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Steam Generator Degradation Specific Management

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Committee for Alternate Repair Limits

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SGDSM - WORK IN PROGRESS

The attached report describes the SGDSM concept and implementation. An addition to the report is planned but not yet available. This addition will not change the approach but will rather provide additional detail. This planned addition is described below.

1. Test data are an important part of the SGDSM approach and such test data will continue to become availabe from pulled steam generator tubes and laboratory specimens. Work is underway to provide generic guidelines for leak and burst testing of pulled steam generator tubes or laboratory specimens for use in SGDSM. When these guidelines are completed and reviewed they will be included in this SGDSM report.

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ABSTRACT

The purpose of this document is to provide an industry accepted generic prescription for steam generator degradation specific management. Degradation specific management is an option that a utility may follow for evaluating, repairing and/or monitoring tube degradation if certain specific conditions exist and requirements are met.

In addition to this report which summarizes the elements of the degradation specific management option, degradation specific management requires development of supporting reports covering generic tube inspection guidelines and degradation specific inspection and repair criteria. A report on generic tube inspection guidelines and reports covering two specific degradation mechanisms (roll transition PWSCC and ODSCC at tube support plates) are referenced and summarized in this document. This report also provides guidance and requirements for development of additional degradation specific inspection and repair criteria reports in the future.

This document and existing supporting reports constitute the degradation specific management option for which NRC review and approval is requested.

Implementation of degradation specific management requires specific utility actions. These actions include:

- modification of plant Technical Specifications to allow the degradation specific management option.
- development of a plant degradation specific management program plan.
- development and implementation of steam generator tube inspection procedures in conformance with industry tube inspection guidelines and degradation specific inspection and repair criteria reports.
- plant specific calculations to establish repair limits and demonstrate limited leakage under accident conditions.
- participation in an on-going industry-wide degradation specific management data base and related support for developing, implementing and maintaining degradation specific management
- repairing tubes and reporting program results to the NRC.

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Definitions

Allowable Accident Leak Rate - The primary to secondary leak rate that assures that the dose rate at the site boundary during an accident is less than the 10CFR100 limits under postulated faulted loading conditions.

Degradation Specific Damage - Steam generator tube degradation which is caused by a unique degradation mechanism and which is confined to particular locations within the generator.

Degradation Specific Management - A generic approach for establishing and implementing degradation specific inspection, evaluation and repair criteria.

Degradation Specific Management Data Base - An on-going industry-wide data base and related support for developing, implementing and maintaining degradation specific management.

Degradation Specific Inspection and Repair Criteria Reports - Documents which provide inspection, evaluation and repair criteria for degradation specific damage. When the degradation specific damage affects a number of steam generators, industry degradation specific inspection and repair reports may be developed. Examples of industry reports are references (4) and (5).

Industry Tube Inspection Guidelines - Generic guidelines for steam generator tube inspections developed by the industry: PWR Steam Generator Examination Guidelines; Revision 2 (6).

Plant Degradation Specific Management Program - The combination of documents and activities carried out by a utility to implement degradation specific management.

Plant Degradation Specific Management Program Plan - The plan prepared by a utility to implement the degradation specific management option. The plan identifies the industry tube inspection guidelines to be utilized and the degradation specific inspection and repair criteria reports to be implemented. The plan will include a commitment to develop plant specific inspection procedures in conformance with these documents. The plan will include the plant specific structural limit and allowable accident leak rate limit. The plan will define plant specific calculations to be performed and reports to be provided to the NRC following each inspection.

Plant Specific Calculations - Calculations required to implement degradation specific inspection and repair criteria reports, evaluate inspection results and define repair limits.

Plant Specific Tube Inspection Procedures - Plant specific tube inspection procedures which conform with industry tube inspection guidelines, degradation specific inspection and repair criteria reports and other inspection requirements defined in this document.

Plant Specific Inspection Report - A report prepared following each inservice inspection where degradation specific management inspection, evaluation and repair limits have been utilized. The report includes the tube repair limit, examination results, repaired tubes, tubes with indicated degradation to be left in service and calculated leakage under accident conditions.

Tube Inspection Parameter - The inservice NDE inspection parameter used to define the extent of degradation specific damage, e.g., bobbin coil voltage for ODSCC at tube support plates.

Tube Repair Limit - The value of the tube inspection parameter which, if exceeded, required tube repair. The tube repair limit is based on the tube structural limit plus margins for NDE uncertainty and degradation growth.

Tube Structural Limit - The value of the tube inspection parameter limit that shall not be exceeded in service. The tube structural limit is based on tube load, material properties and appropriate safety margins.

EXECUTIVE SUMMARY

Various types of corrosion degradation (wastage, pitting, stress corrosion cracking, etc.) have been diagnosed in Alloy 600 steam generator tubes at various locations within pressurized water reactor (PWR) steam generators. When tube repair limits based on eddy current measured degradation depth alone are applied, many tubes may require repair that is unnecessary from either a safety or reliability standpoint. Degradation specific repair limits can be used in combination with enhanced inspection to maintain steam generator reliability. Degradation specific inspection and repair criteria can be used to develop the most cost-effective means to maintain plant safety and acceptable steam generator reliability.

This report provides a prescription for development and implementation of a degradation specific management program. This program consists of the requirements contained in this document, generic steam generator tube inservice inspection guidelines, degradation specific inspection and repair criteria and certain plant specific actions. Degradation specific management is based on a combination of enhanced inspection, limits on the degree of degradation in individual tubes, and limits on the number of degraded tubes allowed to remain in service. It is anticipated that utility submittals that deviate from this prescription will require a longer, more detailed NRC review.

This document has been prepared by a committee of U.S. and international industry participants who are experts on technical and licensing issues associated with development and implementation of a steam generator tube inspection and repair program. This document contains the committee's prescription for use by utilities, industry, and regulatory groups for the development, implementation, and evaluation of degradation specific inspection and repair criteria. Implementing the requirements of this document will assure that the requirements of the General Design Criteria of Appendix A to 10CFR50 are met.

Degradation specific management includes the following:

- Inspection methods that can reliability detect the specified degradation mechanism.

- Determination of the measurement error (including analyst error) associated with degradation specific inspection procedures.

- A NDE measured parameter (e.g., crack depth, crack length, eddy current voltage) that can be used to assess the structural and leakage integrity of degraded tubes.

- An experimental data base to correlate the rupture strength of degraded tubes with the NDE measured parameter.

- Repair criteria that will ensure adequate margins against tube rupture consistent with regulatory requirements.

- An experimental data base to correlate the leak rate from degraded tubes with the NDE measured parameter.

- Repair criteria that ensure leakage during postulated faulted loads will result in off-site doses that are less than the 10CFR100 limits.

- An up-to-date degradation growth rate data base.

- An on-going industry-wide degradation specific management data base and related support for developing, implementing and maintaining degradation specific management.

- A degradation specific inspection and repair criteria report that provides the bases and implementation procedure for each degradation specific mechanism.

Implementation of a degradation specific management program on a plant specific basis requires the following utility action:

- Modification of plant Technical Specifications to incorporate a degradation specific management option.

- Implementation of an allowable leak rate limit of 150 gpd per steam generator during normal plant operation.

- Development of a plant degradation specific management program plan.

- Development of plant specific tube inspection procedures in conformance with industry tube inspection guidelines and degradation specific inspection and repair criteria reports.

- Confirmation of the degradation specific mechanism using inservice inspection criteria and/or pulled tube examination results.

- Enhanced inspection of the tubes in the region of the steam generator affected by the degradation specific mechanism using the inspection techniques identified in the degradation specific report for that mechanism.

- Determination of a plant specific degradation growth rate. If the inspection and repair criteria for a degradation specific mechanism are being implemented for the first time, the generic growth rates can be used until sufficient plant specific data have been obtained.

- Calculation of a tube structural limit in terms of an NDE measured parameter that corresponds to an acceptable margin against tube rupture using plant-specific data on material properties, and normal and postulated faulted loads.

- Calculation of a tube repair limit based on the tube structural limits, degradation growth rates, and inspection measurement error.

- Calculation of the leak rate expected during postulated faulted loads from the tubes that remain in service to ensure that the leak rate results in off-site doses that are less than the 10CFR100 limits.

- Participation in an on-going industry-wide degradation specific management data base and related support for developing, implementing and maintaining degradation specific management.

- Repairing tubes and reporting program results to NRC.

Implementation of these elements constitutes a defense - in - depth approach developed to ensure adequate levels of safety and compliance with the General Design Criteria (GDC) in 10CFR50, other applicable regulatory requirements, and the ASME Code. The inspection and repair criteria developed and implemented as described in this document ensure adequate margins against failure and excessive leakage consistent with the requirements of the ASME Code, and the applicable GDC and other regulatory requirements and guidelines.

The prescription described in this report applies to distinct degradation specific mechanisms for which degradation specific inspection and repair criteria have been or will be developed by various vendor or industry groups and implemented by utilities on a plant specific basis. This prescription has been applied previously to develop degradation specific criteria for primary water stress corrosion cracking in expansion zone roll transitions (EZ PWSCC) and for stress corrosion cracking on the tube outer diameter at tube to tube support plate intersections (ODSCC). This prescription need not apply to previously developed degradation specific inspection and repair criteria that already have received plant specific licensing approval (e.g., P* and F*).

Section 1 of this document provides the background and summarizes the approach used to develop degradation specific management inspection and repair criteria. Compliance with ASME Code and regulatory requirements and guidelines are discussed in Section 2. Section 3 provides steam generator tube examination guidelines that are to be implemented when a degradation specific management program is implemented. Section 4 prescribes the steps needed for the development and implementation of a degradation specific management program, and summarizes currently available degradation specific management criteria. Mandatory plant specific analyses and implementation procedures for degradation specific management programs are presented in Section 5. Appendix A is a sample license amendment submittal for a steam generator degradation specific management program plan.

A summary of the elements of degradation specific management are shown in Figure S-1.

