letter, provide a convenient means for assessing potential accident-induced leakage in cases where a relationship has been established between leakage and measured flaw size or other NDE parameter. These relationships (models) generally relate leakage to a single flaw parameter (e.g., flaw length) and to material properties. However, other flaw parameters (e.g., profile of crack depth over crack length, ligaments between crack segments, tortuosity of leakage path) also tend to affect the leakage value for a given flaw. Because these parameters may not be specifically accounted for in the models, the model estimates may incorporate significant uncertainty, depending on the complexity of the flaw morphology. Leakage data as a function of the flaw size parameter will therefore tend to exhibit scatter. To ensure conservative leakage estimates, the guidelines provide that the models should account for this uncertainty. The guidance provides that the tubing be assessed to provide high confidence that the tubing condition satisfies the performance criteria. Accordingly, this guidance states that all significant uncertainties affecting the outcome of the assessment should be accounted for, and provides guidelines for accomplishing this objective.

DG-1074 indicates that leakage models may be empirically-based or analytically-based. Analytical models typically do not explicitly quantify uncertainties in the model estimates, and, thus, the regulatory guide provides that the models should be developed to produce bounding estimates as confirmed by test. Empirical models often provide the most convenient and realistic treatment of the leakage potential of complex flaw morphologies. The supporting data set should cover the range of flaw morphologies and flaw sizes to which the correlation is to be applied. These guidelines state that empirical models should satisfy standard statistical "goodness-of-fit" and "significance-of-correlation" criteria and should quantify significant uncertainties including model parameter uncertainties and uncertainties indicated by data scatter. These uncertainties, the variability of material properties, and eddy current measurement error/variability should be considered in the leakage assessment such as to ensure an upper bound estimate of the total leak rate from all indications in the faulted steam generator with a probability of 0.95. The subject of crack size and severe accident evaluation is addressed in the response to issue 5 and in the appendix to this report.

Consistent with the staff position taken in GL 95-05 with respect to IGSCC at the tube to tube support plate intersections, the DG-1074 guidance provides that burst and leakage behavior may be correlated with the NDE voltage response to tubing flaws in lieu of direct NDE measurements of flaw depth or flaw length. Such correlations with the NDE voltage response are subject to the same criteria as those with respect to measured flaw depth and length; namely, such correlations must be supported by statistically significant number of data covering the spectrum of flaw morphologies being addressed by the correlation, satisfy standard statistical tests, and that model uncertainties be quantified and considered in the tube integrity assessments. Tube integrity assessments must account for the variability (repeatability) of voltage response to a given flaw and the variability among analysts in measuring the voltage response.

In conclusion, the staff understands the limitations of current NDE techniques in terms of both detection and sizing of IGSCC as well as the potential pitfalls of following empirical approaches for ensuring SG tube integrity. The staff has adequately accounted for these NDE limitations in the manner in which it has regulated the SG tube integrity issue to date. Future alternative strategies for managing SG degradation are adequately addressed in the DG-1074 guidance.

Response to Issue 2. "MSLB Leakage Issue":

NUREG-1477, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes" documented the staff's consideration of proposed voltage-based repair criteria (which subsequently developed into GL 95-05). In this report, the staff explicitly analyzed steam generator tube leakage during secondary side depressurization events, including main steam line breaks (MSLB). Thermal-hydraulic analyses were performed using the RELAP code to assess plant response to MSLB events involving a range of primary-to-secondary leakage. Calculations were performed over a range of leak rates from a few hundred gpm to over one thousand gpm. For these calculations, tube leakage was not assumed to begin until the differential pressure increased to above normal operating levels.

The results indicate that the primary-to-secondary pressure differential remains relatively constant immediately following the break, since primary as well as secondary pressures are decreasing. This is due to the cooling effects on the RCS caused by the high steam flow rates resulting from the secondary side break. Following depletion of the secondary inventory in the affected generator, the RCS pressure begins to increase as the cooling effect ceases and the emergency core cooling system (ECCS) injection increases the reactor coolant inventory. This causes the pressure differential across the tubes to exceed the normal operating differential pressure. The pressure differential would increase to approximately 2230 psid assuming no tube leakage and normal operator actions.

If cracks in the tubes begin to leak substantially, the increased leakage will counteract the pressure increase from ECCS injection. The increased pressure differential across the tubes is expected to open cracks somewhat, leading to higher leakage, but at some point, an equilibrium will be reached when tube leakage and ECCS injection match. The RCS pressure will then stop rising and the leak rate will stabilize. At this equilibrium point, tube differential pressure will essentially equal the RCS pressure, since the secondary is depressurized. The resulting tube differential pressure is expected to be greater than the differential pressure during normal operation (i.e., about 1400 psi) because some additional pressure differential is required to start the leakage.

Since the tube leakage is equal to the ECCS injection rate once the equilibrium is reached, the tube leakage is limited by ECCS pump capacity. A characteristic of pumps used in ECCS applications is that their capacity decreases with increasing discharge pressure. Since the RCS pressure is above the normal operating primary-to-secondary differential pressure, ECCS flow rate is less than that corresponding to an RCS pressure of 1400 psi. This flow rate is in the range of 500 to 1000 gpm (depending on plant design) which established the leakage range used in the thermal-hydraulic studies. The analyses concluded that provided several key operator actions are assumed, RWST capacity is expected to be sufficient to cool down the plant in the event of MSLB accompanied by tube leakage.

In conjunction with recent activities to revise the regulatory approach for maintaining SG tube integrity , the staff, through its contractor, arrived at the same conclusion regarding RWST capacity in INEL-95/0641, "Steam Generator Tube Rupture Induced from Operational Transients, Design Basis Accidents, and Severe Accidents." In these calculations, the primary-to-secondary leakage was assumed to commence coincident with start of the secondary depressurization. Provided operator action was taken to throttle ECCS injection, depressurize the RCS to allow placing the residual heat removal (RHR) system in service and achieving cooldown on RHR, the event could be mitigated before exhausting RWST inventory. In these analyses, RWST inventory was considered insufficient only if no operator actions were taken or if a large number of tubes (on the order of 10) simultaneously ruptured.

From the assessments documented in these reports, the staff concluded that if primary-to-secondary leakage occurred during a secondary depressurization (leak rates as great as that of a tube rupture), that sufficient margin in RWST capacity existed to mitigate the event before core damage occurred. The risk associated with the potential for cool down not being accomplished promptly in accordance with plant procedures is discussed in the response to issue 3.

Response to Issue 3. "Risk Increase Issue":

This issue was originally raised with respect to the core damage frequency (CDF) associated with steam generator tube leakage induced by MSLB events. More recently, it was expanded to include other issues identified by the staff involving the behavior of steam generator tubes under high temperature conditions that are associated with the core oxidation phase of severe accidents. These are two separate sets of sequences involving tube behavior under different conditions with different effects on the results of a probabilistic risk assessment (PRA). Therefore, they are addressed separately.

The risk associated with MSLB-induced tube rupture or leakage involves tube pressure differentials above normal conditions at nearly normal operating temperatures. The consequences affect both the CDF and containment bypass release frequency (CBRF) (releases of radioactive materials from the damaged core). In contrast, the risk of tube rupture or extensive leakage induced by core damage conditions involves significantly elevated temperatures, and impacts only the CBRF because it simply shifts some of the previously calculated CDF into the CBRF from other release categories.

Risk from MSLB Induced Tube Leakage:

In staff assessments of risk associated with tube leakage induced by steam generator secondary side depressurization, the impacts on CDF and CBRF were found to be much smaller than the value proposed in the DPO. NUREG-1477 evaluated the risk associated with tube leakage induced by depressurization of steam generator secondary sides by stuck-open steam line relief valves and feedwater line breaks in addition to main steam line breaks. It concluded that the frequency of the initiating depressurization events was dominated by stuck-open relief valves, with a frequency in the range of 10^{-3}/reactor year (RY), which was estimated from operational event data. Steam line breaks and feedwater line breaks also contributed to the initiating event frequency when estimated on a conservative basis. The probability of tube leakage on the order of 500 to 1000 gpm was assumed to be 1.0 for this analysis, although existing repair criteria and future proposed criteria are directed at maintaining induced leakage at significantly lower values. Further, none of the actual depressurization events have demonstrated significantly increased leakage. The resulting risk estimate was an increase in CDF and CBRF of about $2x10^{-6}$/RY.

Human error was the dominant failure to mitigate the event, with some contributions from failures of auxiliary feedwater and high pressure ECCS. Human error probabilities were estimated as 10^{-3}/cool down under the conditions resulting from these sequences. Other analyses, documented in draft INEL-0641, used a higher initiating event frequency of

$7.6x10^{-3}$/RY, which is dominated by stuck-open valves. Considerations of leak-before-break in the short sections of the steam and feed lines that are not isolable resulted in insignificant contributions to the initiating event frequency. The human error probability estimated in draft INEL 95/0641 is also higher, at 10^{-2}/cooldown. A very conservative approach is to substitute these values in the NUREG-1477 analyses. The results would be an increase in CDF and CBRF of about $2x10^{-5}$/RY if the probability of extensive leakage is still assumed to be 1.0. Note that even following this conservative analysis, the value of containment bypass with core damage given in the DPO is not approached.

At lower leak rates, there is more time for operator actions, so the corresponding human error rate is expected to be much lower than the INEL estimate. Also, since any acceptable regulatory approach (whether it is ultimately a generic letter or the NEI initiative) would limit the allowed leakage under depressurized secondary conditions to much lower values, the results discussed above are considered to be very conservative.

