CANDU STEAM GENERATOR FITNESS-FOR-SERVICE GUIDELINES

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ABSTRACT

Steam generator tubes function as part of the CANDU[3] reactor primary heat transport system coolant pressure boundary. The occurrence of degradation mechanisms in nuclear steam generators has led to the need for industry-standard documents that contain uniform evaluation methods for the fitness-for-service assessment of degraded steam generator tubes. Proposed Fitness-for-Service Guidelines (FFSG) for in-service evaluation of steam generator tubes and preheater tubes in CANDU nuclear power plants are under development. The FFSG are intended to provide Canadian nuclear industry-standard acceptance criteria and evaluation procedures for assessing individual flaws detected during inspection. In addition, they provide a consistent approach for assessing the adequacy of the condition of the entire population of steam generator tubes in a reactor unit. The FFSG are based on safety-related *Performance Criteria* that require tube structural integrity be maintained during the evaluation period, that operational leak rate is monitored and does not exceed the allowable limit, and consequential leakage during postulated upset or abnormal events is acceptable. The assessments would typically be used to justify continued operation for a planned interval and/or to justify the level of in-service inspection. This paper provides an introduction to the proposed FFSG for CANDU steam generator tubes.

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1. INTRODUCTION

Steam generator tubes function as part of the CANDU reactor primary heat transport system coolant pressure boundary. The occurrence of degradation mechanisms in nuclear steam generators has led to the need for industry-standard documents that contain uniform evaluation methods for the fitness-for-service assessment of degraded steam generator tubes. There is a need to formalize periodic monitoring of the condition of aging tubes, and to quantify safety margins regarding tube integrity and potential radioactive releases to the environment. A significant effort is underway under the Electric Power Research Institute (EPRI)/Nuclear Energy Institute steam generator degradation management program to develop industry standard evaluation methods for degraded steam generator tubes [1]. Previously, U.S. NRC criteria that address the limits for tube repair was provided in Regulatory Guide 1.121 [2]. More recently, a performance based approach was proposed by the U.S. NRC in the form of the Steam Generator Tube Rule which subsequently became the NRC draft Generic Letter 97-XX [3] and the draft Regulatory Guide DG-1074 [4].

There is a need for fitness-for-service guidelines specifically for CANDU reactors used in the Canadian nuclear industry. This is due to the unique set of tube and tube support configurations, the use of Monel 400 and Incoloy 800 materials in addition to Inconel 600, and other CANDU-specific issues. A review of international fitness-for-service assessment approaches for steam generator tubes, from the viewpoint of applicability to CANDU steam generator tubes, is provided in reference [5]. Methodologies for probabilistic assessments specific to CANDU steam generator tubes are provided in references [6] and [7].

A multi-disciplinary team with representatives from Ontario Hydro and Atomic Energy of Canada Ltd. are developing Fitness-for-Service Guidelines (FFSG) for in-service evaluation of steam generator tubes and preheater tubes in CANDU nuclear power plants [8]. The proposed FFSG are intended to provide industry-standard acceptance criteria and evaluation procedures that will lead to more consistent assessments being produced throughout the Canadian nuclear industry. This paper provides an introduction to the proposed CANDU steam generator tube FFSG. The role of the FFSG in the steam generator tube degradation management program is first provided, followed by an overview of the FFSG. The concept of *Performance Criteria* is described, along with the approach for periodic assessments. A qualitative overview of an example assessment is then given, followed by current issues and planned future developments.

2. ROLE OF FFSG IN STEAM GENERATOR DEGRADATION MANAGEMENT

Steam generator tube degradation management includes water chemistry control, in-service inspection and monitoring, condition assessment of the steam generator tubes, and steam generator cleaning and other maintenance activities.

Condition assessments of steam generator tubes in CANDU plants are under the jurisdiction of the Canadian Standard CAN/CSA-N285.4-94 [9]. The rules in CAN/CSA-N285.4-94 permit acceptance of indications with predicted end-of-evaluation-period depths not exceeding 40% of the wall thickness. When an indication does not satisfy this criterion, a fitness-for-service assessment to demonstrate integrity is required. The proposed FFSG are intended to provide the evaluation procedures and acceptance criteria for such fitness-for-service assessments. FFSG assessments would typically be used to justify continued operation of steam generator tubes in a degraded condition, and/or as a means to justify the level of in-service inspection. Degradation mechanisms covered by the FFSG include intergranular attack, stress corrosion cracking, fatigue, pitting, fretting wear, wall thinning caused by corrosion and erosion, and localized tube deformation such as denting.