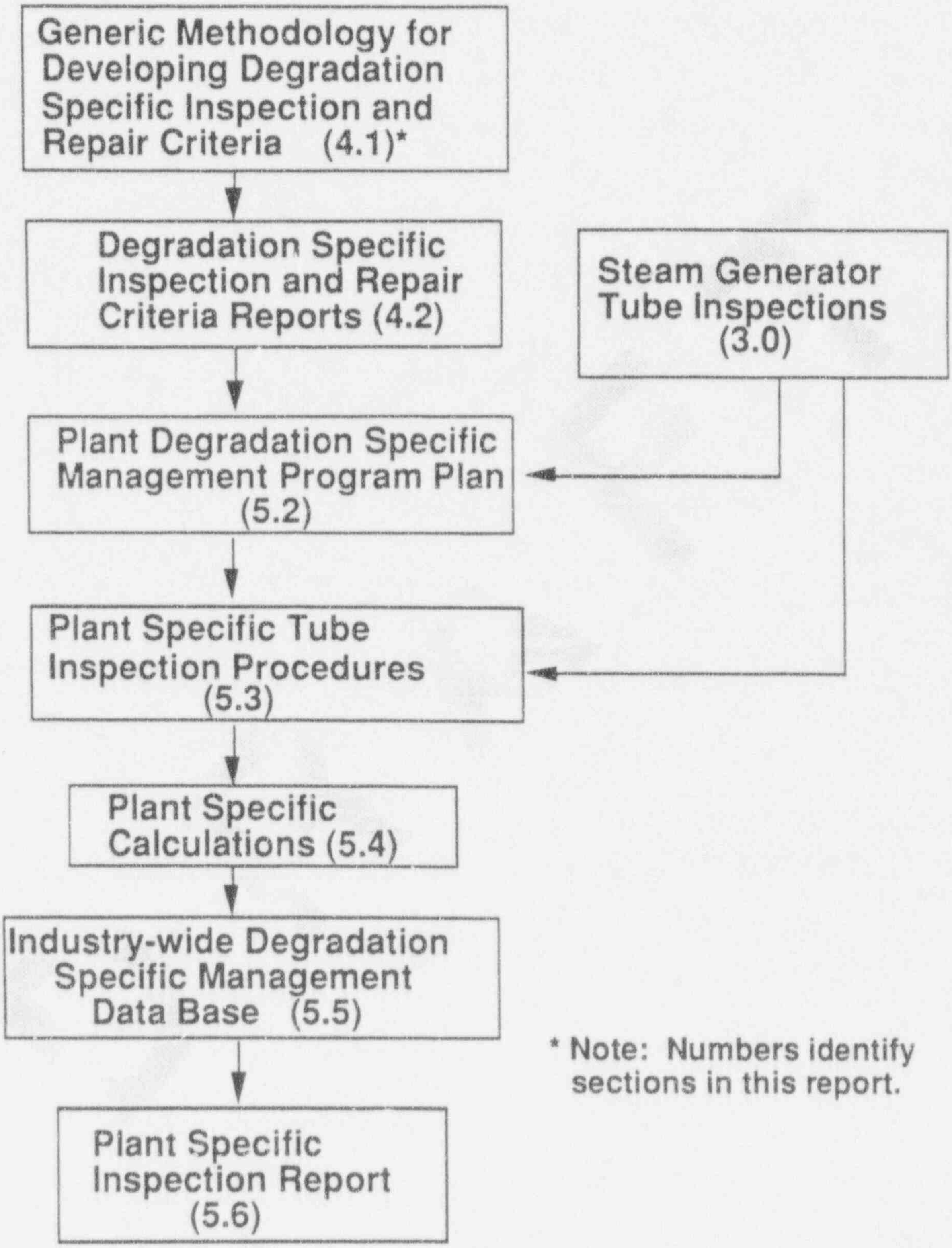

Figure S.1: Elements of Degradation Specific Management

Section 1

INTRODUCTION AND OBJECTIVE

1.1 BACKGROUND

Over the last 20 years in the United States and internationally, steam generator tubes in PWRs have experienced various types of degradation of the tube wall. Initially, this degradation was wastage and pitting; more recently forms of stress corrosion cracking and intergranular attack have been found. Generally, degradation associated with corrosion has resulted in limited leaks and very few tube ruptures during service.

The inspection program and repair criteria are contained in the technical specifications for each operating plant. Normally, the technical specifications require the condition of steam generator tubes to be assessed periodically by inservice inspections typically performed using bobbin coil eddy current techniques. The frequency of inspection, the number of tubes inspected, and the inspected location depend on the degree of degradation found in the steam generator. Initially, 3% of the tubes are inspected randomly throughout the steam generator to assess the general condition of the tubes in the steam generator. If degradation is detected, the sample size is increased in locations where the degradation is diagnosed. If degradation is extensive, the sample size can expand to 100% of the tubes in the steam generator. In practice, more extensive inspections than required by Technical Specifications have been utilized following industry tube inspection guidelines (see Section 3).

For most plants, the current criterion for steam generator tube integrity requires steam generator tubes to be repaired when the defect depth is diagnosed by bobbin coil inspection to be equal or greater than 40%[1] of the tube wall thickness. The basis for this repair limit is a uniform wall thinning degradation mechanism, such as wastage.

Recent research indicates that a tube repair limit based on bobbin coil indicated defect depth is not appropriate for all forms of steam generator tube degradation. Consequently, several U.S. and international programs (1,2,3) recently have been directed toward defining inspection and repair criteria specifically developed for distinct, individual degradation mechanisms. This work has led to the previous development of generic inspection and repair criteria for two specific degradation mechanisms (4, 5).

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[1] A few operating plants have a depth limit of 50% of the tube wall thickness.

1.2 GOALS AND OBJECTIVES

The objective of this document is to provide a prescription the utilities and industry will follow for preparation of inspection and repair criteria pertinent to distinct, individual degradation mechanisms, and for the plant specific implementation of these criteria once they are developed. This uniform, industry developed methodology for implementing degradation specific inspection and repair criteria is consistent with current regulatory requirements, and is based on comprehensive analytical, test or plant data bases. Plant specific implementation of degradation specific management programs will increase steam generator reliability and maintain adequate safety margins.

By implementing a degradation specific management program that ensures increased steam generator reliability and maintains adequate safety margins, the following safety and economic benefits can be obtained:

- Reduce unnecessary tube repair.
- Reduce personnel radiation exposure.
- Minimize further reductions in thermal margins due to unnecessary tube repair.
- Achieve increased steam generator reliability and availability.
- Provide a well documented industry-wide practice for determining steam generator operability.
- Reduce NRC review time.
- Reduce plant outage time.
- Incorporate technical improvements and their understanding in a timely manner on an industry-wide basis.
- Maintain flexibility for long term repair options.

1.3 STRATEGIES FOR ASSURING TUBE INTEGRITY

Table 1-1 illustrates several possible strategies for ensuring the structural and leakage integrity of steam generator tubes.

One strategy would be to maintain the bobbin coil inspection sampling program and the 40%[2] of wall defect depth repair criteria now typically specified in the plant Technical Specifications. This strategy would be chosen if the steam generators have little in service degradation or if the original steam generators have been replaced with designs which incorporate industry recognized concepts that significantly reduce operational corrosion and mechanical damage. This is Option 1 in Table 1-1.

A second strategy would be to maintain the 40% of wall defect depth repair criteria now typically specified in the plant Technical Specifications, and implement the inservice inspection sampling strategy, data analysis methods, qualification of examination personnel and performance demonstration requirements of the PWR Steam Generator Examination Guidelines (6) described in Section 3 of this report. For example, this strategy could be chosen if the steam generator currently had little in service degradation, but contingencies were being considered for implementation of a degradation management program in the future. This is Option 2 in Table 1-1.

A third strategy would be to implement a degradation specific management program. Implementation of a degradation specific management program requires certain actions in combination with actions currently required by the Technical Specification depth based inspection and repair program. These actions are summarized generally here and are described in detail in Sections 4 and 5.

- The prior documentation of degradation specific methods for the inspection and evaluation of the degradation specific mechanisms. This methodology must have adequate analytical, test or plant data bases for justification of the inspection and repair criteria. The inspection and repair criteria must satisfy the requirements of the NRC General Design Criteria (GDC) and be consistent with necessary safety margins (e.g., Regulatory Guide 1.121).

- The owner shall implement without exception the generic degradation specific inspection and repair criteria for the identified mechanism to develop the plant specific inspection and repair limits. Generic inspection procedures shall be in conformance with industry tube inspection guidelines as defined in the PWR Steam Generator Examination Guidelines (6).

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2 Some plants have a depth based repair limit of 50%.

Table 1.1 Strategies for Assuring Steam Generator Tube Structural and Leakage Integrity.

<u>OPTION 1 - CURRENT DEPTH BASED INSPECTION AND REPAIR</u>

- Inspection Requirements
    * Current Technical Specifications
- Repair Requirements
    * Current Technical Specifications

<u>OPTION 2 - CURRENT DEPTH BASED INSPECTION AND REPAIR WITH AUGMENTED INSPECTION REQUIREMENTS</u>

- Inspection Requirements
    * Industry Tube Inspection Guidelines
- Repair Requirements
    * Current Technical Specifications

<u>OPTION 3 - DEGRADATION SPECIFIC MANAGEMENT</u>

- Inspection Requirements
    * Industry Tube Inspection Guidelines
    * Degradation Specific Inspection and Repair Criteria Reports
- Repair Requirements
    * Current Technical Specifications (for non-degradation specific damage)
    * Degradation Specific Inspection and Repair Criteria Reports
    * Plant Specific Calculations

Section 2

COMPLIANCE WITH REGULATORY REQUIREMENTS

This section identifies the regulatory guidelines and requirements that have been and should be used to develop degradation specific inspection and repair criteria for steam generator tubes.

2.1 COMPLIANCE WITH 10CFR50.55a

Paragraph 10CFR50.55a defines the inspection requirements (including evaluation of inspection results) for components in commercial water cooled reactor systems. The following paragraphs summarize these requirements for steam generator tubes.

Paragraph (g) (4) of 10CFR50.55a requires in part that:

> "Throughout the service life of a boiling or pressurized water-cooled nuclear power facility, components (including supports) which are classified as ASME Code Class 1, Class 2 and Class 3 shall meet the requirements, except design and access provisions and preservice examination requirements, set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda that become effective subsequent to editions specified in paragraphs (g)(2) and (g)(3) of this section and are incorporated by reference in paragraph (b) of this section to the extent practical within the limitations of design, geometry and materials of construction of the components."

This requirement includes the inspection and repair criteria included in Section XI of the ASME Code. An exception for inspection of steam generator tubes is included in paragraph (b)(2)(iii) of 10CFR50.55a. This exception states that:

> "If the technical specifications of a nuclear power plant include surveillance requirements for steam generators different than those in Article IWB-2000, the inservice inspection program for steam generator tubing shall be governed by the requirements in the technical specifications."

Section XI of the ASME Code allows flaws detected during inservice inspection to remain in service without additional evaluation if they are less than a certain size. These flaw sizes are defined in Article IWB-3500, Acceptance Standards, of the ASME Code. The Code acceptance standards for steam generators tubes are defined in IWB-3521 and are:

> "IWB-3521.1 Allowable Flaws for U-Tube Steam Generators. For single or multiple flaws or cracks, wastage, or intergranular corrosion in tubing of SB-163 material meeting the requirements of NB-2550 and having an r/t ratio of less than 8.70, the depth of an allowable O.D. flaw shall not exceed 40% of the tube wall thickness", and

> "IWB-3521.2 Allowable Flaws for Straight-Tube Steam Generators. In the course of preparation."

Although IWB-3521.1 specifically mentions that the evaluation standards for U-tube steam generators are for flaws initiating on the tube outer diameter and that acceptance standards for straight-tube steam generators are in the course of preparation, the 40% of wall thickness defect limit historically has been used by the NRC and industry for degradation for which corrosion induced cracking, wastage, or pitting are the diagnosed mechanisms. These acceptance standards are consistent with the safety margins used historically to demonstrate compliance with Regulatory Guide 1.121 for tubes with uniformly thinned walls.

Section XI of the ASME Code also allows flaws larger than those specified in the acceptance standards of IWB-3521 to remain in service, provided a detailed evaluation is performed subsequent to inspection to demonstrate that adequate safety margins will be maintained during the next operational interval. These flaw sizes are defined in Article IW-3600, Analytical Evaluation of Flaws.

Previously, Code criteria for evaluation of flaws in reactor vessels have been developed and are contained in Article IWB-3610. More recently, Code criteria for evaluation of flaws in piping have been developed and are contained in Articles IWB-3640 and IWB-3650 for austenitic and ferritic piping, respectively.

The Code criteria for steam generator tubes is specified in IWB-3630, Acceptance Criteria for Steam Generator Tubing, which states:

> "Evaluation of cracks, wastage, or intergranular corrosion in steam generator tubes that exceed the allowable flaw standards of IWB-3521 shall be performed by analyses acceptable to the regulatory authority having jurisdiction at the plant site."

This document presents a prescription for the development of evaluation procedures and repair criteria that can be used for steam generator tube degradation that exceeds the allowable flaw standards of IWB 3521.