Risk from Station Blackout (SBO) Core Damage Sequence Induced Leakage and Ruptures:

When considering interim plugging criteria for SG tubes under NUREG-1477 and GL 95-05, the staff did not have the benefit of the PRA policy statement to focus its attention on severe accident risk. Nonetheless, the staff assessed the need to consider severe accident effects under IPC, and judged that the amount of allowed tube leakage under IPC was low enough that the existing analyses for tube response under high pressure severe accidents would remain valid. The staff assessment was based on the nature of the degradation involved, cracking (ODSCC) that is confined within the tube support plates. The staff reasoned that leakage from confined cracks would be limited and that confined cracks would have low probability of burst. Cracks were assumed to remain confined since the secondary depressurization involved does not exert the forces on steam generator secondary components which are postulated during a MSLB. Therefore, tube support plate displacement is not expected. Since the leakage calculation detailed in GL 95-05 is based on leakage rates measured from free-span cracks, the staff reasoned that actual leakage would be less. Further, if a confined crack did leak, the jet would be deflected by the tube support plate in a direction parallel to adjacent tubes, so that impingement effects could be neglected.

Since GL 95-05 was issued, related analysis of MSLB blowdown effects on TSP position relative to the tubes supports the initial staff conclusion that the severe accident risk is not significantly affected under voltage-based repair criteria. Since the dynamic forces during an SBO are less than in a MSLB, the assumption that the cracks will remain within the TSP appears reasonable.

NUREG-1570 addresses the risk of tube failures induced by the high temperature conditions associated with core damage induced by SBO and other similar events. These events are already included in the CDF. Therefore, the consideration of inducing extensive tube leakage or ruptures increases risk only by increasing the frequency of releases that bypass containment. The CDF remains unchanged. For this study, it was necessary to estimate the distribution of flaw sizes that occur in currently operating plants or that would exist in plants meeting the criteria in the proposed generic letter and draft regulatory guide. This is a very uncertain and sensitive parameter in the risk assessment process. Therefore, the staff considered a range of flaw distributions in its risk assessment. The results of a demonstration calculation for a Surry-like plant indicate that the CBRF from tube failures during the SBO core damage sequences is approximately $3.9x10^{-6}$/RY. About 39% (i.e., $1.5x10^{-6}$) of this comes from failure of unflawed tubes in sequences where reactor coolant pump seals have failed and the coolant loop water seal has cleared on a loop where a steam generator secondary has become depressurized. Of the 61% (i.e., $2.4x10^{-6}$/RY)that comes from failure of flawed tubes, about $1.5x10^{-6}$/RY is induced by the elevated differential pressure before temperatures increase, and about $9x10^{-7}$/RY is induced by the high temperatures that occur later in the sequence. Specific issues associated with the relationship of tube flaws and high temperature conditions, specifically the effects of tube leakage, are further addressed in the response to issue 5 below.

Response to Issue 4. "Iodine Spiking Issue":

DPO Author's Premise

In the July 13, 1994 memorandum to the Executive Director for Operations (EDO) regarding the use of voltage-based interim repair criteria for SG tubes (GL 95-05), the DPO author stated that the use of the iodine spiking factor of 500 cannot be supported for the assessment of MSLB accidents for those plants which implement the voltage-based repair criteria. The DPO author postulated that, should such plants experience a MSLB, the large differential pressure between the primary side coolant and the broken steam line would initiate an iodine spike (increase in release rate of iodine from the fuel to the reactor coolant) response which would be larger than the spiking value of 500 commonly assumed by the staff and licensees in their accident analyses.

The spiking factor of 500 originates from the SRP Sections for the MSLB and SGTR and is a common assumption in the analysis of the consequences of such accidents. In the performance of typical SRP evaluation, it is usually assumed that the reactor coolant is at the technical specification (TS) allowable 48 hour value for dose equivalent ^{131}I, typically 1 Ci/g. When a plant implements the voltage-based criteria, the DPO author postulated that the consequences of a MSLB accident would be exasperated by the event-induced leakage from the tubes to which GL 95-05 criteria was applied. The author postulated that for these circumstances the iodine spiking factor would be greater than 500.

The DPO author also indicated that steps taken by licensees to reduce the TS allowable 48 hour value for dose equivalent ^{131}I in primary coolant to offset the increase in doses resulting from the primary to secondary leakage emanating from the event-induced leakage is inappropriate and can result in Part 100 doses being exceeded. The basis for this premise was at lower primary coolant activity levels, e.g., 0.01 Ci/g, a MSLB accident could induce a spiking factor greater than 500, e.g. 750. Consequently, if this premise is correct, reducing the 48 hour TS value for dose equivalent ^{131}I could have a detrimental effect on the capability of the plant to meet Part 100 doses if the amount of activity available for release would actually increase.

Thus, the essence of the DPO author's position is that if the voltage-based criteria is applied to a plant's SG tubes and that plant experiences a MSLB, the result could be an iodine spike greater than the value of 500 commonly applied in present consequence assessments. Licensee's actions to lower the 48 hour value for dose equivalent ^{131}I in primary coolant does not ensure that doses would be maintained below Part 100 doses guidelines since spiking factors greater than the value of 500 could be experienced at reduced primary coolant activity levels. Thus, the quantity of dose equivalent ^{131}I actually available for release could increase from the activity which was available for release when the primary coolant activity of dose equivalent ^{131}I was at 1 Ci/g and the spiking factor was 500. Therefore, Part 100 doses could be exceeded.

Staff Reassessment of Spiking Factor

In the development of the radiological dose calculation portion of the staff effort to revise the regulatory approach on SG tube integrity, the staff re-assessed the manner in which dose calculations are performed for both SGTR and MSLB accidents involving degraded steam generator tubes. One of the principle aspects of the re-assessment was the iodine spiking factor. Differing views of iodine spiking were presented to the staff. On the one hand industry representatives claimed that the iodine spiking factor of 500 was unrealistic and too conservative. On the other hand, the DPO author claims that the iodine spiking factor of 500 is non-conservative when a plant, which has utilized the voltage based criteria, experiences a MSLB. The DPO author claims that this is particularly true when the primary coolant activity levels of dose equivalent ^{131}I are below 0.35 Ci/g. It was necessary that the staff address these two divergent views.

The staff participated in several meetings in which NRC licensees and representatives from EPRI and Nuclear Energy Institute (NEI) discussed iodine spiking. At these meetings, industry representatives indicated that they considered the iodine spiking factor of 500, which is referenced in SRP Sections 15.1.5 and 15.6.3, as too high and unrepresentative of the actual spiking data. To support that position, industry submitted for staff review two reports, one by A. K. Postma entitled, "Empirical Study of Iodine Spiking in PWR Power Plants, TR-103680, Rev. 1 and a second by Lewis and Iglesias entitled, "An Iodine Spiking Model for Pressurized Water Analysis".

Postma performed an empirical study of the iodine spiking phenomenon in PWRs. The intent of Postma's study was to quantify iodine spiking in postulated MSLB/SGTR sequences. The study included iodine spiking data from over 200 normal operational reactor transients and 2 SGTR events. From these data, Postma developed an empirical model to relate iodine activity levels in the primary coolant to the magnitude of power and pressure transients imposed upon the reactor. The staff had a contractor assess Postma's model. The contractor concluded that, since most of the data utilized by Postma originate from conditions approximating a SGTR rather than a MSLB, these data could be utilized to draw conclusions on iodine spiking for SGTR events. However, absent additional justification, these data could not be extrapolated to support an iodine transport analysis for a MSLB event. The staff's contractor concluded that an appropriate justification might be the "first principles" iodine spiking model of Lewis and Iglesias that relates the iodine release from the fuel rods to the primary coolant thermal/hydraulic conditions. However, in order to provide such justification, validation and verification of the Lewis and Iglesias computer code, which implements the model, was necessary. The staff had the contractor proceed with such a validation and verification.

When the contractor performed the validation and verification of the Lewis and Iglesias code, they concluded that the code was insufficiently mature for use. It appeared that the code had not undergone any quality assurance. In addition, the code appeared to have large uncertainties when it was used to predict iodine spiking behavior **a priori.** This was particularly true when default input parameters were used to predict spiking response. Consequently, the staff concluded that the Lewis and Iglesias code could not be utilized to predict the spiking behavior in the event of a MSLB. Since these conclusions were drawn, industry had not provided any additional insights or information to address the staff's concerns on this model.

In the DPO author's July 13, 1994 memorandum to the EDO, it was indicated that iodine spikes as high as 10,000 had been observed. A figure was attached to that memorandum to demonstrate that point. The figure contained the iodine spiking factor as a function of initial iodine activity level. The memorandum indicated that the figure was obtained from LER-009/03L-0.