3. OVERVIEW OF FFSG

3.1 Approach

The main objectives of the FFSG are to provide reasonable assurance that: steam generator tube structural integrity is maintained; there are adequate margins between estimated emissions and applicable site dose limits with consideration of possible consequential steam generator tube failure. The FFSG are based on methods previously developed to assess tube degradation at Ontario Hydro Bruce A and B Nuclear Generating Stations. The FFSG are also based on safety-related *Performance Criteria*, which have been proposed under the EPRI program [1], as well as by the U.S. NRC [3,4]. *Performance Criteria* are described in Part 4 of this paper. Tube leakage is permitted in this approach, provided the *Performance Criteria* are satisfied.

The FFSG provide procedures for determining the Maximum Tolerable Flaw Size (MTFS), which is used to determine the acceptance limit for tube repair. Procedures are provided for a backward looking, *Condition Monitoring Assessment* of the entire population of tubes to validate and or adjust predictive methods based on service experience. The FFSG also include a forward-looking, *Operational Assessment* to demonstrate that the *Performance Criteria* will be satisfied during the next evaluation period. Both deterministic and probabilistic assessments are covered. The FFSG place a significant reliance on the use of statistical analysis methods. This is due to the large number of steam generator tubes in a reactor unit, the statistical nature of Probability Of Detection (POD) of flaws and flaw sizing error, and uncertainty and variability in flaw growth rate.

3.2 Organization of FFSG

The FFSG consist of two sections. Section I contains the acceptance criteria and evaluation procedures. These include the *Performance Criteria*, procedures for evaluation of individual detected flaws, procedures for the *Condition Monitoring Assessment* and *Operational Assessment* of the entire tube population, procedures to assess the reactor shutdown operational leakage limit, and guidelines for tube repair. The main body of Section I provides mandatory acceptance criteria, while detailed evaluation procedures are provided in the nonmandatory Appendices. Material properties and derived parameters, such as material flow stress and leak rates, are provided in Section II. The technical bases for the FFSG will be provided in a separate document.

4. PERFORMANCE CRITERIA

The bases for the acceptance criteria in the FFSG are the *Performance Criteria*. The *Performance Criteria* require that tube structural integrity be maintained during the evaluation period, that operational leak rate is monitored and does not exceed the allowable limit, and that consequential leakage during postulated upset or abnormal loading events is acceptable. Deterministic and probabilistic *Performance Criteria* are provided.

4.1 Structural Integrity Criteria

Adequate structural integrity of the tubes must be maintained during the evaluation period. Separate *Performance Criteria* apply to detected or postulated flaws, and to localized tube deformation.

4.1.1 Sharp, Crack-Like, or Blunt Flaws, Wall Thinning

For blunt flaws, prevention of crack initiation must be demonstrated. Otherwise, the blunt flaw must be treated as crack-like. Safety margins on load must be evaluated using either acceptance criteria prohibiting leakage, or acceptance criteria permitting leakage. The required safety factors on load are equivalent to those in the ASME Boiler and Pressure Vessel Code, Sections III [10] and XI [11]. The acceptance criteria prohibiting leakage require that safety factors on load for prevention of penetration of the flaw through the wall must be satisfied. In addition, protection against unstable rupture of a related postulated through-wall flaw must be demonstrated. The acceptance criteria permitting leakage allow through-wall flaw penetration provided *Leak-Before-Break* is demonstrated using required safety factors on load for prevention of tube rupture, and provided acceptable consequential leakage through all flaws during the most limiting postulated upset or abnormal loading events, including accident events, is demonstrated. In-situ pressure testing is permitted as a method of demonstrating structural integrity margins.

4.1.2 Localized Tube Deformation

Localized tube deformation includes denting. Prevention of crack initiation must be demonstrated. Otherwise, the localized tube deformation must be evaluated as crack-like.