Evaluation procedures and criteria that satisfy the inspection and analytical evaluation requirements of 10CFR50.55a have been developed previously for the degradation mechanisms PWSCC and ODSCC. This document also provides a prescription for implementation of plant degradation specific management programs.

2.2 COMPLIANCE WITH 10CFR50 GENERAL DESIGN CRITERIA

Development of inspection and repair criteria for detection and evaluation of individual degradation mechanisms shall be based on the guidelines specified by the GDC in 10CFR50, in addition to the specific margins used to satisfy Regulatory Guide 1.121. Based on a review of the GDC it was concluded that GDC 2, 4, 14, 15, 30, 31, and 32 are applicable generally for the development of degradation specific inspection and repair criteria for steam generator tubes. These GDC require adequate margin against tube rupture and leakage, and require access to be provided for inspection. The following presents the applicable GDC and provides general guidelines for developing degradation specific inspection and repair criteria to meet these guidelines.

- Criterion 2 - Design bases for protection against natural phenomena. Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

Degradation specific inspection and repair criteria shall include appropriate consideration for protection against natural phenomena. This includes considering the effects of the natural phenomena both individually and in combination with normal or accident conditions. Of particular importance for developing degradation specific inspection and repair criteria are the combined effects of a pipe break accident and a safe shutdown earthquake (SSE). Repair limits for degradation specific mechanisms shall be shown to comply with appropriate structural integrity criteria for such combined effects.

° Criterion 4 - Environmental and dynamic effects design bases. Structures, systems, and components important to safety shall be designed to accomodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

Degradation specific inspection and repair criteria shall include consideration of environmental and dynamic effects. Dynamic effects associated with some postulated pipe ruptures may be excluded from consideration if approved by the NRC in accordance with GDC 4.

° Criterion 14 - Reactor coolant pressure boundary. The reactor coolant boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

R.G. 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes" provides explicit and implied safety margins for tube loading. R.G. 1.121 explicitly states that tube loading should have a safety factor of 3.0 under normal operating conditions. The regulatory guide further states that the margins of safety against tube rupture under postulated accident conditions should be consistent with the margin of safety determined by the stress limits specified in Section III of the ASME Boiler and Pressure Vessel code. Repair limits developed for degradation specific mechanisms shall be shown to meet all of the above acceptance criteria.

Following implementation of a degradation specific tube repair criterion, steam generator tube integrity is maintained both by inspection and by measuring steam generator primary-to-secondary leakage. To further ensure that adequate safety margins are maintained during service, enhanced inservice inspections shall be performed for the affected regions during each refueling outage. Any tubes found to have flaws larger than those necessary to maintain acceptable margins against tube rupture and abnormal leakage (including consideration of additional flaw growth during service) will be repaired.

Degradation specific repair criteria shall be used only for those mechanisms where service experience indicates normal operating leakage levels can be expected to remain at very low levels. A maximum leak rate of 150 gpd per steam generator has been established for normal operation. This leakage level provides added assurance that adequate margins against tube rupture and excessive leakage are maintained.

- Criterion 15 - Reactor coolant system design. The reactor coolant system and associated auxiliary, control, and protection system shall be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Because the steam generator tubing represents a large portion of the total primary system pressure boundary, factors of safety of 3 on normal pressure loads and 1.43 on accident loads shall be used for degradation specific repair criteria to define the maximum degradation allowed to remain in service and to ensure that the steam generator tube integrity is maintained during normal operation including anticipated operational occurrences. In addition, a maximum leak rate of 150 gpd per steam generator at normal operation shall be established.

- Criterion 30 - Quality of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of the reactor coolant leakage.

With implementation of degradation specific management, those tubes with identified degradation will be evaluated to assess compliance with the repair criteria. Tubes that are not in compliance with the repair criteria will be repaired.

During reactor operation, the secondary side of the steam generator will be monitored to detect leaks in steam generator tubes. If leakage exceeding 150 gpd per steam generator is detected during normal operation, the unit will be shut down and the steam generator tubes will be inspected to determine the source of the leakage.

- Criterion 31- Fracture prevention of reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state, and transient stresses and (4) size of flaws.

To ensure that tubes behave in a nonbrittle manner and the probability of rapidly propagating fracture is minimized during operating, maintenance, testing and postulated accident conditions, the tubes are manufactured from ductile materials and conservative margins are applied to normal, maintenance, testing and postulated accident loads. These margins shall be confirmed from experiments with tubes that have representative or simulated degradation for which degradation specific repair criteria are being developed. This testing shall be used as the basis for development of the repair criteria.

The margins shall be determined considering the temperatures and pressures at normal and postulated accident loads, and the uncertainties associated with material properties, stresses, degradation measurement error, and in-service crack growth. Again, as noted above, the margins shall be in compliance with those specified in Regulatory Guide 1.121, the ASME Code, and where the measured values for these variables are unavailable, conservative values shall be used.

- Criterion 32 - Inspection of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leak tight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

Generally, enhanced inspections shall be performed for the tube segments that are susceptible to the degradation specific mechanism for which repair criteria are being developed. Performing these inspections shall provide assurance that the degradation is within limits such that safety margins used in the structural integrity evaluation are maintained during service conditions, and steam generator primary-to-secondary leak rates during normal and postulated accident conditions remain within required limits.

2.3 ENSURING TUBE INTEGRITY

When degradation specific management programs for steam generator tube inspection and evaluation are developed and implemented to comply with 10CFR50.55a and the GDC as described in Sections 2.1 and 2.2, adequate structural and leak rate margins will be maintained and steam generator reliability will be enhanced. The enhanced tube integrity and compliance with regulatory requirements are achieved by employing the following activities when developing and implementing degradation specific management programs.

- The inspection techniques must reliably detect the degradation mechanism at the location where it is likely to occur. Implementation of a degradation specific management program includes the means to sample and confirm the degradation mechanism. This will provide a high probability of degradation detection, characterization, and proper disposition.

- Degradation specific management programs include adoption of industry tube inspection guidelines to assure a consistent manner for obtaining and evaluating inspection results.

- Tube repair criteria will be based on margins specified in R.G. 1.121 and the ASME Code to ensure adequate margins against tube rupture during normal and postulated faulted loads. Tube repair criteria also will consider leak rate to ensure compliance with the GDC. The structural and leakage limits incorporated into the repair criteria will be confirmed by data to ensure the acceptable margins against rupture and abnormal leakage will be maintained during operation.

- Allowance for inspection error will be included in the tube repair limit calculation. This will be developed for each degradation mechanism and associated inspection technique.

- Allowance for degradation growth rates will be included in the tube repair limit calculation. This will be developed individually for each degradation mechanism and associated inspection technique.

Section 3

STEAM GENERATOR TUBE INSPECTIONS

An important element in the implementation of steam generator degradation specific management is an effective NDE inspection program. Consequently, a prerequisite to adoption of the degradation specific management option is for the industry to develop and a utility to implement steam generator tube inspection guidelines. Steam generator tube inspection guidelines are needed to enhance the ability for early detection and characterization of tube degradation. In addition, inclusion of an inspection guideline in the degradation specific management option assures industry consistency in data recording and analysis.

Steam generator tube inspections include the following elements:

- sampling strategies
- data acquisition techniques (for various tube degradation forms and precursors, for example, ODSCC at drilled tube support plate intersections)
- data analysis
- qualification of examination personnel
- performance demonstration.

A committee of industry representatives (the Steam Generator ISI Guidelines Committee) under the auspices of EPRI and the Steam Generator Reliability Project (SGRP) has developed a comprehensive guideline that address the elements listed above (6). The contents of this report are summarized in this section and are referred to in this document as the industry tube inspection guidelines.

Those wishing to implement the degradation specific management option must develop plant specific tube inspection procedures in conformance with the industry tube inspection guidelines (Section 3.1). Development of these plant specific tube inspection procedures will not preclude use of currently licensed depth based repair criteria. On the other hand, development of these procedures is a prerequisite for, and a part of, a degradation specific management program.

In addition to generic tube inspection procedures, degradation specific inspection procedures shall be developed following degradation specific inspection requirements (Section 3.2) and other inspection requirements included in this document (Section 3.3).

3.1 INDUSTRY TUBE INSPECTION GUIDELINES

During the early 1980's, EPRI and the Steam Generator Owners Group informally issued nondestructive evaluation (NDE) guidelines for steam generator tube examinations (the 1981 original and a 1984 revision) for the utility community. The purpose of these guidelines was to provide reliable NDE strategies for the diversity of damage mechanisms that had been experienced. With additional industry experience, increased understanding, and the emergence of new NDE technologies, an update of the tube examination guideline was completed and released in 1988 (6).

The PWR Steam Generator Examination Guidelines provide specific guidelines and recommendations, based on research results and plant experience, that help utilities achieve the maximum benefit from periodic steam generator tube examinations. Implementation of the guideline will reduce the likelihood of forced outages due to tube leaks, assure that established regulatory requirements are met, and minimize unnecessary tube repair.

It was recognized at the time of the development of PWR Steam Generator Examination Guidelines that improved NDE methods would make possible characterization of degradation that might exceed current Technical Specification repair limits but which would not affect steam generator reliability or safety. Consequently, it noted that degradation specific repair criteria should be developed in parallel.

The PWR Steam Generator Tube Examination Guidelines (6) cover recommended practices for steam generator tube nondestructive examinations. The main body of the document addresses steam generator sampling strategies or examination programs, recommended non-destructive examination methods, data analysis and performance demonstrations. The appendices provide much of the background for the recommendations and describe steam generator operating experience, NDE experience with various damage forms and other background information.

The guideline document provides:

- An overview of steam generator NDE objectives.

- General recommendations which are designed to help utilities implement a steam generator examination program.

- A recommended practice for steam generator examination.

- A summary of steam generator operating experience which documents damage mechanisms, vulnerable locations within the steam generator based on past experience, and NDE experience.

- An industry-wide survey of forced outages due to leakage for which specific causes are identified.

Specific recommendations are provided for:

- Random tube sampling strategies for general steam generator surveillance supplemented with additional augmented sampling for units with active damage mechanisms.

- Methods for monitoring tube integrity and damage precursors.

- Performance demonstration protocols for qualifying data acquisition techniques and analysis personnel.

3.2 DEGRADATION SPECIFIC INSPECTION REQUIREMENTS

Degradation specific inspection requirements will be defined in degradation specific inspection and repair criteria reports. As pointed out in Section 4.1, these reports will define methods to reliably detect the degradation specific damage at the locations where it is active. Further, these methods will be demonstrated to be effective in characterizing the extent of degradation for the mechanism and the location at which the mechanism occurs, and include guidance for determining the measurement error. Finally, inspections will be performed with sufficient completeness that the extent of degradation specific damage is bounded and all likely locations of damage which are potentially safety significant are examined.

The inspection requirements of the degradation specific inspection and repair criteria reports shall be addressed and incorporated into the plant specific tube inspection procedures.

3.3 OTHER INSPECTION REQUIREMENTS

Another necessary degradation specific management program element is verification of degradation specific damage. Currently verification has been accomplished by utilizing a combination of inspection methods and pulled tube examinations. Degradation verification is required of the utility who chooses to implement the degradation specific management option. The objective of this element in the inspection process is to show that the degradation found in the field is represented and bounded by the data base used to establish the degradation specific tube repair criteria. Each utility shall include a means of confirming the presence of a particular degradation mechanism as part of the plant specific tube inspection procedures.