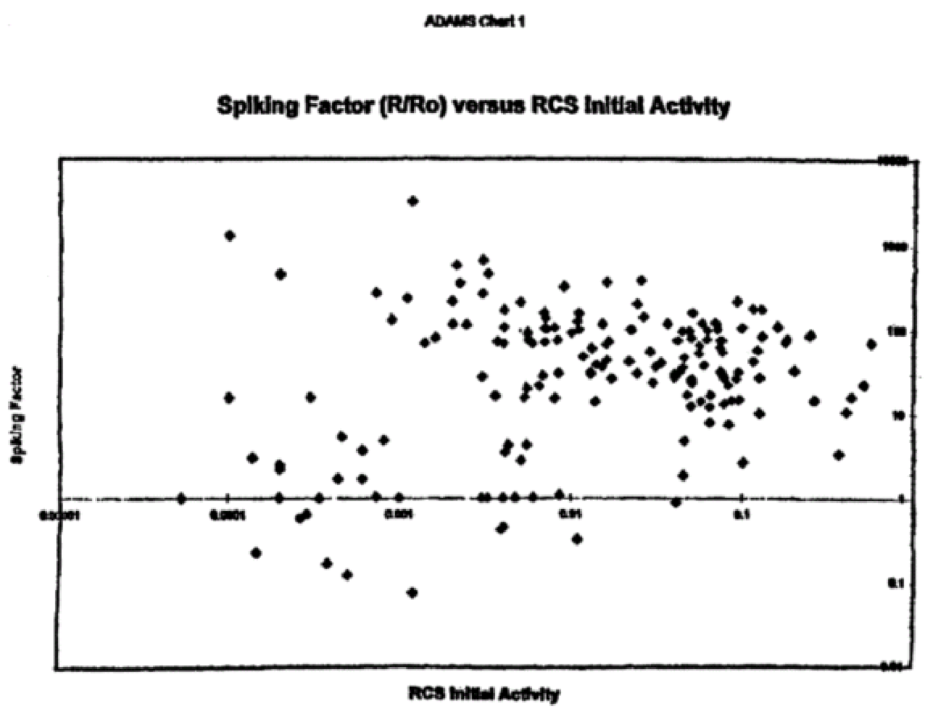
The staff has reviewed existing iodine spiking data. An article by Adams and Atwood published in Nuclear Technology in June 1991 and an earlier article by Adams and Sattison published in a May 1990 issue of Nuclear Technology were reviewed. The Adams and Sattison article was entitled, "Consequences of a SGTR Event" and it presented the iodine release rate ratio (spiking factor) for 58 events. The spiking factors ranged from a low of 1.7 to a high of 908. The initial iodine activity level at the time of the event ranged from 0.004 Ci/g to 0.943 Ci/g. The data included three cases where the spiking factor exceeded the SRP value of 500. For each of these cases, the large spiking factors were associated with low initial activity levels of iodine, 0.013, 0.014 and 0.02 Ci/g, respectively. For these three cases, the maximum iodine activity level never exceeded 3.5 Ci/g. Consequently, Adams concluded that events in which large spiking factors occurred tended to be associated with moderately small iodine concentrations and, therefore, small iodine release rates. Adams and Sattison concluded that use of the iodine spiking factor of 500 resulted in an overly conservative analysis for a SGTR with a coincident iodine spike. They recommended that the data base be expanded to include more operational data to reduce the uncertainties associated with their assessment. It should be noted that the data which Adams and Sattison reviewed were only representative of SGTR type of events and not MSLB events.

In an article entitled, "The Iodine Spike Release Rate During a Steam Generator Tube Rupture," Adams and Atwood performed an analysis to bound the actual maximum iodine activity level for each iodine spiking event. They considered such an analysis necessary because primary coolant samples are not taken continuously in the event of a transient. Samples are typically taken once every four hours. The bounding analysis performed by Adams and Atwood was obtained from data in LERs. Adams and Atwood postulated that the maximum activity level of dose equivalent ^{131}I resulting from a reactor trip was no more than a factor of three greater than any value measured 2 to 6 hours after the trip. Therefore, they bounded the maximum values by taking the actual measured values of primary coolant activity and multiplying them by a factor of 3. Adams and Atwood presented data in a figure which includes not only the bounding value of 3 noted above but also an adjustment to the data based upon the assumption that the maximum activity level will occur two hours after the trip. It is only when the actual measurements have been increased by this factor of three and adjusted to assume that the maximum occurs at two hours after the transient that there arises an instance where one spiking factor is approximately 10,000.

Adams and Atwood concluded that the spiking factor could be reduced. In their article they stated, ".... the iodine release rate assumed in the calculation of a SGTR event could be reduced substantially, (e.g., by a factor of 15) and still result in a conservative analysis."

It seemed appropriate for the staff to utilize the existing spiking data to reassess the suitability of the spiking factor. Figure 1, which includes Adams data without the bounding factor of three, presents data on spiking factor as a function of initial primary coolant activity level.

FIGURE 1



The staff had initially anticipated that these data might demonstrate that the spiking factor was a function of initial primary coolant activity level of iodine and that an expression could be developed demonstrating such a correlation. However, no such expression could be developed from the data. The best that could be done was to envelope the data and to state that the spiking factor could be no higher than a given value, independent of the activity of iodine in the primary coolant. Assessing the spiking data independent of the primary coolant activity level, the staff determined the 95% value. This value was 335 for a SGTR event.

Drawing conclusions for the spiking factor for a MSLB is a much more difficult issue to address. The existing spiking data is representative of a SGTR type of event and not a MSLB event. It is not possible to extrapolate the iodine spiking data base for a SGTR event to that for a MSLB in a manner that is rigorously defendable. As an example, if it is the pressure drop rate that dominates the primary coolant iodine behavior, and there is a linear relationship between this rate and the resultant iodine activity level or release rate, then one could judge that a MSLB would result in an iodine activity level increase of two or three over that of a SGTR event. If the relationship between the pressure drop rate and the iodine release is quadrature, the overall increase would be four to nine. In discussions with Adams, he indicated that the actual mathematical relation between the pressure drop/drop rate and iodine activity level/release rate is not known, however it is reasonable to expect that the relation is probably not of a higher order than a quadrature. In fact, it could be even less, e.g., a square root relation. Consequently, one could estimate the primary coolant iodine activity level/release rate magnitude for a MSLB to be within a factor of 10 times that for a SGTR based upon pressure drops and pressure drop rates being within a factor of three.

As previously noted, Adams indicated that the SGTR spiking factor may be conservative by an order of magnitude. Given that order of magnitude conservatism and the estimation of the increase in an order of magnitude of the MSLB over the SGTR, the staff determined that the two factors tend to offset each other. Absent data from operating plants which demonstrate the effect of a MSLB on iodine spiking, the only means for determining the potential consequences on the fuel of such a break would involve the performance of laboratory tests. Because the MSLB event is an event of much lower probability than a SGTR event, the staff has concluded that there appears to be insufficient justification for performing such tests. Therefore, based upon the above discussion, the staff concluded that a change to the current SRP value of 500 used in the MSLB analysis is inappropriate at this time.

Assessment of DPO Concern

To assess the DPO author's concern, the staff performed a parametric analysis to determine the impact of an increase in spiking factor on the acceptability of the consequences of a MSLB event when reductions in reactor coolant activity levels of dose equivalent ^{131}I are allowed. The staff performed this parametric assessment utilizing data from a plant which recently received a GL 95-05 repair criteria amendment. This plant was a 3-loop Westinghouse reactor.

In the parametric analysis the staff performed an assessment for a base case. This case was based upon the SRP conditions. For this assessment the staff assumed that reactor coolant activity level of dose equivalent ^{131}I was at its 48 hour TS value of 1 Ci/g. Total primary to secondary leakage was at the TS value of 1 gpm with an assumed 150 gpd/SG primary to secondary leak for the intact SGs and the remainder of the 1 gpm primary to secondary leak assumed to be to the faulted SG. Consistent with the SRP for the MSLB accident, a spiking factor of 500 was assumed. Releases were calculated for two time periods, 0-2 hours and 0-8 hours, which are representative of the exclusion area boundary (EAB) and low population zone (LPZ) doses.

In the amendment which was processed for this particular GL 95-05 repair criteria application, the licensee assumed that it took 8 hours following the MSLB before the faulted SG was isolated. The staff assumed the same for the base case. It should be noted that eight hours is the maximum time that would be anticipated to elapse before the faulted SG would be isolated and for the reactor coolant system to be depressurized thereby halting any primary to secondary leakage. If the period of time in which the faulted SG was isolated was reduced to less than 8 hours, a reduction in the release quantities to the environment would occur since the major pathway to the environment is through the faulted SG.

In the parametric assessment the staff assumed that any primary to secondary leakage, even to the intact SGs, was released immediately to the environment. The staff took no credit for partitioning in the intact SGs. While this was a conservative assumption, the staff concluded that, if releases occur for 8 hours from the faulted SG, the overall impact of this assumption results in a small contribution to the overall release because of the dominance of the release from the faulted SG.

Because of the variability of meteorology associated with the reactor sites it was decided that the parametric assessment would not calculate doses. Rather it was assumed that the 0-2 and the 0-8 hour releases of ^{131}I would be calculated for the base case. The parametric analysis would then determine the value for the spiking factor in order to limit the total release of ^{131}I for any case to the quantity which had been calculated for the base case. For example, if the base case calculated a release of 111 Ci of ^{131}I for the 0-8 hour period, then the parametric analysis calculated how large the spiking factor could be at a given primary to secondary leak rate and at a given reactor coolant activity level before it resulted in a release of 111 Ci.

Having established the ^{131}I releases for the SRP case as the basis for the acceptability of a reduced reactor coolant activity level and the primary to secondary leak rate, the parametric analysis was performed for reactor coolant activity levels of 0.5, 0.1, 0.05, 0.01, and 0.005 Ci/g of dose equivalent ^{131}I. The total primary to secondary leakage rate was allowed to vary. Calculations were performed at 10, 35 and 100 gpm for each of these reactor coolant activity levels. For each case, a determination was made as to how large the spiking factor could be before the releases would be equal to those calculated for the base case. The result of this parametric analysis are presented in Table 2.

To place into proper perspective the information contained in Table 2, the following example is provided. Table 2 demonstrates that, at certain reactor coolant activity levels, the spiking factor must be less than a value of 500 before the release will be equivalent to that of the base case. For example, it can be seen in Table 2 that where the primary coolant activity level of dose equivalent ^{131}I is 0.5 Ci/g and the primary to secondary leak rate is 35 gpm, if the spiking factor is above 12.5, the releases of ^{131}I will be greater than the base case. The standard assumption is that the spiking factor is a value of 500. Consequently, for this particular example, if it is assumed that the base case resulted in a dose which was at the SRP acceptance criteria, in this case 30 rem thyroid, then the dose criteria for this particular case has been exceeded because the spiking factor would have needed to be 12.5 or less. Otherwise, the releases would be greater than the base case. This example from Table 2 clearly demonstrates that, if the desire is to maintain the primary to secondary leak rate at a given value, the primary coolant activity level must be reduced until the spiking factor of 500 or greater is reached. For the above noted example, Table 2 indicates that a primary coolant activity level between 0.01 Ci/g and 0.05 Ci/g would be required.

Summarizing, from Table 2, it can be observed that for some combinations of the primary to secondary leak rates and reactor coolant activity levels, a spiking factor of less than 500 would be required to maintain releases equivalent to those for the base case. This infers that the dose criteria of 30 rem thyroid would be exceeded with the assumed spiking factor of 500. To compensate for the unacceptable releases (doses) necessitates a reduction in either the TS allowable values for primary to secondary leakage or reactor coolant activity level of dose equivalent ^{131}I. Because the purpose of GL 95-05 amendments is to allow tubes to remain in service and thus, maintain a given primary to secondary leak rate, the item most likely to be reduced would be the reactor coolant activity level of dose equivalent ^{131}I. The reactor coolant activity would continue to be reduced until the spiking factor was at least 500 thereby assuring that the 30 rem criteria was met for the assumed leak rate. It should be noted that the spiking factor would have to reach at least a value of 5000 (300 rem/30 rem x 500) before a dose would ever reach a Part 100 guideline. In Table 2, at and below 0.01 Ci/g, the spiking factors would need to be in the range of 5200-51,800 before the dose guidelines of Part 100 would be exceeded. This is not a likely case. For activity levels above 0.01 Ci/g but less than 1.0 Ci/g the consequences of the analysis push the primary coolant activity level continually down until a spiking factor of 500 is achieved thus, assuring that Part 100 doses could only be exceeded for spiking factors greater than 5000. There exists no data to suggest that the spiking factor is likely to be >5000 nor is there data for the SGTR spiking conditions to suggest that even with a factor of 10 increase in the spiking factor, the spiking factor will be greater than 5000. Therefore, the staff concluded that adequate margins exist with present spiking factor of 500 to maintain doses below Part 100 guidelines in the event of a MSLB even at lower reactor coolant activity levels.

Table 2

Spiking Factors Which Would Result in Releases Equal to Those of the MSLB SRP Case#

		Spiking Factors##	
RCS Activity Level (Ci/g Dose Equivalent ^{131}I)	**Primary to Secondary Leak Rate (gpm total)**	**0-2 Releases**	**0-8 hour Releases**
0.5	100 35 10	* 12.5 86.3	8.18 26.6 98.6
0.1	100 35 10	37 133 502	57 150 511
0.05	100 35 10	91 283 1020	118 305 1030
0.01	100 35 10	526 1490 5180	606 1540 5150
0.005	100 35 10	1070 2999 10400	1220 3090 10300

SRP Case based upon reactor coolant activity level of 1 Ci/g of dose equivalent ^{131}I, a spiking factor of 500 and 1 gpm total primary to secondary leak rate.

Spiking factors would need to be multiplied by a factor of 10 in order to equate to a dose of 300 rem thyroid.

* Maximum allowable leak rate would be limited to approximately 60 gpm.

Response to Issue 5. "Severe Accident Issues":
The assessment of the CBRF due to thermally induced failures of flawed tubes, referenced in the response to issue 3 above, is based on a probabilistic treatment of the creep behavior of the surge line, hot legs, and steam generator tubes with various flaw sizes. These analyses yielded the probabilities of whether axially cracked tubes would fail before other portions of the RCPB under the temperature and pressure conditions predicted in different SBO core damage sequences. Those results were then combined with estimated flaw size frequency distributions to calculate the probability that one or more tubes would fail before another part of the RCPB for each accident sequence. In these analyses, the staff recognized that through-wall flaws which are too short to burst at normal temperatures may burst later when temperatures increase. The staff also assumed that the effects of impingement of a hot steam jet from one burst tube may lead to rapid failure of an adjacent tube due to erosion. So, if a tube was calculated to be the first RCPB failure, then no credit was taken for later depressurization of the RCS by a subsequent failure of some other component.

Impact of Small Cracks

At tube temperatures before surge line failure, critical flaw lengths for axial cracks (i.e., those flaws that would burst) were usually in the range of 0.4 to 0.6 inch. In the staff analysis, flaws as short as 0.25 inch that propagated through-wall during the challenge were assumed to leak so severely at high temperatures that they would be equivalent to ruptures. Staff estimates of the opening of a 0.25 inch long crack at elevated temperatures resulting from erosion and creep or deformation could lead to adjacent tube effects in a short time (minutes to an hour). However, the degree of erosion of small cracks due to the high velocity passage of superheated steam has not been directly assessed. Further, the nature of other contributing factors, such as duration of peak temperature conditions, and possible effects of differential thermal expansion, make an explicit determination of crack opening rate difficult.

The staff acknowledges that under certain conditions of tube degradation and sustained high RCS temperature and pressure conditions, crack opening could be a concern. As indicated above, the model used in the staff's risk assessment assumed that through-wall cracks of .25 inches in length would lead to tube failure. The .25 inch length was based on consideration of corrosion cracking mechanisms and typical aspect ratios observed for cracking mechanisms. In order to assess the potential for the existence of shorter through-wall cracks, the staff reviewed available data from destructive examination of tubes removed from service due to a variety of typical degradation mechanisms. The evaluation of depth/length ratios of cracks, provided in the appendix, concludes that one type of tube degradation (e.g., primary water stress corrosion cracking in the hard roll transition at the top of the tube sheet) presented a concern that very small through-wall cracks could be present at a limited number of plants. The staff does not have data to conclusively demonstrate the behavior of very small through-wall cracks under core damage conditions. However, based on evaluation of erosion data at similar conditions for similar types of materials, the staff has assumed that such defects could induce tube failure. Therefore, consideration of alternate repair criteria will need to include assessment of the severe accident risk contribution from all degradation in a plant, including possible crack opening and leakage effects.

Leakage Effects on SG Inlet Plenum Mixing

At the March 5, 1997 meeting of the ACRS Materials and Metallurgy/Severe Accidents Subcommittee, the DPO author made a presentation that disputed conclusions reached in NUREG-1570, "Risk Assessment of Severe Accident Induced Steam Generator Tube Rupture." A basis of the discussion was that the impact of tube leakage on steam generator inlet plenum mixing during the course of a high pressure core damage event was not considered.

The criteria used in the staff analysis reasonably accounts for the risk that may be contributed by tubes leaking during severe accidents. In NUREG-1570, tube rupture was considered the failure criterion contributing to risk, since a large release of fission products could then be anticipated. Tube leakage was not considered to have a significant impact since leakage of sufficient magnitude to disrupt natural circulation flows would not be expected unless a tube rupture had occurred. The basis for this can be seen by examining the thermal-hydraulic analyses documented in NUREG/CR 5214, "Analyses of Natural Circulation During a Surry Station Blackout Using SCDAP/RELAP5." The results indicate that hot leg flows of approximately 3 kg/sec can be expected. Using a tube bundle to hot leg flow ratio of 2 (approximation from test results), a tube bundle flow (m_t) of about 6 kg/sec can be anticipated. To disrupt the tube bundle natural circulation flow pattern, a leaking tube would need to divert a sizeable fraction of this flow.

To estimate the magnitude of leak flow that could be anticipated during the accident, an anticipated crack size was first determined. Under steam generator degradation specific management (SGDSM), accident-induced leak rates of about 100 gpm are conceivable during design basis accidents (DBAs) that result in elevated tube differential pressure. Equations from Appendix I of NUREG/CR-4483, "Reactor Pressure Vessel Failure Probability Following Through-wall Cracks due to Pressurized Thermal Shock Events" (see equations for sub-cooled conditions, below) were used to find the crack size that would result in a leak rate of 100 gpm under elevated differential pressure conditions associated with a DBA. This leakage model is the same as that used in NUREG-1477, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes." Using the model, a crack size of approximately 0.12 in^2 was found to correspond to 100 gpm leakage.

During the core degradation phase of the SBO progression, superheated steam will be entering the steam generator from the hot leg. In the analyses for NUREG-1570, tube differential pressures resulting from secondary system depressurization were considered. Thus, the flow through a crack in a tube can be computed assuming choked orifice flow (the choked flow assumption holds since secondary side pressure is below the critical pressure of approximately 1300 psi).

Using the equation for superheated conditions (see below), leak rates for superheated steam under the conditions calculated by SCDAP/RELAP5 can be estimated for various crack sizes. The table below gives estimated leak rates for different crack sizes corresponding to: (1) a 0.25 in long, 0.01 in wide opening, (2) a crack corresponding to a 100 gpm leak under design basis accident (DBA) conditions, and (3) a crack equal in size to the open end of a tube (7/8 in. diam.).

Leak Rate	**Crack Size (in^2)**		
	0.0025	**0.12**	**0.47**
kg/sec	0.027	1.3	5.13
lbm/sec	0.06	2.8	11.31
% of m_t	0.5	22	86

Equations for sub-cooled conditions:

$$m = cA\sqrt{\left(2g_c\, p(P_o - P_{sat})\right)}$$

$$P_o = P_{rcs} - P_f$$

$$P_f = 0.5 f \frac{L}{D_h} \frac{(m/A)^2}{g_c p}$$

$$f = (2 \ln(\frac{D_h}{2\delta}) + 1.74)^{-2}$$

Equation for super-heated conditions:

$$m = cAP_{rcs} \sqrt{\left[\frac{g_c}{RT} k \left(\frac{2}{k+1} \right)^{(k+1)(k-1)} \right]}$$

where

A crack opening area
c coefficient of discharge (0.6)
D_h hydraulic diameter
f friction factor
g_c conversion (32.2 lbm-ft/lbf-sec^2)
k specific heat ratio (c_p/c_v) = 1.26
L crack length
m mass flow rate
P_{sat} saturation pressure at RCS conditions
P_{rcs} RCS pressure
P_f frictional pressure drop
R gas constant (85.8 ft-lbf/lbm-R)
T steam temperature in Rankine
roughness height in crack (2x10^{-4} in. from NUREG 4483)
liquid density @ P_{sat}

The first crack size shown in the table, 0.0025 in^2, would yield about 2 gpm leakage under DBA conditions. In the analysis, this is the smallest size crack that the staff assumed to fail by rupture under the predicted conditions during a severe accident. The steam leakage through this crack is a very small fraction of the total tube bundle flow, 0.5% of m_t, and is not expected to have an impact on inlet plenum mixing. As mentioned previously, small through wall cracks are expected to be present in only a few cases involving one type of degradation.