At this time, inspection instrumentation calibration requirements are included only in the ODSCC report (5). When degradation specific management is implemented and inspection instrumentation calibration requirements are not specified in the applicable report, appropriate calibration requirements shall be included as part of the plant specific tube inspection procedures (see Section 5.3.2)

Inspection results from successive tube inspections are needed to calculate degradation growth (Section 4.3). Therefore, the requirement to retain and store inspection results shall be included in the plant specific tube inspection procedures (see Section 5.3.2).

Section 4

DEGRADATION SPECIFIC INSPECTION AND REPAIR CRITERIA

The application of a degradation specific management program will require several steps including: (1) identification of a specific degradation mechanism, (2) development of inspection and repair criteria for the identified degradation mechanism, (3) development of a report that describes and presents the technical bases for the degradation specific inspection and repair criteria, and (4) plant specific implementation of the inspection and repair criteria using the guidelines contained in the report.

The remainder of this section summarizes the elements that should be used to develop a degradation specific inspection and repair criteria report. The specific inspection and repair criteria previously developed for primary water stress corrosion cracking (PWSCC) and stress corrosion cracking on the outer tube diameter at tube to tube support plate intersections (ODSCC) are also summarized. Utility actions required to implement degradation specific criteria on a plant specific basis are described in section 5.

4.1 DEGRADATION SPECIFIC INSPECTION AND REPAIR CRITERIA DEVELOPMENT

Degradation specific inspection and repair criteria development shall consist of the following:

- Degradation experience from the U.S. and, where available, international sources to confirm the extent and character of the degradation.

- Inspection methods that reliably detect the degradation specific mechanism at the locations where it is active shall be defined. These methods must be demonstrated to be effective in characterizing the extent of degradation for the mechanism and the location at which the mechanism occurs, and include guidance for determining the measurement error. The inspection shall be performed with sufficient completeness that the extent of degradation specific damage is bounded and all likely locations of damage which are potentially safety significant are examined.

- An NDE tube inspection parameter that is best suited to assess structural and leakage integrity shall be identified and qualified in accordance with industry tube inspection guidelines.

- Data to predict the degradation specific growth rate. While these growth rates may be plant specific, an industry wide data base should be assembled where the mechanism occurs at a number of plants. The industry data can be used for developing trends and estimating growth rates for individual plants until plant specific data becomes available.

- A means to predict burst pressure as a function of the inservice inspection results. Correlations between burst pressure and the appropriate tube inspection parameter should be established using tubes with degradation representative of inservice degradation (which may include tubes removed from service).

- Repair criteria that provide adequate margins against tube rupture during normal operation and accident loading conditions. These margins shall be established by using: (1) a relationship between burst pressure and the NDE measurement parameter, (2) margins defined by the ASME Code and regulatory requirements or guidelines applied to normal loads and the licensing basis (FSAR) faulted load, and (3) adjustments for NDE uncertainty and degradation growth.

- A mechanism specific evaluation to confirm that application of degradation specific management provides an acceptably low probability of tube rupture (i.e., application of degradation specific management does not significantly increase the historic tube rupture frequency).

- An inservice leakage limit of 150 gpd per steam generator will be used with a degradation specific management program to provide added assurance that adequate margins against tube rupture and leakage are maintained. This limit is conservative in comparison to NRC conclusions in NUREG-0844 regarding leakage limits needed to ensure tube integrity.[1]

- Correlations between leak rate and the appropriate NDE measurement parameter will be established using tubes with degradation representative of inservice degradation (including tubes removed from service).

- A deterministic analysis procedure to predict the leakage that may result from degraded tubes during postulated faulted loads at the end of the next operating interval. The predicted leak rate shall be established based on: (1) a relationship between leak rate and the NDE measurement parameter, (2) adjustments for NDE uncertainty and degradation growth, and (3) the licensing basis (FSAR) faulted load.

- Repair criteria that provide adequate margin against excessive leakage from degraded tubes during accident loading conditions. Adequate margin against

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1 "The staff notes, however, that limits higher than 0.35 gpm could be justified if backed up by appropriate test data indicating that such limits adequately ensure the integrity of leaking tubes during normal operating and postulated accident conditions (7)."

excessive leakage shall be achieved by ensuring that the sum of the predicted leak rates from all degraded tubes will be no greater than the leak rate required to maintain, with a 95% probability at a 95% confidence level, 10CFR100 dose limits. If dose limits less than the 10CFR100 criteria are applied, the corresponding probability and confidence level shall also be reduced.

° The degradation specific inspection and repair criteria must explicitly define the actions and requirements necessary for implementation.

° The degradation specific inspection and repair criteria and their bases must be described in a report that can be used by utilities and the NRC for implementation and evaluation of degradation specific programs.

The elements described above have been used to develop the degradation specific inspection and repair criteria described in Section 4.2 (4,5) and will be used to develop additional degradation inspection and repair criteria in the future.

4.2 CURRENTLY AVAILABLE DEGRADATION SPECIFIC INSPECTION AND REPAIR CRITERIA

Two degradation specific reports have been prepared previously by a committee of industry representatives (Committee for Alternative Repair Limits) under the auspices of EPRI and the Steam Generator Reliability Project (4, 5).

The following is a summary of each of these reports. Table 4-1 summarizes the degradation specific inspection and repair criteria and includes a comparison with the current depth based criteria. The information in these reports may be revised from time to time as more information is obtained. The summaries provided in this section are provided for illustration only and will not be revised when the degradation specific inspection and repair criteria described in (4, 5) are revised.

4.2.1 PWSCC

This report provides the technical support for a length-based repair limit for axial PWSCC in roll transition expansion zones of steam generator tubes in U.S. PWR power plants. In this approach, crack length is used as the measure of tube integrity and tube leakage potential, and operation with cracks that may be through-wall is permitted.

This length-based repair criteria approach is based on tube burst, leak rate, PWSCC growth data, and demonstrated crack length measurement capabilities using rotating pancake coil eddy current equipment and procedures. Together, these support an alternative to the eddy current depth-based repair criteria used for most U.S. PWRs, and can be used to support the site-specific submittals to the U.S. NRC for revision of tube repair criteria. The elements of the approach include:

- Application of EZ PWSCC remedies to limit the number of cracks expected in service.

- At each scheduled inspection outage, performance of rotating pancake coil eddy current inspections of 100% of the roll transitions of tube expansion zones in regions of the steam generator (i.e., the hot leg) where the tube expansion zones are susceptible to PWSCC.

- Repair of tubes with axial cracks longer than a conservatively established repair limit which includes the following elements:

  - Use of a tube rupture curve bounded by published tube burst test correlations.
  - Application of USNRC Regulatory Guide 1.121 safety factors.
  - Use of lower bound tube material properties.
  - Correction for tubesheet constraint.
  - Allowance for crack growth between inspections.
  - Allowance of eddy current crack length measurement uncertainty.
  - Restriction on spacing between adjacent axial cracks.
  - A limit on leakage during postulated faulted loads so that leakage during postulated faulted loads will result in off-site doses that are less than the 10CFR100 limits.

A graphical representation of this sequence is presented in Figure 4-1.

Figure 4-2 shows a schematic of a tube and the tubesheet along with important geometric parameters for EZ PWSCC roll transition cracks. The tube mean radius and thickness are designated as R and t, respectively. The axial crack length, a, is the extent of the longest crack in the axial direction. As indicated in figure 4-2, there are three possible locations of the EZ roll transition relative to the top of the tubesheet. In case A, the roll transition is at the top of the tubesheet and is typical of a full-depth roll tube. In case B, the roll transition is below the top of the tubesheet. This is the case of a partial depth roll tube or of a tube which has been "under" rolled. In the third case shown in figure 4-2, case C, the roll transition is above the top of the tubesheet. This would be typical of a tube which had been "over" rolled.

The length-based repair limit has been developed for tubes with full- and partial-depth roll expansion zones. For tubes with full-depth roll expansion zones, crack length is used to determine that adequate leakage limits are maintained at faulted load and there is acceptable margin against tube rupture. For the tubes with partial-depth roll expansion zones, confinement of the cracked section of the tube within the tubesheet precludes tube burst at a pressure less than that of an unflawed tube and the repair limit is based on maintaining acceptable leakage limits at faulted load for the distribution of cracked tubes.

Field experience indicates that circumferential cracks or combinations of circumferential and axial cracks occasionally occur in tube expansion zones. To provide added assurance that combined axial and circumferential cracks will not reduce margin against tube rupture or increase leakage at faulted load beyond allowable limits, an additional repair limit has been defined. This limit requires the spacing between adjacent axial crack indications to be 10mm (0.4 in.) or more.

The criteria apply to axial or inclined cracks where the axial extent of the inclined crack is greater than the circumferential extent. The criteria do not apply to inclined cracks where the circumferential extent is larger than the axial extent, or to NDE indications evaluated to be distinct circumferential cracks or cracks with significant circumferential components, e.g., "L". "T", "U", or "Z" shaped cracks, examples of which are shown in figure 4-3. Tubes with identified distinct circumferential cracks should be repaired.

Application of the repair limits developed in this document requires at each scheduled inspection outage, performance of rotating pancake coil eddy current inspections of 100% of the roll transitions of tube expansion zones susceptible to PWSCC (i.e., the hot leg).

A leak rate analysis should be performed for cracked tubes that do not require repair to determine if leakage at faulted load will result in off-site doses that are less than the 10CFR100 limits. If results from the accident leak analysis indicate an excessive leak rate, tubes should be repaired selectively until the predicted leak rate is less than the accident allowable leak rate.

To provide assurance against abnormal leakage and tube rupture at normal and faulted loads, a leak rate of 150 gpd per steam generator has been established as the allowable primary to secondary leak rate limit during normal operation.

Figure 4-4 presents a graphic overview of the evaluation procedure developed for degradation specific management of PWSCC at the tubesheet expansion zone region.

4.2.2 ODSCC

A repair limit has been developed for steam generator tubes having axial stress corrosion cracks that initiate at the tube outer diameter in the tube support plate (TSP) region. This form of degradation is called outside diameter stress corrosion cracking (ODSCC). The repair limit was developed for PWR power plants in the United States and is applicable to drilled hole, egg crate and broached TSP designs.

In addition to application of remedies to limit axial ODSCC degradation, the elements of this approach include:

° Confirm that axial ODSCC at TSPs is the damage mechanism.

° At each scheduled inspection outage, perform eddy current inspections of 100% of the TSP where ODSCC has been detected.

° Repair of tubes with ODSCC using a conservatively established repair limit which includes the following elements:

  - Use of a conservative correlation between tube rupture and eddy current voltage.
  - Application of USNRC Regulatory Guide 1.121 safety factors.
  - Use of lower bound material properties.
  - Allowances for ODSCC growth between inspections
  - Allowance for eddy current voltage measurement error.
  - A limit on leakage during postulated faulted loads so that leakage during postulated faulted loads will result in off-site doses that are less than the 10CFR100 limits.

ODSCC is illustrated in figure 4-5 for a drilled hole TSP design. ODSCC refers to a range of stress corrosion cracking morphologies which have been observed to occur along the outside diameter of Alloy 600 steam generator tubes within the TSP intersection. The dominant morphology of ODSCC is axial stress corrosion cracks which occur either singularly or in networks of multiple axial cracks. This network morphology has been termed cellular corrosion. Shallow cellular corrosion may contain both axial and circumferential cracks but exhibits a transition to dominantly axial cracking as the cracking progresses in depth. Limited local patches of intergranular attack (IGA) have sometimes been observed as well.

The term ODSCC as used in this document covers the range of degradation morphologies consistent with the above description. Leak rate and burst test data bases fully cover this range of ODSCC degradation.