The 0.12 in^2 crack (1 inch long, 0.12 inch wide), which corresponds to 100 gpm leakage under DBA conditions, would leak under severe accident conditions at a significant rate compared to the total tube bundle flow. This size flaw is considered equivalent to a tube burst if it is located in a single tube. It is 4 times as long and more than 10 times the area of the crack the staff assumed would burst under severe accident conditions. The largest tube opening shown in the table would divert a very significant portion of the tube bundle flow. These two flaws, which would both be assumed to rupture in the staff's analysis, could be expected to change the mixing conditions. Only the largest leak rate of 11.31 lbm/sec associated with this size opening is in the range of leakage cited in the DPO presentation (10 - 250 lbm/sec). Under the criteria used by the staff in the NUREG-1570 analysis, a tube would be considered failed upon rupture, which could result in crack openings on the order of a tube diameter. Once failed, further analysis of the thermal-hydraulic effects of this magnitude of leakage is not required to assess the risk impact.

Operating experience has shown that the occurrence of tube leakage is often associated with a series of flaws; that there is not only a single flawed tube (e.g., ANO Unit 2, 1992; McGuire, Unit 1, 1992; Maine Yankee, 1990 and 1994). The accident leakage that would be permitted under SGDSM is an aggregate value from all flaws that would leak. Thus, it would be reasonable to expect that in a severe accident situation, the leakage from flaws would be distributed throughout various locations in the tube bundle, and not necessarily confined to one tube or location. Therefore, the leakage effects on inlet plenum flow patterns, and mixing which is expected to occur, should be impacted to an even smaller degree than is indicated by the comparison of leak flow to bundle flow shown in the table above.

Uncertain factors in this discussion are the potential for cracks to open while they leak under severe accident conditions, and for leaking tubes to cause cascading failures. Preliminary staff estimates conclude that through-wall cracks could open at a significant rate if the high temperature and differential pressure conditions are sustained. However, the duration of very high tube temperatures is on the order of minutes. Also, the rate of cascading failure predicted in the DPO seems overestimated based on preliminary staff estimates which show only a modest erosion rate for tubes adjacent to a leak.

For analysis of severe accident risk, the staff chose 0.25 inch as the crack length at which to assume tube failure during severe accident thermal challenge. Based on a subsequent staff assessment of data from destructive examination of degraded tubes removed from operating plants as well as its assessment of leakage effects from cracks, this criterion is reasonable. The staff believes that there is a potential for through-wall cracks shorter than the staff's threshold of 0.25 inch to exist in a small number of plants. However, the risk significance of such defects requires further understanding of the frequency and nature of the plant specific sequences that could challenge tube integrity. The staff plans to investigate sequences in conjunction with the

IPE follow up program.

Fission Product Deposition and Heating

By letter dated May 20, 1998, the DPO author forwarded to the ACRS slides from a presentation given by a Japan Atomic Energy Research Institute (JAERI) representative at the May 1998 Cooperative Severe Accident Research Program (CSARP) meeting. The JAERI analysis addressed the additional heating of steam generator tubes, during a station blackout sequence with secondary side depressurization, caused by the deposition of fission products on the inner tube surface.

The JAERI analysis, while it predicted the steam generator tubes would survive the transient, with a small margin, also predicted substantial fission product heating of the steam generator tubes due to the deposition of fission products. This is in contradiction to the NRC conclusion, based on analysis, that the effect of deposited fission products in tube heating is trivial. The staff, therefore, evaluated the JAERI analysis to understand why it produced substantially different results concerning fission product deposition, from corresponding NRC calculations.

A review of the JAERI analysis revealed several differences from the NRC evaluation. First, it was learned that the JAERI analysis assumed the temperature of the steam entering the tube bundle was equal to the temperature of the unmixed vapor in the steam generator inlet plenum (from the hot leg). Assuming the temperature of the steam entering the tube bundle is equal to that of unmixed vapor from the hot leg is a severe conservatism; in contrast, the NRC analysis includes mixing of steam in the inlet plenum based on experimental data. (Use of unmixed vapor temperatures would produce excessively high tube temperatures irrespective of fission product deposition.) Use of the temperature of the unmixed vapor in the steam generator inlet plenum as the temperature of the vapor in the tube in the JAERI analysis apparently resulted in a temperature difference between the vapor and the steam generator tube wall of up to 250K inside the first section of the tube. It is this high temperature difference that is responsible for a large thermophoretic fission product deposition in the JAERI calculation. It is unrealistic to assume such a large temperature difference in this region without water on the secondary side of the steam generator given the heat transfer across the thin tube wall. The NRC's analysis showed a temperature difference of about 15K in this region, which results in minimal deposition by thermophoresis.

The NRC's VICTORIA code analysis of this sequence showed that the dominant mechanism for deposition was settling inside of the steam generator u-tubes onto the upward facing surfaces in the bends of the u-tubes. This is reasonable given the recirculatory flow patterns in the RCS and the lack of other driving forces for deposition other than settling. The JAERI analysis did not appear to model settling on upward facing surfaces inside of the steam generator u-tubes.

Finally, in another area, the JAERI analysis did not explicitly consider the primary source of heat (i.e., cladding oxidation) in their calculation of heat up of the steam generator tube wall. The relative importance of deposited fission product heating was determined by JAERI by comparison against the heat carried to the tubes by superheated steam heated only by decay heat in the core. Because of their thermal hydraulic assumptions and analytical treatment, the staff concludes that the finding of the JAERI analysis, relative to the significance of fission product heating, is not relevant to our evaluation of this matter. As a result, the staff's previous conclusions regarding the severe accident risk implications of degraded steam generator tubes remain unchanged.

Conclusion

The staff recognizes that alternate repair criteria could be proposed that might lead to the presence of short through-wall cracks. As mentioned previously in this document, the staff is encouraging licensees who propose alternate repair criteria to follow risk-informed approaches which may mean that licensees will need to address the issues discussed in this section as appropriate dependent on the nature of the proposal. Currently, with the exception of accident leakage rates associated with GL 95-05 repair criteria (which represent a different risk concern due to the confined nature of the cracks), accident leak rates for all other forms of degradation are limited to current licensing basis values (typically 1 gpm or less) which is well below the leakage rates contemplated above (i.e., on the order of 100 gpm). The staff will not approve any proposed repair criteria as an alternative to compliance with existing deterministic requirements if it unacceptably increases risk.

IV. REFERENCE DOCUMENTATION REVIEWED TO IDENTIFY DPO ISSUES

1. Memorandum from Gillespie to Heltemes "GSI-163, Multiple Steam Generator Tube Leakage," September 28, 1992.
2. Memorandum from Hopenfeld to Reed "Concerns Regarding Generic Letter On Voltage-Based Repair Criteria," February 18, 1994.
3. Memorandum from Hopenfeld to Taylor "Differing Professional Opinion Regarding Voltage-Based Interim Repair Criteria for Steam Generator Tubes," July 13, 1994.
4. Memorandum from Hopenfeld to Shao" NRC Generic Letter 95-03: Circumferential Cracking of Steam Generator Tubes," May 16, 1995.
5. Memorandum from Hopenfeld to Serpan "Comments on Draft EPRI Report TR-106194, Risk from Severe Accidents Involving Steam Generator Tubes Leaks or Ruptures January 29, 1996," March 7, 1996.
6. Memorandum from Hopenfeld to Chairman Jackson "Steam Generator Degradation and Plugging Criteria," March 28, 1996.
7. Memorandum from Milhoan to Hopenfeld "Resolution of Differing Professional Opinion Regarding Voltage-Based Repair Criteria for Steam Generator Tubes, Dated July 13, 1994," May 1, 1996.

8. Memorandum from Kress (ACRS) to Taylor "Proposed Rule on Steam Generator Integrity" November 20, 1996.
9. Memorandum from Hopenfeld to Taylor "Recommendation for an External Review of 'Differing Professional Opinion regarding Voltage-Based Repair Criteria for Steam Generator' July 15, 1994," December 3, 1996.
10. Memorandum from Mitchell to Hopenfeld "DPO regarding Voltage-Based Repair Criteria for Steam Generator Tubes-Issues," March 12, 1997.
11. Memorandum from Hopenfeld to Callan "Differing Professional Opinion Regarding Voltage-Based Repair Criteria for Steam Generator Tubes: Steam Generator Rulemaking," June 16, 1997.
12. Memorandum from Callan to Commission "J. Hopenfeld's Differing Professional Opinion Concerning Voltage-Based Repair Criteria for Steam Generator Tubes: Steam Generator Rulemaking," June 27, 1997.

Appendix: Steam Generator Tube Flaw Geometry

ATTACHMENT 2-APPENDIX

STEAM GENERATOR TUBE FLAW GEOMETRY[2]

The assessment of steam generator tube degradation identified during eddy current examinations has historically focussed on tubes with indications that could potentially challenge the structural and leakage integrity of the tube under postulated secondary side depressurization events. However, only a limited amount of information is available on the number of tubes that contain degradation smaller in size relative to the larger, more significant flaws that are readily detected during eddy current examinations. Under severe accident conditions, elevated pressures and temperatures may induce steam generator tube failure by a time-dependent failure mechanism of short, through-wall flaws that exhibit no or marginal signal response during nondestructive examinations (NDE). Because of the limitations in inspection technology, it has been postulated that the existence of these flaws could lead to steam generator tube failure under severe accident conditions. In order to have the potential for failure, these undetected flaws must have a through-wall or near through-wall morphology to introduce a primary-to-secondary leak path that will eventually lead to ablation of the flaw surfaces and expand the dimensions of the defect. This could then lead to increased primary-to-secondary leakage of hot gases through the defective tube and subsequent failure of surrounding tubes by jet impingement.