The repair criteria are applicable for tubes where axial ODSCC is the dominant degradation mechanism. The ODSCC repair limit is applicable to both 3/4-inch and 7/8-inch diameter tubing and degradation lengths of up to approximately 3/4 inch. The repair limit described in this document is not applicable to tubes having NDE indications evaluated to be distinct circumferential cracks.

Distinguishing between different degradation mechanisms at tube support plates can be difficult. Consequently, the user who chooses to apply the repair limits must demonstrate that axial ODSCC is the dominant degradation mechanism at the location for which the criteria are to be applied.

Eddy-current voltage is used to define the repair limit for axial ODSCC in the TSP region. This approach employs laboratory and field degraded tubes to correlate bobbin coil eddy current voltage with leak rate and burst pressure. The correlations are developed from and are applicable to tubes with axial ODSCC having depths up to 100% wall thickness.

When axial ODSCC is located entirely within the TSP for drilled hole designs, the TSP limits tube radial deformation and tube burst is precluded. However, analyses indicate that a TSP may, in certain instances, move during postulated faulted loads and reduce the constraint effects. Consequently, the repair criteria for the drilled hole TSP design and egg crate and broached designs are based conservatively on burst of an uncovered tube span, and include margins against burst that are consistent with U.S. regulatory guidelines for normal and postulated faulted loads.

Less restrictive repair limits can be used provided it can be demonstrated that the ODSCC will remain covered during faulted loads. Demonstration that ODSCC will remain confined within a drilled hole TSP should incorporate detailed analyses of TSP motion during postulated faulted loads; the analysis should include phenomena that may limit TSP motion (e.g., denting). If it can be shown that ODSCC will be covered by the TSP under all loading conditions, it is likely that tube rupture need not be considered and repair limits can be based on leakage limits for faulted loads.

Field application of this approach includes inspection of the TSP region by bobbin coil for 100% of the tubes in regions of the steam generator where ODSCC has been detected. These affected regions include the hot leg, and the cold leg down to the lowest TSP where ODSCC has been diagnosed. Those tubes where the bobbin coil voltage is greater than that established for ensuring adequate margin against rupture will be repaired.

The repair limit includes adjustments for inspection error and ODSCC growth to ensure with a high level of confidence that adequate margins against burst and unacceptable leakage are maintained throughout the subsequent operating interval. To provide additional assurance against abnormal leakage and tube rupture at normal and faulted loads, a leak rate of 150 gpd per steam generator has been established as the allowable primary to secondary leak rate limit during normal operation.

The inspection results from the tubes not requiring repair will be used to determine the eddy current voltage distribution in the steam generator, and a leak rate analysis will be performed for the accident loads to determine if the leakage will result in off-site doses that are less than the 10CFR100 limits. If results from the accident leak analysis indicate an excessive leak rate, tubes will be selectively repaired.

In some instances, inspections using rotating pancake coil eddy current technology may be used either to confirm that detected degradation is axial ODSCC or to establish a basis for leak rate predictions.

Figure 4-6 presents a graphic overview of the evaluation procedure developed for degradation specific management of ODSCC.

Table 4-1. Degradation Specific Inspection Leakage & Repair Conditions

Degradation Type	Degradation Parameter, Inspection Sample Size	T.S. Operational Leak Rate Limit	Plant Specific Defective Tube Condition
1. Generic	Degradation depth determined by Bobbin Coil (BC): Sample size (*Tech Spec* Table 4 4-2).	500 gpd per steam generator	Indicated depth greater than [ ]% of wall thickness
2. Axial PWSCC in the expansion zone rolll transition (a)	Crack length determined by Rotating Pancake Coil (RPC)	150 gpd per steam generator	
2a Full depth roll	Sample size - 100% of tubes in the affected region		Tubes with axial crack length >[ ] inch Tubes with circumferfential cracks Tubes with axial crack length > [ ] inch & circumferential spacing between axial cracks < 10 mm Tubes that do not allow leak rates to be maintained below [ ] gpm during postulated accidents.
2b Partial depth roll			Tubes with circumferential cracks, Tubes where the sum of the circumferential distances between axial cracks less than 10mm apart is >[ ] inch. Tubes that do not allow leak rates to be maintained below [ ] gpm during postulated accidents
3. Axial ODSCC at tube to tube support plate intersections (b)	BC voltage; Sample size = 100% of tubes in the affected region	150 gpd per steam generator	Tubes with BC voltage >[ ] volts, Tubes with BC voltage <[ ] volts that do not allow leak rates to be maintained below [ ] gpm during postulated accidents

(a) NP-6864-L, Rev. 1, "PWR Steam Generator Tube Repair Limits: Technical Support Document for Expansion Zone PWSCC in Roll Transitions, Revision 1," December 1991.

(b) TR-100407, "PWR Steam Generator Tube Repair Limits: Technical Support Document for Outside Diameter Stress Corrosion Cracking at Tube Support Plates," March 1992.

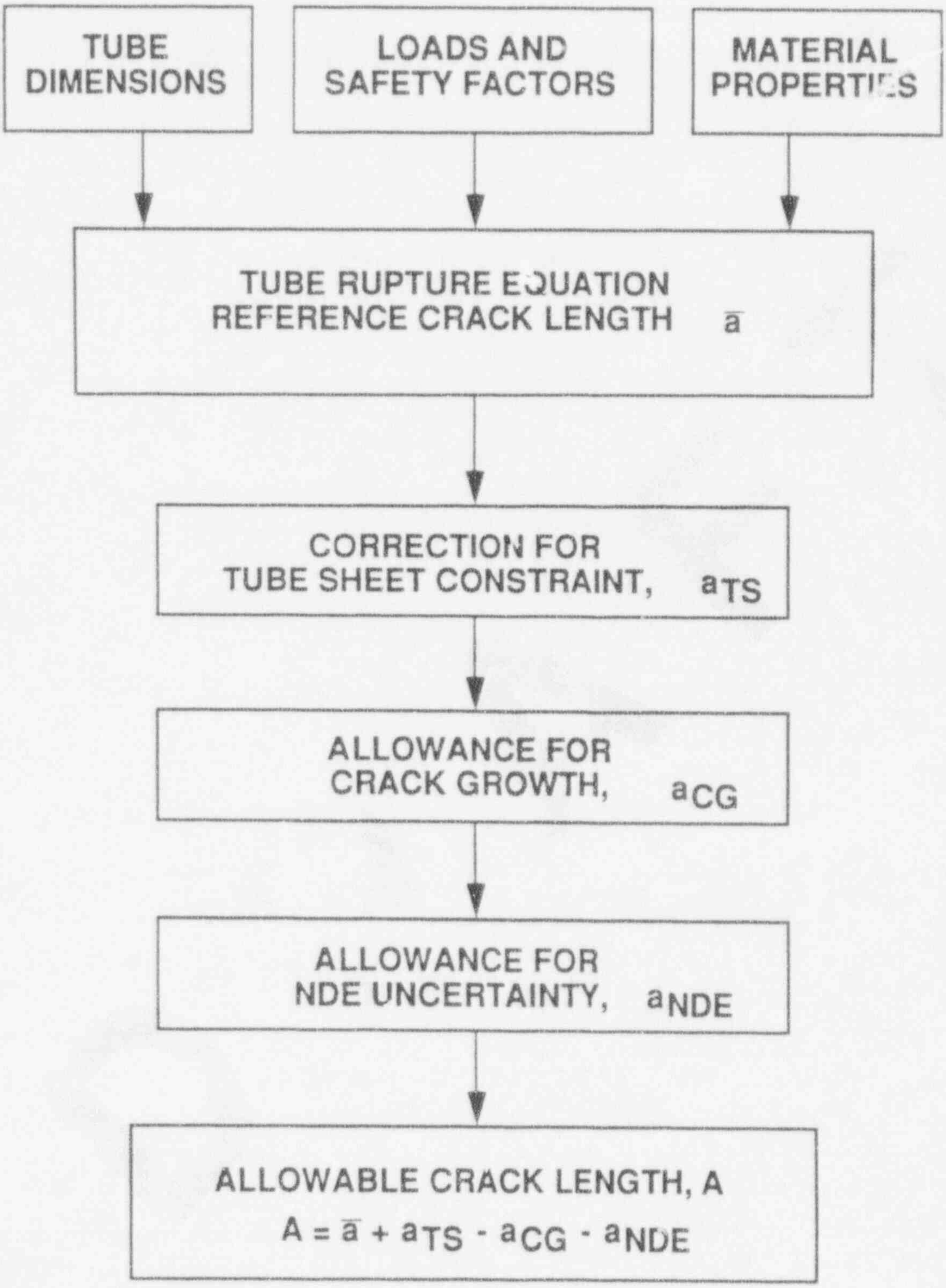
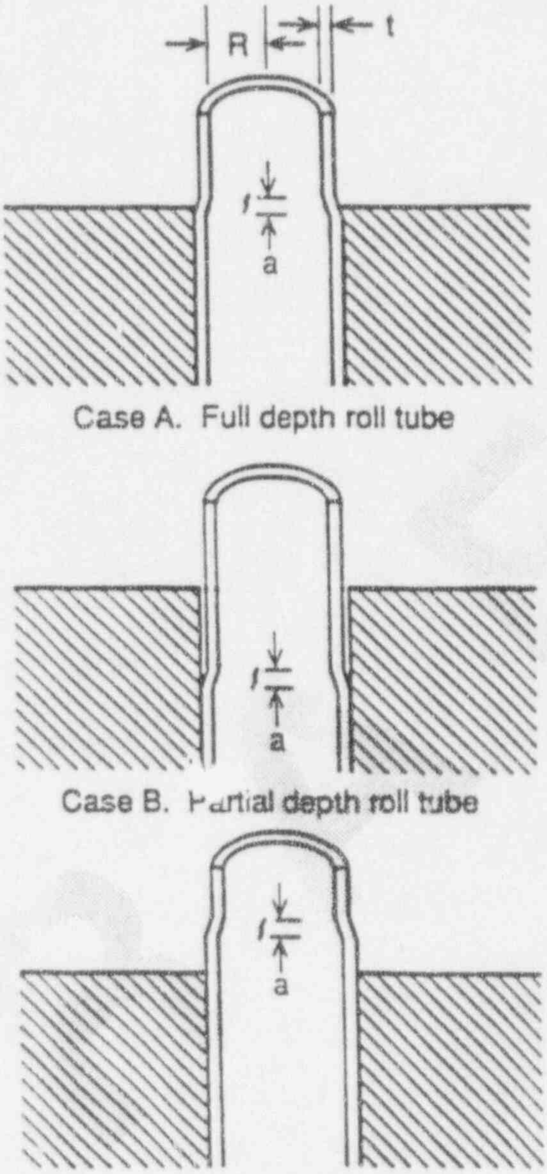
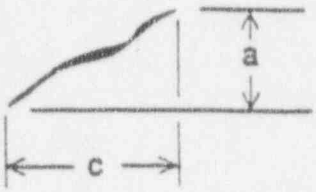


Figure 4-1. Allowable EZ PWSCC Axial Crack Length Calculation Steps



a - axial length of EZ PWSCC crack
R - tube mean radius
t - tube thickness

Figure 4-2. Schematic of EZ Roll Transition Axial Cracks



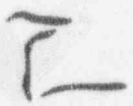
Circumferential and predominately circumferential cracks (c>a)

Axial cracks with distinct circumferential crack component

"L" Shaped Cracks

"T" Shaped Cracks

"U" Shaped Cracks



"Z" Shaped Cracks

Figure 4-3. Examples of EZ-PWSCC crack configurations with distinct circumferential crack components for which this document does not apply and therefore should be repaired.

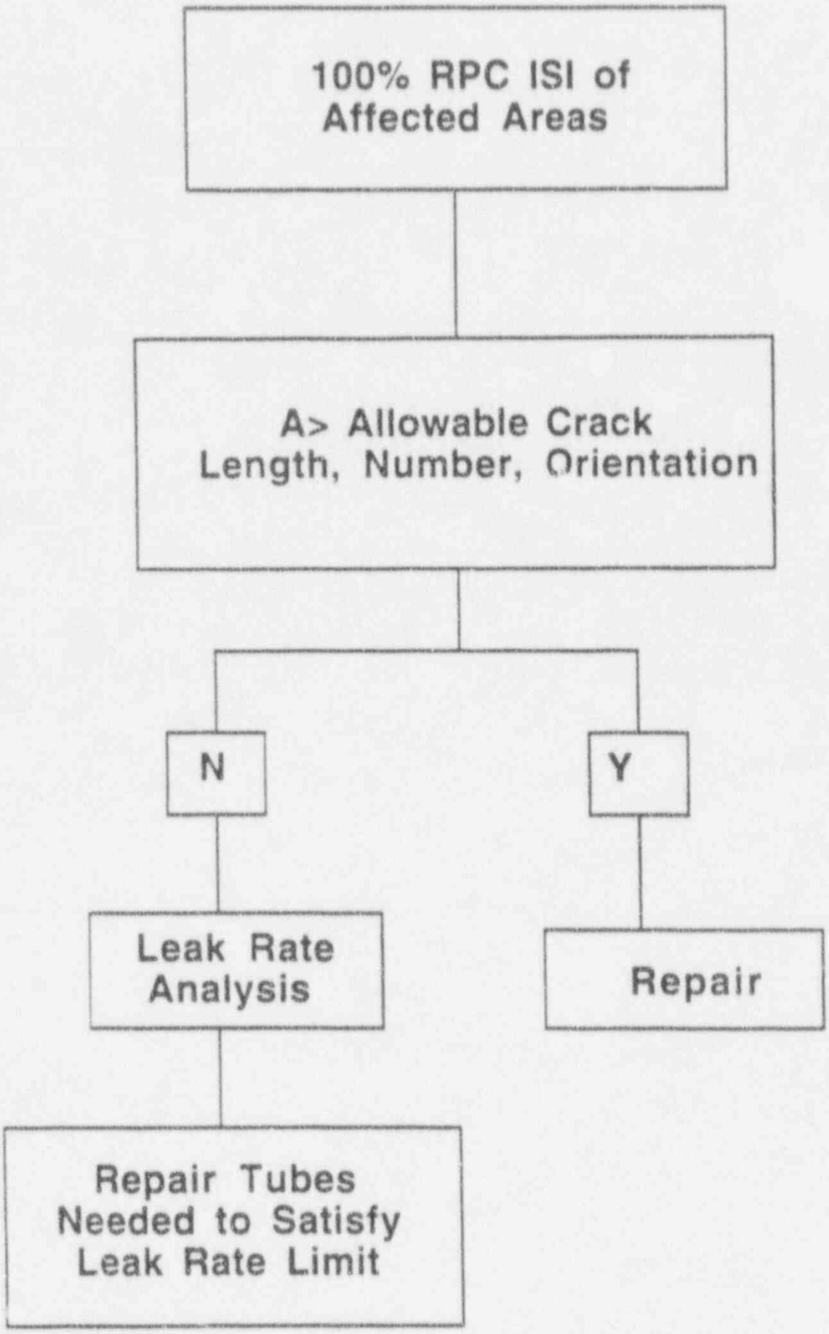


Figure 4-4. Evaluation Procedure for Implementation of an Alternative Repair Criterion for Tubes with EZ PWSCC

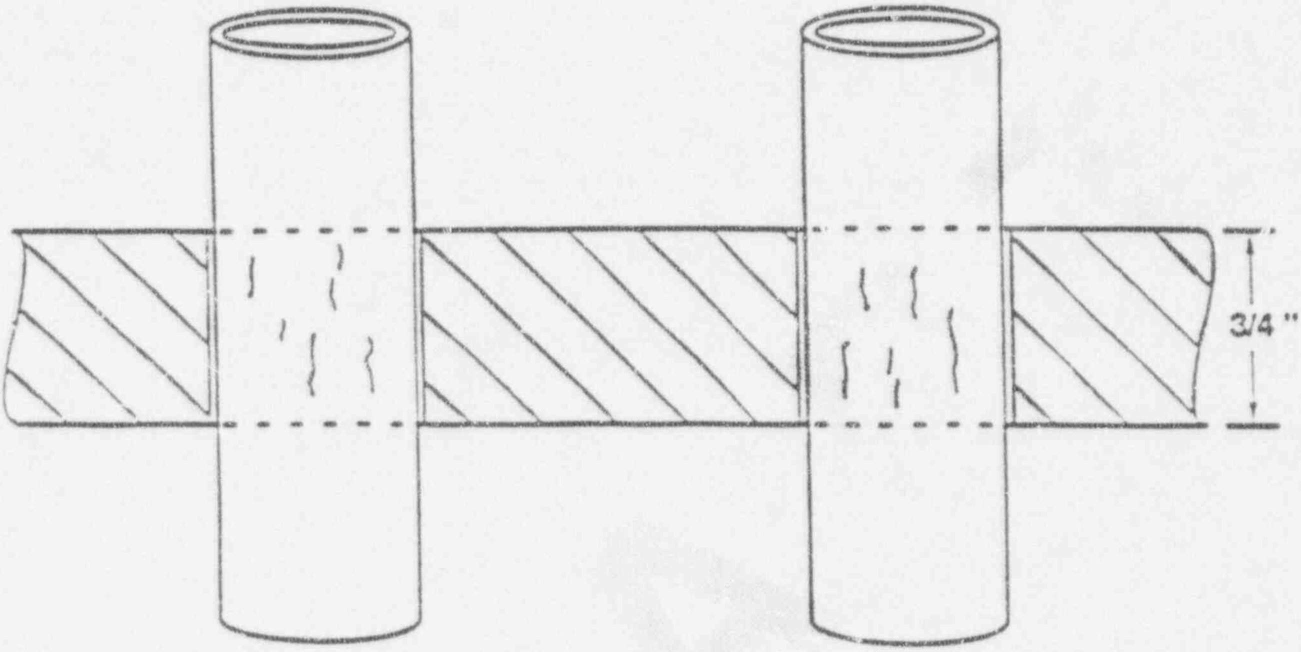


Figure 4-5. Illustration of ODSCC in steam generator tubes in the TSP region.

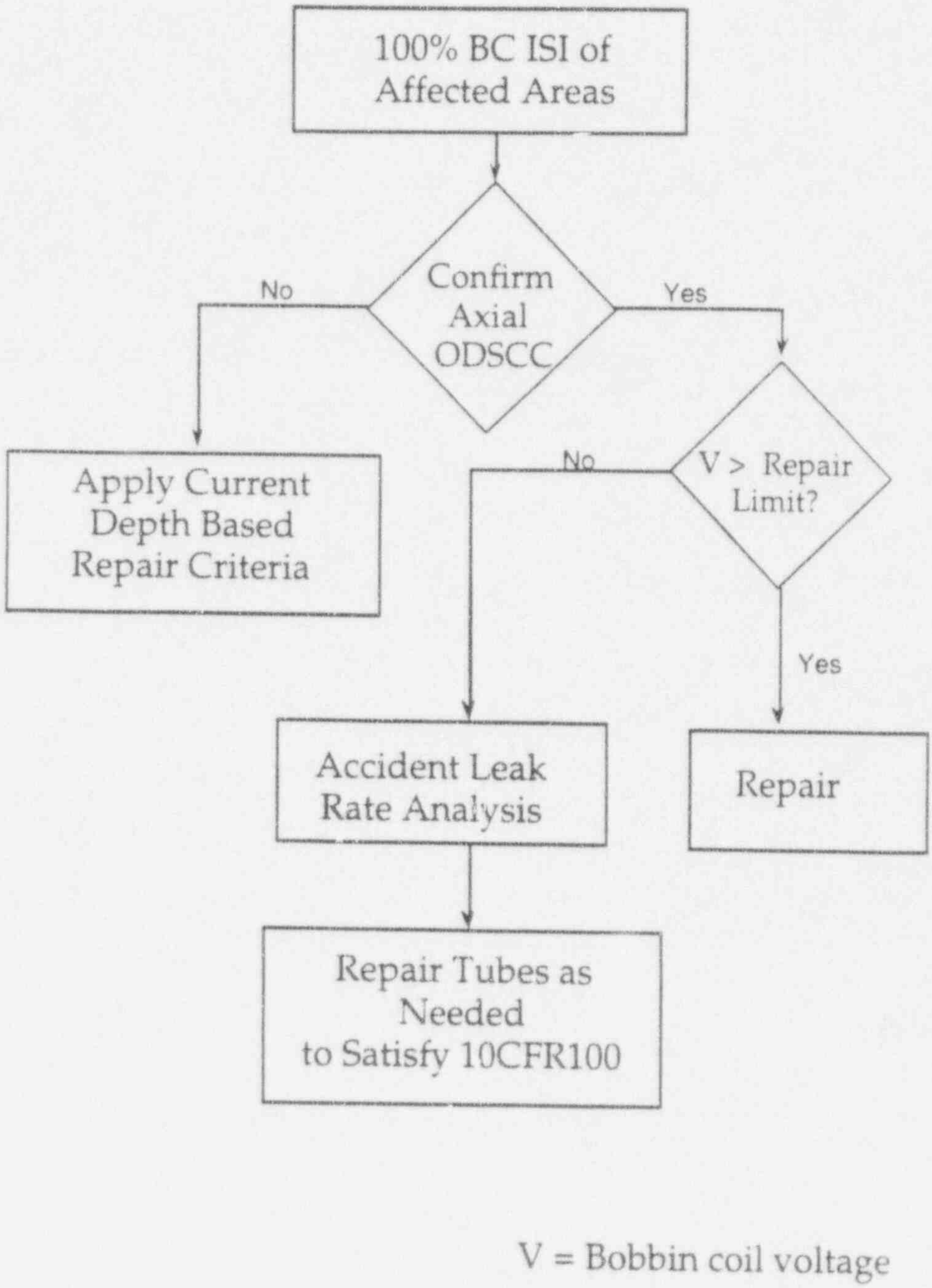


Figure 4-6. Evaluation Procedure for Implementation of an Alternative Repair Criterion for Tubes with ODSCC at TSPs.

Section 5

PLANT SPECIFIC IMPLEMENTATION

As pointed out in Section 1.3, a utility has the options of continuing with current Technical Specification required steam generator tube inspections and depth based repair criteria, augmenting current Technical Specification requirements with improved steam generator tube inspection methods or developing the option of having available, should it be needed, degradation specific management.

The option of continuing with current technical specification requirements (Option 1) does not require any additional plant specific actions to be implemented.

The option of augmenting current technical specification requirements by implementing improved steam generator tube inspection methods (Option 2) requires that a utility develop plant specific inspection procedures in conformance with the PWR Steam Generator Examination Guideline document (described in Section 3). In addition, plant Technical Specifications must be changed to allow for inspection sampling strategies defined in the guideline.

The degradation specific management option (Option 3) involves development of a plant degradation specific management program. The program will include plant specific tube inspection procedures in conformance with tube inspection guidelines and degradation specific inspection and repair criteria. The program will also include plant specific calculations.

This section provides plant specific implementation requirements for the degradation specific management option.