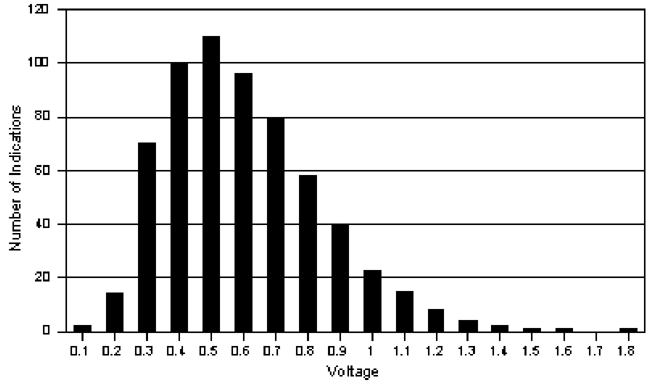
The risk from the failure of longer, through-wall steam generator tube defects (length >0.25 inches) under severe accident conditions has been examined in previous work and is conservatively included in the risk assessment model. In order to address the issue of short crack failures under severe accident conditions, a scoping study was completed that included a review of metallurgical data obtained from steam generator tube destructive examinations and correlations developed relating flaw depth and length. The focus of the study was to attempt to determine if there is a significant population of tube defects in steam generator tubes characterized by short lengths (<0.25 inches) with through-wall or near through-wall depths that could develop primary-to-secondary leaks in the freespan tube areas under severe accident conditions.

Background

Utilities occasionally submit to the NRC distributions of steam generator tube flaws as a function of some NDE parameter. These can be used to explain the overall significance of the degradation identified during an inspection in terms of the total number of degraded tubes identified and the nature of the indications in terms of NDE measurements (e.g., voltage, estimated length, depth). Figure 1 represents the typical form for an end-of-cycle voltage distribution of outside diameter stress corrosion cracking (ODSCC) indications at tube support plate (TSP) intersections. As seen in the figure, the number

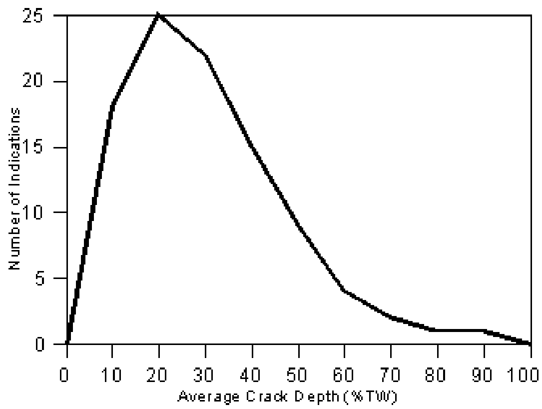
of indications is low within each voltage bin for low voltages but increases rapidly for slightly larger voltages. At higher voltages the number of indications gradually tails off toward zero.

Figure 1: Voltage Distribution



These characteristics are consistent with voltage distributions reported to the NRC for indications detected at drilled tube support plate intersections. In addition, this observation is consistent with other reported distributions such as that depicted in Figure 2. This figure shows the number of flaws as a function of the estimated average crack depth. Again, the number of indications increases, peaks at some value, and then decreases toward the tail of the distribution.

Figure 2: Crack Distribution



Although these distributions could lead one to conclude that flaws with low voltages or limited through-wall depth occur less frequently than those with slightly higher voltages or depths, this would not be a valid conclusion for several reasons. First, the above figures are representative data derived from eddy current inspections. Although sensitive to small flaws, eddy current technology has limitations. Steam generator tube defects that exhibit lower voltages will, by definition, have only a small influence on the eddy current signal. Since degradation generally occurs in crevice locations or in areas where the tube geometry is changing, other signals may overwhelm the defect response and mask its detection. Data analysts will also influence the threshold of detection for tube degradation. Although the inspection system may display a distortion due to the presence of a flaw, an analyst may not be as sensitive to these small signal changes compared with the signals from more obvious indications. A true flaw size distribution cannot be obtained from eddy current inspection data for smaller flaws (length, depth, voltage) because of the limitations of the analysts and the sensitivity of the technology used for steam generator tube examinations.

In order to better understand the distribution of cracks below the threshold of detection for eddy current inspection techniques, examining data in figures similar to that in Figures 1 and 2 is of little value due to the limitations of NDE. An estimate of the number of flaws below the NDE threshold could be obtained by extrapolating the distribution to lower voltages and depths. However, extrapolating the data may introduce additional errors and incorrectly model the actual population of flaws over the range of the extrapolation. Although it may be reasonable to conclude that there is a larger number of much smaller tube defects than shown in flaw distributions obtained from eddy current inspection data, based on NDE data, little can be said for the morphology of these flaws.

To determine if a significant population of short, through-wall flaws exist within operating steam generators, metallographic examination data from tubes removed from inservice steam generators (i.e., pulled tubes) were reviewed. The pulled tube data assessment included a review of data for several different modes of steam generator tube degradation: (1) axial primary-water stress corrosion cracking (PWSCC) at dented tube support plate (TSP) intersections, (2) outside-diameter stress corrosion cracking (ODSCC) at TSP intersections and tubesheet expansion-transitions, (3) PWSCC at tubesheet expansion-transitions, and (4) pit-like intergranular attack (IGA) indications. Considering that the focus of most tube steam generator pulls is obtaining information on larger, more potentially significant defects, the amount of metallurgical data on short flaws is limited and sometimes biased. Nevertheless, some data have been reported to the NRC that include details on the shorter, less significant steam generator tube degradation.

One set of data examined in this study included lengths for PWSCC flaws at tubesheet roll transitions. The focus of the report from which the data were obtained was to quantify the leakage through axial PWSCC flaws [1]. Therefore, the data presented in the report only included flaw lengths associated with through-wall cracks that leaked during testing. It is likely that a number of short, part through-wall flaws existed in the tube sections examined in the PWSCC leakage study. However, because these cracks did not leak, these data would not normally be reported in such a study.

Discussion

The threshold of detection for eddy current inspection methods is particularly sensitive to the volume of material affected [2]. That is, if a sample of flaws has similar electrical and geometric properties (i.e., orientation), then the eddy current response will largely be a function of the relative size of each indication. Results from actual inservice inspections of steam generator tubes indicate that eddy current inspection techniques are very capable of detecting part through-wall defects in steam generator tubes with lengths on the order of 0.25 inches. Although eddy current techniques can detect shorter flaws, it is assumed for this investigation that all through-wall flaws exceeding 0.25 inches in length would have a high degree of detectability.

Surface flaws can take on an infinite number of geometries with varying lengths and depths. One approach used in fracture mechanics to simplify the description of flaw geometry is to assume a semi-elliptical flaw shape [3]. With this assumption, a surface breaking flaw can be described by two parameters, the length along the surface and the through-wall depth. Given that the longest flaw length considered here is 0.25 inches and assuming a maximum tube wall thickness of 0.050 inches, the bounding flaw aspect ratio (length-to-depth ratio) is five. Since the focus of this assessment is on short, through-wall cracks, such flaws would have aspect ratios less than this bound.

Many data were available from destructive examinations of tubes containing ODSCC indications at TSP intersections. Although this mode of degradation is confined within the TSP and leakage from these flaws could not impinge on neighboring tubes, this mode of degradation is similar that found in other areas of the tube. ODSCC at TSPs and cracking that develops above the tubesheet in the sludge pile both develop in low stress, tight crevice tube areas. The morphology of the flaws that develop in these areas is expected to be similar. Therefore, it is assumed that conclusions can be drawn for sludge pile cracking based on an examination of the data that exists for cracking within TSPs.

EVALUATION OF PULLED TUBE DATA

Data shown in Figure 3 represent the results obtained from destructive examinations of tubes removed from Diablo Canyon, Unit 1, in 1995 [4]. The tubes contained OD and ID axial and OD circumferential defects located at dented and non-dented drilled tube support plate intersections. The tube pulls targeted the larger, more significant indications of degradation. However, not all of the flaws presented in Figure 3 were detected by eddy current inspection techniques prior to pulling. With one exception, all of the flaws had lengths on the order of 0.2 inches and maximum crack depths significantly less than through-wall. The only flaw that penetrated through the entire tube wall was the long axial indication that was easily identified with bobbin coil and rotating pancake coil probes.

Figure 3. Diablo Canyon Tube Pull Summary
[contains proprietary information]

The licensee characterized the data in Figure 3 as the macrocrack analysis results. Additional flaws were identified in the destructive examinations; however, the level of detail was less than that provided for the above data. Some useful information was provided on the overall number and depth of less significant flaws identified from the destructive examination. For instance, at one TSP intersection an estimated 33 OD cracks were observed at the mid-TSP level. Additional cracks were seen at other locations as well. The maximum and average crack depths determined from transverse sectioning of the tube specimens were 24 and 11-percent through-wall, respectively. Therefore, although a sizable number of flaws were seen in the metallographic analyses, all of these were shallow. A similar observation regarding flaw depths was made at one other TSP intersection. Four other TSP intersections were either defect free or found to have only a small number of shallow OD defects.

The Steam Generator Degradation Specific Management (SGDSM) database contains a substantial amount of pulled tube data on ODSCC at TSP intersections. The data in this database are classified by tube diameter, either 3/4-inch or 7/8-inch. Figure 4 contains the information available in the database on flaws that leaked during leak rate testing [5]. The ID/OD flaw ratio represents the total length of crack on the ID surface divided by the OD length. Since the flaws originated from the OD, these values are generally less than unity. As seen from the figure, many of the data were obtained from laboratory grown cracks from model boiler tests. In addition, these flaws appear to have more extensive through-wall degradation than that observed in the pulled tube samples. As seen from the data no through-wall cracks were identified with OD lengths less than approximately 0.25-inches. Therefore, the 7/8-inch pulled tube data in the SGDSM database does not support the assertion that short through-wall flaws exist in large numbers. In fact, based on the available data, no short through-wall flaws were present. One note, the data in this figure represents only those tubes that leaked during testing. Therefore, the samples used in the SGDSM database may contain through-wall cracks that did not leak at elevated pressures.