5.1 MODIFICATION OF TECHNICAL SPECIFICATIONS

Steam generator tube inspection, acceptance criteria and reporting requirements are currently specified in plant Technical Specifications. Typically these requirements are in section 3/4.4.5, "Steam Generators" of most Technical Specifications currently in use. In addition, a limitation on primary-to-secondary leakage through steam generators is also specified in plant Technical Specifications, generally in section 3/4.4.6, "Reactor Coolant System Leakage."

Current plant Technical Specifications are specific as to how tubes are to be inspected and evaluated based on indicated degradation depth. To utilize the degradation specific management option, Technical Specifications must be changed to include this option. Specifically, it is recommended that a paragraph be added to what is generally section 3/4.4.5.2, "Steam Generator Tube Sample Selection and Inspection," which reads as follows:

> When specific types of steam generator tube wall degradation have been diagnosed and verified, a degradation specific management program plan may be implemented. Degradation specific management includes limits of applicability, definition of affected tube segments, inspection methods and scope, inspection results classification, tube repair criteria, and augmented leak rate calculations. For those tube segments where degradation specific management has been applied, the requirements of the EPRI document "Steam Generator Degradation Specific Management" shall be used.

When modifying plant Technical Specifications to make degradation specific management an option, the operational primary-to-secondary leak rate per steam generator shall be limited to 150 gpd. Consequently, plant Technical Specifications shall be changed to specify that primary-to-secondary leakage through any one generator be limited to 150 gpd (if this limit is not already so specified in plant Technical Specifications).

5.2 PLANT SPECIFIC PROGRAM PLAN

When a utility implements the degradation specific management option, a plant degradation specific management program plan shall be prepared.

The plant specific program plan identifies the industry tube inspection guidelines to be utilized and the degradation specific inspection and repair reports to be implemented. The plan shall include a commitment to develop plant specific inspection procedures in conformance with these documents. The plan will also include the plant specific structural limit and the allowable accident leak rate limit. The plan shall define plant specific calculations to be performed and reports to be provided to the NRC following each inspection.

5.3 TUBE INSPECTIONS

Implementation of degradation specific management requires both generic and degradation specific tube inspections. These are described in Sections 5.3.1 and 5.3.2.

5.3.1 Generic Requirements

Degradation specific management as defined in this document includes development of plant specific tube inspection procedures in conformance with the PWR Steam Generator Examination Guidelines report (6) described in Section 3. These procedures will have different inspection sample size and expansion requirements than what is now required when a depth based repair criterion is utilized. In addition, the plant specific tube inspection procedures will have additional requirements such as qualification of examination personnel and performance demonstrations which are not now specified in plant Technical Specifications.

5.3.2 Degradation Specific Requirements

In addition to implementation of the industry tube inspection guidelines, when the degradation specific management option is to be followed, degradation specific tube examinations must also be performed. These examinations have two objectives. The first objective is to verify the presence of a specific degradation mechanism. The second objective is to characterize the degree of degradation.

As pointed out in Section 3, there are currently no generic guidelines for verifying the presence of a specific degradation mechanism. To date, this has been accomplished by utilizing various inspection methods in combination with limited pulled tube examinations and tests. As experience is gained in the form of correlations between inspections and pulled tube results, it should no longer be necessary to pull tubes to confirm damage mechanisms but verification can be expected to be based on inspection results alone.

Degradation specific tube examinations to characterize the extent of damage are specified in the degradation specific inspection and repair criteria reports. The scope, methods and interpretations of these examinations shall follow the applicable report. When degradation specific characterization is required, the methods and procedures for this should be incorporated into plant specific tube inspection procedures.

If degradation specific inspection instrumentation calibration requirements are not specified in the applicable degradation specific tube inspection and repair criteria report, appropriate calibration requirements shall be developed and included as part of the plant specific tube inspection procedures.

In addition, the requirement to retain and store inspection results for identified degradation less than the repair limit shall be included in the plant specific tube inspection procedures.

5.4 PLANT SPECIFIC CALCULATIONS

Degradation specific inspection and repair criteria reports require certain plant specific calculations to be made. These calculations are needed to establish the plant specific tube repair limit and demonstrate that the potential leak rate expected during postulated faulted loads from degraded tubes that remain in service will result in off-site doses that are less than the 10CFR100 limits. Specific guidance for both these calculations is given in the degradation specific inspection and repair criteria reports.

Calculation of the tube repair limit is based on the inputs which, in many cases, are plant specific. These inputs typically include:

- Tube loads during normal operation and under accident conditions. In most cases the dominant and only significant tube load will be primary-to-secondary differential pressure with the largest load calculated for a main steam line break accident. When defining tube loads, all load combinations (e.g., LOCA + SSE) which could influence the integrity of the specific form of degradation shall be considered.

- Tube material properties. These properties may be established from construction records with appropriate correction for temperature or, if construction information is not available, generic lower bound properties may be available in the degradation specific inspection and repair criteria reports.

- NDE measurement error when characterizing tube degradation. This error is dependent on the degradation mechanism and inspection technique.

- Anticipated degradation growth between inspections. This may be based on past plant specific inspection results if available, or based on industry experience included in degradation specific inspection and repair criteria reports. The calculation of degradation growth shall be updated after each inspection.

The methods and data correlations needed for calculating the tube repair limits along with the appropriate safety factors to be applied to tube load are specified in degradation specific inspection and repair criteria reports.

Calculation of the potential leak rate expected during postulated faulted loads from tubes that remain in service is also based on plant specific inputs. These inputs typically include:

- The number of tubes with identified degradation above a threshold value along with the degree of degradation for these tubes. The threshold and degree of degradation are in terms of the NDE degradation characterization parameter.

- Tube primary-to-secondary differential pressure under accident conditions.

The methods and data correlations needed for calculating the potential leak rate expected during postulated faulted loads from tubes that remain in service are specified in the degradation specific inspection and repair criteria reports. Where probabilistic and deterministic methods of leak rate analysis are provided as options, deterministic methods shall be used. When degradation specific management is applied for more than one degradation specific mechanism, the combined leakage from all tube segments covered by degradation specific management shall be calculated. If the calculated leakage is not less than the allowable accident leak rate, repair of additional tubes is necessary.

5.5 INDUSTRY-WIDE DEGRADATION SPECIFIC MANAGEMENT DATA BASE

As part of plant implementation of degradation specific management, a utility will commit to participate in an on-going industry-wide degradation specific management data base. This industry-wide effort will consist of those activities necessary for uniform and consistent data support for both generic and plant-specific activities. This will help ensure coordination and cognizance of all appropriate data sources for review, analysis and evaluation in relation to on-going degradation specific management activities.

Participation in this industry-wide effort will include a commitment by each utility to provide appropriate plant data to a centralized organization. The specific data to be provided will be determined by the industry and will be dependent on the specific degradation mechanism, the existing data and other relevant factors. Typical data expected to be included is that from both domestic and foreign sources of inservice inspections, pulled tube examinations and laboratory specimens. Specific data to be included will address items such as degradation growth, NDE uncertainty, tube burst, tube leakage, morphology, etc., as determined by the industry to be necessary for each degradation mechanism.

The centralized data base will provide a variety of functions and resources for use in utility implementation activities including: identifying generic and plant-specific data required; obtaining, analyzing and incorporating data into correlations; and disseminating data and correlations to industry participants for use in updating reports and other documents.

5.6 TUBE REPAIR AND REPORTING

Steam generator tubes must be repaired if the NDE measured level of degradation exceeds the degradation specific repair limit. Further, if the calculated leakage for all degraded tubes left in service is not less than the allowable accident leak rate, additional tubes must be repaired. Degraded tubes which remain in service must be examined at each inservice inspection period utilizing examination methods identified as appropriate for the degradation specific mechanism. At that time, plant specific calculations should be reviewed and, if necessary, a new plant specific repair limit established. Tube repair decisions and leakage calculations are then repeated utilizing the new repair limit.

A plant specific inspection report documenting the calculated plant specific repair limit, examination results, repaired tubes, tubes with indicated degradation to be left in service and calculated leakage under accident conditions should be prepared by the utility. This report is required after each inservice steam generator inspection. The repair limit and the number of tubes plugged in each steam generator should be reported to the NRC within 15 days of the inspection. The complete report should be available within 12 months of the inservice inspection.

Section 6

REFERENCES

1. Flesch, B., and Cochet, B., "Leak-Before-Break in Steam Generator Tubes," International Journal of Pressure Vessels & Piping, 43 (1990) pp. 165-179.

2. Esteban, A., Bolaños, M. G., and Figueras, J. M., "A Plugging Criterion for Steam Generator Tubes based on Leak-Before-Break," International Journal of Pressure Vessels & Piping, 43 (1990) pp. 185-186.

3. Roussel, G., and Mignot, P., "Behavior of Steam Generator Tubes in Belgium and its Consequences," Nuclear Engineering and Design, 133 (1992) pp. 63-39.

4. PWR Steam Generator Tube Repair Limits: Technical Support Document for Expansion Zone PWSCC in Roll Transitions (Revision 2), EPRI Draft Report, EPRI NP-6864-L, Revision 2, August 1993.

5. PWR Steam Generator Tube Repair Limits: Technical Support Document for Outside Diameter Stress Corrosion Cracking at Tube Support Plates, Revision 1, EPRI Draft Report, EPRI TR-100407, Revision 1, August 1993.

6. PWR Steam Generator Examination Guidelines: Revision 3, EPRI Final Report, EPRI NP-6201, Revision 3, November 1992.

7. NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity, NRC Final Report, NUREG-0844, September 1988.

APPENDIX A

SAMPLE LICENSE AMENDMENT SUBMITTAL

for

STEAM GENERATOR DEGRADATION SPECIFIC MANAGEMENT

TO: USNRC
etc.

SUBJECT: Proposed Revision To (Plant) Technical Specifications For Steam Generator Degradation Specific Management

This letter and its attachments constitutes a request by (Utility) for an application for an amendment to the Technical Specifications for (Plant) to implement Steam Generator Degradation Specific Management.

This amendment request is comprised of the following items:

Attachment 1 — Revised Technical Specifications.
This attachment describes the proposed revisions to the (Plant) Technical Specifications needed to incorporate Steam Generator Degradation Specific Management.

Attachment 2 — No Significant Hazards Consideration Evaluation.
This attachment describes the evaluation of the proposed change and the bases for concluding that the change involves no significant hazards consideration.

The overall approach of Steam Generator Degradation Specific Management is described in detail in the EPRI document "Steam Generator Degradation Specific Management" and includes multiple elements to maintain plant safety and operation. These multiple elements are:

- tube repair limits which maintain required margins against structural tube failure
- improved inservice inspection procedures and data analysis methods
- enhanced nondestructive examination at each outage to provide a high level of confidence that tubes exceeding the allowable limit are repaired or removed from service
- reduced allowable leak rate during normal operation
- maintaining site boundary doses within existing limits during postulated accidents.

Implementation of these elements constitutes a defense-in-depth approach that was developed to ensure adequate levels of safety. The structural margins, the tube repair limits, and the leak rate limits help ensure that margins against tube rupture and radiological doses are maintained within required limits.

The benefits of this change are: reduced occupational exposure due to less steam generator tube repair, minimizing further reductions in thermal margins due to repair of tubes, minimizing reductions in plant efficiency due to tube repair, reduced cost and duration of plant outages, and maintaining flexibility for long term repair options.

The requirements for degradation specific management have been prepared by a committee of U. S. and international industry participants who are experts on technical and licensing issues associated with development and implementation of a steam generator tube inspection and repair program.

The ACRS has previously reviewed the concept of the industry approach on which this change is based and they have encouraged the timely establishment of such a revised approach.[1]

Prompt NRC review and approval of this proposed change is requested.

(Utility)

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1 Letter from David A. Ward, Chairman of ACRS, to Ivan Selin, Chairman of USNRC, "Steam Generator Tube Repair Limits," dated November 15, 1991.

ATTACHMENT 1

PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS

FOR IMPLEMENTATION OF

STEAM GENERATOR DEGRADATION SPECIFIC MANAGEMENT

Proposed Changes to Technical Specifications
for Implementation of
Steam Generator Degradation Specific Management

A written description of the proposed generic changes to the Tech Specs is provided below. Following this written description is a mark-up of a sample Tech Spec to more graphically show the suggested revisions and deletions.

REACTOR COOLANT SYSTEM

3/4.4.5 STEAM GENERATORS

SURVEILLANCE REQUIREMENTS

4.4.5.2 Steam Generator Tube Sample Selection and Inspection

Create two sections: 4.4.5.2.1 and 4.4.5.2.2 as described below.

4.4.5.2.1 Same as original 4.4.5.2

Add a new section 4.4.5.2.2:

4.4.5.2.2 When specific types of steam generator tube wall degradation have been diagnosed and verified, a degradation specific management program plan may be implemented. Degradation specific management includes limits of applicability, definition of affected tube segments, inspection methods and scope, inspection results classification, tube repair criteria, and augmented leak rate calculations. For those tube segments where degradation specific management has been applied, the requirements of the EPRI document "Steam Generator Degradation Specific Management" shall be used.

4.4.5.5 Reports

Add the following sentence to the end of item a:

If degradation specific management has been applied, this report shall also identify the plant specific repair limits.

Add the following sentence to the end of item b:

If degradation specific management has been applied, this report shall also include the plant specific repair limits, examination results, repaired tubes, tubes with indicated degradation left in service, and calculated leakage under accident conditions.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION OF OPERATION

3.4.6.2 Revise leak rates of item c. to:

c. 150x (where x = number of steam generators) gallons per day primary-to-secondary leakage through all steam generators and 150 gallons per day through any one steam generator.

REACTOR COOLANT SYSTEM

BASES

3/4.4.5 STEAM GENERATORS

In second paragraph, revise leak rate to 150 gallons per day per steam generator.

Delete the third paragraph in this section and replace it with:

Steam generator degradation specific management allows steam generator tube primary-to-secondary leakage to be determined by the degradation specific steam generator tube wall degradation type. The most limiting allowable primary-to-secondary leakage is identified in 3.4.6.2. Primary-to-secondary leakage during certain accident conditions will be limited such that the associated radiological consequence as a result of this leakage is less than the 10CFR100 limits.

Tube repair limits are defined to maintain adequate margins against tube rupture and excessive leakage. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation significantly less than that associated with the repair limits. If a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Repair will be required for all tubes with imperfections exceeding the repair limits.

REACTOR COOLANT SYSTEM

BASES

3/4.4.6.2 OPERATIONAL LEAKAGE

Replace the third paragraph with:

Maintaining an operating leakage limit of 150 gpd per steam generator will minimize the potential for unacceptable leakage during certain accident conditions. Steam generator degradation specific management allows steam generator tube primary-to-secondary leakage to be determined by the degradation specific steam generator tube wall degradation type. Primary-to-secondary leakage during certain accident conditions will be limited such that the associated radiological consequence as a result of this leakage is less than the 10CFR100 limits.

MARK-UP SHOWING

SAMPLE, GENERIC TECH SPEC CHANGES

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} (TI-0413A&B, TI-0433A&B, TI-0423A&B, TI-0443A&B) above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - 4.4.5.2.1 The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

1) All nonplugged tubes that previously had detectable wall penetrations greater than 20%,

2) Tubes in those areas where experience has indicated potential problems, and

3) A tube inspection (pursuant to Specification 4.4.5.4a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

2) The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

ADD PARAGRAPH 4.4.5.2.2 FROM NEXT PAGE HERE

4.4.5.2.2 When specific types of steam generator tube wall degradation have been diagnosed and verified, a degradation specific management program plan may be implemented. Degradation specific management includes limits of applicability, definition of affected tube segments, inspection methods and scope, inspection results classification, tube repair criteria, and augmented leak rate calculations. For those tube segments where degradation specific management has been applied, the requirements of the EPRI document "Steam Generator Degradation Specific Management" shall be used.

9) Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.8.2;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.8.2 within 12 months following the completion of the inspection. This Special Report shall include:

1) Number and extent of tubes inspected,

2) Location and percent of wall-thickness penetration for each indication of an imperfection, and

3) Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported in a special report to the Commission pursuant to Specification 6.8.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

If degradation specific management has been applied, this report shall also identify the plant specific repair limits.

If degradation specific management has been applied, this report shall also include the plant specific repair limits, examination results, repaired tubes, tubes with indicated degradation left in service, and calculated leakage under accident conditions.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

150X (WHERE X = NUMBER OF STEAM GENERATORS) GALLONS PER DAY

c. ~~1 GPM~~ primary-to-secondary leakage through all steam generators and ~~500~~ 150 gallons per day through any one steam generator, *

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System.

e. 64 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig.

f. 0.5 GPM leakage per nominal inch of valve size up to a maximum of 5 GPM at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve. The maximum allowable leakage of any Reactor Coolant System Pressure Isolation Valve is specified in Table 3.4-1.*

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the limit specified in Table 3.4-1, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*Test pressures less than 2235 psig but greater than 350 psig are allowed. Observed leakage shall be adjusted for the actual test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.

3/4.4.4 RELIEF VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer Code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System (primary-to-secondary leakage = ~~500~~ gallons per day per steam generator). ||*
Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of ~~500~~ gallons per day per ||*
steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

150

~~Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.~~

REPLACE WITH PARAGRAPHS ON NEXT PAGE

Steam generator degradation specific management allows steam generator tube primary-to-secondary leakage to be determined by the degradation specific steam generator tube wall degradation type. The most limiting allowable primary-to-secondary leakage is identified in 3.4.6.2. Primary-to-secondary leakage during certain accident conditions will be limited such that the associated radiological consequence as a result of this leakage is less than the 10CFR100 limits.

Tube repair limits are defined to maintain adequate margins against tube rupture and excessive leakage. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation significantly less than that associated with the repair limits. If a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Repair will be required for all tubes with imperfections exceeding the repair limits.

STEAM GENERATORS (Continued)

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission in a Special Report pursuant to Specification 6.8.2 within 30 days and prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These Detection Systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

~~The total steam generator tube leakage limit of 1 gpm for all steam generators ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.~~

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the Leakage Detection Systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 64 gpm at a nominal RCS

REPLACE WITH PARAGRAPH ON NEXT PAGE

Maintaining an operating leakage limit of 150 gpd per steam generator will minimize the potential for unacceptable leakage during certain accident conditions. Steam generator degradation specific management allows steam generator tube primary-to-secondary leakage to be determined by the degradation specific steam generator tube wall degradation type. Primary-to-secondary leakage during certain accident conditions will be limited such that the associated radiological consequence as a result of this leakage is less than the 10CFR100 limits.

ATTACHMENT 2

NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

The proposed change to the (Plant) Technical Specifications has been reviewed and evaluated in detail. (Utility) has determined that operation of the (Plant) in accordance with the proposed Technical Specifications does not represent a significant hazards consideration according to the criteria set forth in 10CFR50.92(c). This no significant hazards consideration conclusion is based on the program described in the EPRI document "Steam Generator Degradation Specific Management" and the evaluation described below which addresses each of the three criteria in 10CFR50.92(c).

(1) Does not involve a significant increase in the probability or consequences of an accident previously evaluated

Of the various accidents previously evaluated, the proposed change affects only the steam generator tube rupture (SGTR) event evaluation and the postulated steam line break (SLB) accident evaluation.

a. Probability

Regarding the SGTR event, the required structural margins of the steam generator tubes will be maintained as described in the EPRI document "Steam Generator Degradation Specific Management." Regulatory Guide 1.121 margins against tube burst are maintained for both normal and postulated accident conditions. Thus, the tubes permitted to remain in service would be structurally sound and meet required criteria. Furthermore, nondestructive examinations will be performed at each outage to provide a high level of confidence that tubes exceeding the allowable limit are repaired or removed from service. In addition, the reduced allowable leak rate during normal operation will enhance the likelihood of leak detection before tube rupture and minimize leakage during both normal and accident conditions. Therefore, there is no significant increase in the probability of occurrence of a SGTR.

The proposed change can increase the probability of occurrence of the SLB accident only if the probability of occurrence of a SGTR is increased. Since the proposed change results in no significant increase in the probability of occurrence of a SGTR, there is no significant increase in the probability of occurrence of the SLB accident.

The proposed change does not affect other systems, components or operational features. The foregoing factors result in the conclusion that the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

b. Consequences

The consequences of a SGTR event are affected by the primary-to-secondary leakage flow during the event. This leakage flow through a postulated broken tube is not affected by the proposed change since the differential pressure causing the leakage flow to occur is not changed. Therefore, the consequences of the SGTR event are not affected by the proposed change.

The consequences of the postulated SLB accident also relate to the primary-to-secondary leakage postulated to occur during the accident. This leakage is postulated to occur through leaking tubes. Enhanced nondestructive examination will help ensure that only those steam generator tubes meeting the required structural margins will remain in service. The reduced allowable leakage during normal operation provides added assurance that adequate margins against tube rupture and excessive leakage are maintained. Furthermore, the EPRI document "Steam Generator Degradation Specific Management" requires analyses confirming that the leak rate during the postulated SLB will result in off site doses that are less than the 10CFR100 limits.

The proposed change does not affect other systems, equipment or failure modes previously evaluated.

Therefore, based on the above evaluation, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

(2) Does not create the possibility of a new or different kind of accident from any accident previously evaluated

Implementation of the proposed change does not introduce any changes or mechanisms that create the possibility of a new or different kind of accident.

The proposed change will maintain the required structural margin of the steam generator tubes for both normal and accident conditions. Enhanced inservice inspection will be performed at each outage to provide a high level of confidence that only those tubes meeting the required structural margins will remain in service. The reduced allowable leakage rate during normal operation will require operator actions at low levels of primary-to-secondary leakage and thus allow for appropriate actions to maintain plant operation and analyses within current limits. The change does not affect accident mitigation following a SLB since radiological doses are maintained within required limits. Any change in accident leakage is deemed to have a negligible effect on core cooling or containment response to accidents. The change does not introduce any new equipment or any change to existing equipment. No new effects on equipment are created nor are any new malfunctions introduced.

The evaluation described above results in the conclusion that the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Does not involve a significant reduction in a margin of safety

The proposed change maintains the required structural margins of the steam generator tubes for both normal and accident conditions. Confirmation that these margins will be maintained will be accomplished through enhanced inservice inspections at each outage. These inspections will help ensure that only those tubes meeting required margins will remain in service. Margins of safety related to potential primary-to-secondary leakage will be maintained by: (1) implementation of the reduced allowable leakage limit during normal operation and (2) analyses confirming that the leak rate during the postulated SLB will result in off site doses that are less than the 10CFR100 limits.

Plugging of steam generator tubes reduces the reactor coolant system flow margin for core cooling. However, implementation of the proposed change is expected to result in plugging of fewer tubes than with the current criteria. Thus, implementation of the proposed change will maintain the margin of flow that may have otherwise been reduced by tube plugging.

Based on the foregoing description, the proposed change does not result in a significant reduction in a margin of safety.