Figure 4 Geometry of ODSCC TSP Flaws
(7/8" Diameter Tubes)
[contains proprietary information]

Figure 5 represents a similar set of data for 3/4-inch diameter tubes [5]. The 3/4-inch database is more extensive than that for 7/8-inch tubing. However, the conclusions that can be drawn from the data are similar. Once again through-wall cracks were only observed for those flaws that were longer in length, and with one exception for the pulled tube samples, leakage was not observed except for axial defects exceeding 0.45 inches in length.

Figure 5 Geometry of ODSCC TSP Flaws
(3/4" Diameter Tubes)
[contains proprietary information]

Pulled tube destructive examinations of ODSCC samples removed from the Trojan steam generators included a detailed assessment of microcracks in a number of the samples. Figure 6 shows some of the length-depth flaw data reported for these tubes [6]. Although Trojan data are included in the SGDSM database, data shown in Figure 6 are not included in Figure 4. One observation from Figure 6 is that the metallographic examinations identified numerous short, part-through-wall cracks. Despite that large number of flaws present, no cracks penetrated through to the ID surface.

Figure 6 Trojan Microcrack Data

[contains proprietary information]

The examination of Trojan microcrack data is a conservative approach to verify the likelihood of occurrence of short, through-wall flaws. Specifically, since these microcracks are generally part of a larger network of cracks, these flaw networks are much more detectable by NDE techniques. If the flaws shown in Figure 6 are representative of the population of TSP cracks and other flaws that develop in low stress crevice locations, then it appears unlikely that there is a significant number of short (i.e., < 0.25 inch) flaws that have the potential for leakage.

Figure 7 includes the same data as Figure 6 with the through-wall depth shown in absolute dimensions rather than as a percentage of the wall thickness. What becomes more apparent in examining Figure 7 is that most of the flaws had an aspect ratio greater than one. This suggests that ODSCC at TSPs is more likely to grow in length than in depth. Therefore, complete through-wall flaw penetration is not expected except for longer (i.e., >0.25 inch) defects.

Figure 7 Trojan Microcrack Data
[contains proprietary information]

Most of the data presented to this point have come from degradation located at TSP intersections. Although these flaws may not be a significant consideration during severe accident conditions due to the restraint provided by the surrounding TSP, it is assumed that the morphologies observed for these defects is representative of degradation in crevice locations with no elevated stresses. Such degradation could exist in the sludge pile region above the tubesheet secondary face remote from the expansion-transition or within the crevice of a partially expanded tubesheet. Based on data shown in Figures 4, 5, and 6, it appears unlikely that short, through-wall flaws exist in TSP crevices. Given the similarity between the flaws develop in TSP crevices and within the sludge pile region, there is a low likelihood that short, through-wall cracks exist in the sludge pile area of steam generator tubes.

IGA degradation in the freespan areas has recently become an issue of increased attention. Although a relatively small number of tubes have been identified with these indications, pulled tube data suggest that many of these indications may go undetected with current NDE technology. Tubes were removed from the Crystal River, Unit 3 (CR-3), steam generators to assess the burst and leakage integrity of IGA degradation indications located above the lower tubesheet. Some of these data are shown in Figure 8 [7]. Tube defects were identified in the destructive examinations by visual inspection of the OD tube surface. Therefore, all potentially significant degraded areas should have been located in the metallographic examinations. Most of these flaws were not identified by eddy current inspections completed prior to or after the tube pulls.

Figure 8 Patch IGA Indication Geometry
Crystal River 3 Pulled Tube Data
[contains proprietary information]

One observation from the data in Figure 8 is that IGA defect length and depth appear related. By extrapolating a regression to these data, a through-wall defect would not be expected until the length of the defect is approximately 0.16 inches. Although this value falls below the 0.25 inch threshold established for this investigation, what does remain clear is that the depth of IGA is related to length. Hence, one would not expect to identify a sizable population of similar IGA degradation exhibiting short lengths and extensive through-wall penetration. Since IGA is essentially three-dimensional degradation (i.e., volumetric versus crack-like), the detectability of these indications will increase significantly with increasing length, depth, and width. Some of the indications detected by eddy current during the CR-3 inspections were found to have axial lengths as low as 0.02 inches. Despite the short lengths of some of the degraded areas, the bobbin coil probe was able to detect some of these indications. In addition, the large volume associated with deeper defects facilitates their detection during eddy current examinations.

Based on previous experiences with IGA degradation, this mode of degradation appears more resistant to development of a leak path than other forms of tube wall damage (e.g., ODSCC). Although the licensee for CR-3 has identified a several hundred of these IGA patches in its steam generators, primary-to-secondary leakage has not been attributed to these indications. In addition, a number of tubes that recently underwent in-situ pressure testing did not leak. Pulled tube specimens obtained from CR-3 were burst tested to assess their structural and leakage integrity. The tubes were able to retain internal pressures on the order of 10000 psi without any measurable leakage prior to burst. Therefore, the IGA degradation in the tubes removed from CR-3 would not challenge the structural integrity of these tubes during design basis depressurization events.

Figure 9 presents data obtained from a report by the Electric Power Research Institute (EPRI) on axially-oriented PWSCC degradation located in the roll transition zone (RTZ) [1]. The data only include through-wall cracks from pulled tube samples to assess leak rates. The ID flaw lengths ranged from 0.06" up to 0.3". However, most of the data are located in the interval between 0.1 to 0.2 inches. Given that this is the area of interest for this assessment, the PWSCC data from the EPRI report is a useful resource. The vertical axis in the figure represents the ratio of the OD to ID crack length as measured from destructive examinations.

Figure 9 PWSCC Flaws at Roll Transitions
[contains proprietary information]

One notable difference between the data presented in Figure 9 and that shown in the previous figures is that it is widely scattered with no apparent relationship between the crack length measured on the OD of the tube to that on the tube ID. In the previous figures, the flaw depths generally increased as the crack length on the surface increased. In general, this behavior is typical for surface initiated flaws. However, the data in Figure 9 do not follow this convention. If flaw depth is dependent on the length, or vice-versa, one would expect to find an increasing flaw length ratio with increasing ID length, but this is not the case with the PWSCC data reported by EPRI. Since the data were obtained from only nine pulled tube samples, it is reasonable to conclude that there is the potential for short, through-wall cracks to exist in the RTZ. Also, given that leakage was observed from each of the pulled tube samples, one cannot rule out the possibility that the leakage originated from some of the shorter cracks.

CRACKING IN EXPANSION-TRANSITION ZONES

Some steam generator designs secure the tubes in the tubesheet with a partial depth roll expansion. This is a hard roll that begins at the primary face of the tubesheet and extends several inches along the tube before terminating within the tubesheet bore. B&W once-through steam generators (OTSGs) and a small number of Westinghouse models have partial depth expanded tubes. Because the hard roll terminates within the tubesheet, primary jet impingement of hot gases on adjacent tubing during severe accident conditions is precluded. Incidently, axial cracking at RTZs is a significant mode of tube degradation for Westinghouse steam generators with partial depth expansions. Inspections of roll expansions in B&W OTSGs and pulled tube destructive examination results have identified only a small number of cracks in the RTZ to date.

The geometry of RTZ cracking appears different from the ODSCC at TSPs and IGA degradation discussed previously. Further discussion on the differences between the cracking in these areas is warranted to address the potential existence of short, through-wall flaws. Steam generator tube RTZs are highly stressed areas of the tube. Analytical calculations and experimental measurements have concluded that the stresses in these areas are greater than in other stressed areas of the tube such as dents and tight radius U-bends. The residual stress field in the transition zone introduced from the rolling process produces stress levels much greater than that from primary-to-secondary differential pressures.

In order to eliminate the crevice areas of partial depth roll expanded tubes, steam generators later utilized a full depth hard roll expansion. The primary difference between these two types of expansion joints is that the RTZ exists at the top of the tubesheet in full roll designs. Despite the elimination of the crevice region, a dominant mode of tube degradation in RTZs continues to be axial cracking. EPRI completed one investigation into RTZ cracking for tubes removed from the Ringhals 2 steam generators [1]. The Ringhals 2 steam generators were the Westinghouse Model 51 design with a partial depth hard roll expansion. Destructive examinations of pulled tubes concluded that most of the axial PWSCC was located within the transition zone extending up toward the upper transition region. Since the RTZ for full depth expanded plants exists at the top of the tubesheet, it is reasonable to conclude that RTZ cracking is similar to cracking in the freespan area of the tube with regard to primary-to-secondary leakage. Therefore, the existence of many short, through-wall axial cracks that could be a factor under severe accident conditions is possible for steam generators with full depth hard roll expansions.

There are currently nine plants in operation (33 steam generators) with full depth hard roll expansions. However, the utilities for five of these plants (19 steam generators) will replace the steam generators within the next four years with designs that incorporate more corrosion resistant materials and tubes with expansion-transitions that are less susceptible to stress corrosion cracking. Although roll transitions have the most severe state of stress with respect to the development SCC for all methods of expanding tubes into tubesheets, a closer investigation of the mechanisms of cracking within explosive and hydraulic expansions is warranted.

Axial cracking in explosively-expanded steam generator tubesheet joints has been identified by a number of utilities. This applies to steam generators designed by Westinghouse and Combustion Engineering. Limited data are available on the location of axial cracking relative to the top of the tubesheet for these steam generators. However, a recent assessment of inspection findings in the Salem, Unit 1, steam generators concluded that all but nine of 177 axial flaws (ID and OD) were located below the top of the tubesheet [8]. Although this conclusion is based on NDE inspection results and not pulled tube destructive examinations, it is clear that the axial cracking in the explosively expanded tubes at Salem favors an area of the expansion-transition that is located below the tubesheet secondary face. Therefore, if short, through-wall cracking similar to that found in RTZs develops in explosively expanded tubesheet joints, then it is most likely to occur in an area of the tube where primary-to-secondary leakage would not impinge on neighboring tubes.

The current method of forming the tubesheet expansion joint is to hydraulically expand the tubing into an interference fit within the tubesheet bore. The majority of steam generators that entered into service since the early 1980's were fabricated with hydraulic tubesheet expansion joints. Cracking experience in hydraulically expanded tubing has been limited to date. There are no operating plants with these expansion joints that have identified significant levels (i.e., large numbers) of flaws in the transition zone. This is believed to be a reflection of the lower much stresses introduced from the hydraulic expansion process [9]. Nevertheless, given enough time, these tubesheet expansion joints could develop cracking similar to degradation presently observed in steam generators with rolled or explosively-expanded tubesheets. By design, however, the expansion-transition in hydraulically expanded tube terminates below the top of the tubesheet. Therefore, primary-to-secondary leakage from short, through-wall cracks in a hydraulic expansion-transition should impinge on the bore of the tubesheet and not an adjacent tube.

Circumferentially-oriented SCC in tubesheet expansion-transitions affects a number of steam generators. Because this mode of degradation cannot be adequately detected using bobbin coil eddy current probes, utilities employ more sensitive inspection probes to assess the condition of tubesheet expansions. In addition, numerous tubes have been removed from steam generators in order to assess structural and leakage integrity as well as eddy current sizing capabilities. Entergy, the licensee for Arkansas Nuclear One, Unit 2, (ANO-2) submitted a report that included a correlation relating the cracked percentage of tube wall (percent degraded area (PDA)) to the nominal through-wall crack length [10]. Although the NRC staff has not evaluated the validity of the data supporting the correlation, the relationship indicates that a through-wall crack would be expected when PDA is on the order of 10-percent. Using this data along with the dimensions of the tubes at ANO-2, and assuming a single, semi-elliptical flaw geometry, a through-wall crack would be anticipated when the overall crack length exceeds approximately 0.28-inches. Circumferential flaws with such lengths should be detected with present day inspection technology.

It has been noted in many instances that circumferentially-oriented cracks in expansion-transitions are actually a series of shorter length flaws separated by small ligaments. However, due to limitations in eddy current inspection technology and for additional conservatism in assessing structural integrity, small ligaments are often ignored. If these ligaments are accounted for in determining the actual crack lengths, the nominal flaw aspect ratio corresponding to a through-wall crack should decrease. Destructive examination results of pulled tubes containing circumferential cracking do not show evidence of a large number of short and deep cracks, but these examinations have identified networks of circumferentially oriented flaws. The presence of multiple flaws at the same axial tube location improves the chances of detecting this mode of degradation. Therefore, it is unlikely that numerous tubes are left in service with undetected, short, through-wall, circumferential flaws.

CRACKING IN OTHER HIGH STRESS LOCATIONS

Cracking in tubesheet expansion joints has been responsible for a large number of tubes removed from service due to corrosive degradation. This is primarily due to the higher susceptibility to SCC at this location because of the elevated residual tube stresses and temperatures. Tube small radius U-bends and TSP denting are other locations where high stresses favor the initiation of SCC [9]. In fact, early operating experience indicates that cracking in short radius U-bends and RTZ cracking appear after a similar length of service. This would indicate that stress levels (although not necessarily stress distribution) in these two regions are comparable. Several studies have concluded that residual stresses in the U-bend region are similar in magnitude to those in expansion-transitions. Although dented locations also exhibit elevated residual stresses, the potential existence of short, through-wall flaws at these locations will not be addressed here because primary-to-secondary leakage from these flaws would impinge on the adjacent structure partially responsible for the development of the dent.

Cracking in low radius U-bends has been primarily associated with Westinghouse designed steam generators with low temperature mill annealed (LTMA) alloy 600 tubing. This mode of cracking has caused some utilities to take preventative measures in an attempt to mitigate the problems from U-bend degradation. Preventative plugging of low row tubing and additional heat treatments are two examples of these measures. Historically, U-bend cracking has been a difficult mode of degradation to detect. Despite the problems associated with managing this mode of degradation, only limited pulled data is available. Because cracking in the U-bend cannot be extracted by removing the tube from the primary inlet plenum (i.e., tube pull), specimens must be "harvested" from the secondary side of the steam generator. Due to the complexity of this operation, the removal of U-bend tube sections has been attempted only on a few occasions.

Portland General Electric Company, the licensee for the Trojan Nuclear Plant, completed an effort to evaluate U-bend cracking in the early 1980's. Twenty-six row one tubes and three row two tubes were removed for metallographic analysis. The destructive examination of the row 2 tubes did not reveal the presence of any cracking. Although the Trojan study did not include the lengths and depths of U-bend flaws, it did discuss some of the findings on crack geometry identified from tubes removed from Surry 1 and Turkey Point 4. PWSCC found in tubes removed from these units were characterized as "intergranular and extended halfway through the tube wall (aspect ratio of 4)." This observation leads one to the conclusion that, on average, through-wall cracking would not be expected until the flaws have lengths on the order of 0.2 inches. Therefore, U-bend flaw aspect ratios would seem to be greater than cracks observed at other high stress locations such as roll transitions. Thus, the potential for short, through-wall cracks in tube U-bends appears unlikely based on these findings.

One additional note on low radius tube U-bend cracking is that only a limited number of plants are currently affected by this mode of degradation. Recirculating steam generators fabricated by Combustion Engineering have not experienced U-bend cracking similar to that observed in Westinghouse designed steam generators. With one exception, U-bend cracks have only been found in LTMA alloy 600 tubing. Once-through steam generators are a single pass design that do not have U-tubes. Therefore, this mode of degradation cannot occur in these steam generators. These observations further narrow the population of plants that are affected by U-bend cracking.

OTHER FORMS OF STEAM GENERATOR TUBE CRACKING

Recently, indications of freespan cracking have been identified in both Combustion Engineering and Babcock & Wilcox designed steam generators. The industry has generally described the mode of degradation as long indications of IGA. Pulled tube destructive examinations have confirmed that the depths of detected indications are generally less than 50-percent of the total wall thickness. In order to assess the structural and leakage integrity of freespan cracking, licensees have removed several tubes with indications and completed burst and leakage testing. The data indicate that short, through-wall freespan cracks are not present for this mode of degradation. In addition, burst test results have demonstrated that these tubes have considerable margins for tube structural integrity. Burst pressures for axial freespan cracks in tubes removed from the Calvert Cliffs steam generators (CE) were measured on the order of undegraded tubing [11]. Leakage has not been attributed to these flaws during operation, during in-situ pressure testing, or observed in testing conducted after tube removal.

CONDITIONS NECESSARY FOR SHORT, THROUGH-WALL CRACKS

This assessment identified one location susceptible to short, deep steam generator tube flaws. Tubes with hard roll expansions have the conditions suitable for the development of this mode of cracking. The unique geometry of RTZ cracking is likely a consequence of the elevated stresses introduced into the tube by the rolling process. No other locations in a steam tube will have stress levels resulting from the steam generator fabrication processes equal to or greater than those in RTZs. Although the development of RTZ flaws is result of a combination of susceptible materials, environmental conditions, and residual tube stresses, the uniqueness of the morphology of the cracking originates primarily from the state of stress in the material. It may be reasonable to assume that extremes in the other factors for stress corrosion cracking could also lead to flaws with atypical geometries.

With respect to material susceptibility to stress corrosion cracking, steam generator tube materials are relatively homogeneous. It is unlikely that extremes in tube material chemistry would be isolated to areas less than a fraction of an inch giving rise to short, through-wall cracks. Crevices, favorable for the development of stress corrosion cracking, can exist at several locations along a steam generator tube: (1) tubesheet crevices, (2) TSPs, (3) U-bend supports, (4) in the preheater region for certain steam generator designs, and (5) above the tubesheet in the sludge pile. Deposit buildup on tube surfaces has also created freespan crevices that led to cracking, but the number of reported occurrences for this type of degradation has been limited to date. Crevice conditions could potentially exist over a very short length of tubing. However, in general, each of the tube crevice examples given above have length scales larger than the 0.25 inch threshold considered in this study. In addition, crevices primarily result from the tube lying adjacent to a support structure on the secondary side of the steam generator. Potential leakage through any defects originating within the crevice should impact the structure responsible for creating the crevice rather than a neighboring tube. Therefore, it is likely that short, deep flaws with the potential for freespan leakage will only initiate in areas of the tubing where elevated residual stresses exist.

Conclusion

In order to evaluate the plausibility of the existence of a significant population of short (i.e., < 0.25 inch), through-wall steam generator tube cracks destructive examination data were reviewed to assess flaw geometries. The purpose of this assessment was to determine if undetected cracking could lead to primary-to-secondary leakage and subsequent hot gas jet impingement on adjacent undefected tubes under the high temperature and pressure conditions postulated for severe accident conditions. Pulled tube destructive examination results were reviewed for: (1) axial ODSCC at TSP intersections, (2) circumferential cracking at expansion-transitions, (3) pit-like IGA indications in the freespan region, (4) PWSCC, and (5) axial freespan cracking in CE and B&W designed units.

Based on the data presented herein, the existence of short, through-wall cracks that could become significant during severe accident conditions does not appear likely for all modes of steam generator tube degradation with the exception of PWSCC in roll transitions. However, the presence of the tubesheet in partial depth roll expanded tubing and the scheduled replacement of a number of steam generators with rolled tubes susceptible to these cracks limit the number of tubes that could lead to primary-to-secondary leakage affecting surrounding tubes under severe accident conditions. Furthermore, current practice is to remove defects of this type from service upon detection and the detection sensitivity has increased significantly in the last few years.

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1. The ACRS letter to the EDO dated October 10, 1997 "Resolution of the Differing Professional Opinion Related to Steam Generator Tube Integrity" stated the ACRS conclusion that the DPO issues were accurately summarized by the DPO document. The same ACRS letter also indicated that the DPO author thought the issues were adequately summarized.

2. In some cases, the figures discussed in this report contained proprietary information and have therefore been removed from this report to enable this document to be made publicly available.