4.2 Operational Leak Rate Criteria

The operational leak rate limit, and the frequency of leak rate monitoring, must provide reasonable assurance of *Leak-Before-Break* during normal operation, and protection against tube rupture during upset or abnormal loading events. Reasonable assurance must be provided that the contribution of the corresponding through-wall flaw(s) to consequential leakage during upset or abnormal loading events is acceptable.

4.3 Consequential Leakage Criteria

Potential leakage during the most limiting postulated upset or abnormal loading events, including accident events, must be assessed considering all degradation mechanisms. Assessment must be based on a flaw distribution that reflects the detected flaws, and must cover un-inspected tubes. The *Consequential Leakage Assessment* must demonstrate that the leakage during the event is acceptable, based on an adequate margin between estimated total accumulated dose and applicable site dose limits.

5. ELEMENTS OF FFSG PERIODIC ASSESSMENTS

When tube degradation is detected, the condition of the steam generator tubes is compared periodically against *Performance Criteria* to provide reasonable assurance that the tubes remain capable of fulfilling their intended safety functions. The steam generator tubes are assessed for operation over an evaluation period, which may be the time interval between the current steam generator tube inspection and the next scheduled inspection. Periodic assessments include four main elements. The first three are: (i) *Evaluation of Detected Flaws or Localized Tube Deformation*; (ii) backward-looking *Condition Monitoring Assessment* of fitness-for-service; (iii) forward-looking *Operational Assessment* of fitness-for-service. These assessments are performed on a regular basis, and are described below in Parts 5.1, 5.2 and 5.3, respectively. The fourth element is *Assessment of Condition Causing Leakage*, and is performed when a reactor unit that had primary to secondary-side leakage is in a shutdown state, and the source of leakage has been detected. A flowchart of consecutive, periodic assessments is shown in Figure 1. A portion of the effort in the FFSG assessment, such as determining Maximum Tolerable Flaw Size, can be performed prior to the availability of inspection results in order to expedite the assessment during the reactor outage.

5.1 Evaluation of Detected Flaws or Localized Tube Deformation

This element of the FFSG is used to determine the tube repair criteria. The root cause(s) of degradation, including flaw formation and flaw growth, are assessed. For blunt flaws or localized tube deformation, prevention of crack initiation is demonstrated. Otherwise, the blunt flaw or localized tube deformation is evaluated as a crack. For flaws, either acceptance criteria prohibiting leakage or acceptance criteria permitting leakage must be satisfied. When acceptance criteria prohibiting leakage are used, flaw stability against 100% through-wall flaw penetration must be demonstrated for the end-of-evaluation-period flaw size using safety factors on load. For example, for an axial flaw

$$(SF)\sigma_h \leq \sigma_h' \tag{1}$$

where σ_h is the primary membrane hoop stress at the flaw, σ_h' is the hoop stress at ligament instability of the part-through-wall flaw, and *(SF)* is the required safety factor on load. In addition, protection against rupture of the related postulated through-wall flaw must be demonstrated.

When acceptance criteria permitting leakage are used, a *Leak-Before-Break* assessment is performed to demonstrate that required safety factors are maintained prior to detection of the leaking flaw during normal operation and shutdown of the reactor unit. A *Consequential Leakage Assessment* is also performed. For the total population of tubes in the reactor unit, the total leakage due to all degradation mechanisms must not exceed the acceptable total leakage.

$$m_F \leq m_a \tag{2}$$

where m_F is the leakage during the upset or abnormal loading event due to all degradation mechanisms, and m_a is the acceptable total leakage for the upset or abnormal event.

5.2 Condition Monitoring Assessment

When an *Operational Assessment* of fitness-for-service has been performed for the previous evaluation period, a backward-looking, *Condition Monitoring Assessment* of the entire population of tubes in the reactor unit is performed prior to unit restart. The *Condition Monitoring Assessment* evaluates whether the *Performance Criteria* had been satisfied during the previous evaluation period. The *Condition Monitoring Assessment* compares the predictive methods and input variables of the previous *Operational Assessment* against actual steam generator tube degradation. Modifications to the predictive methods and input variables are identified and are implemented into current and future *Operational Assessments*. A flowchart of *Condition Monitoring Assessment* of flaws, including wall thinning, is shown in Figure 2.

The projected flaw size distribution from the previous *Operational Assessment* is compared with the current flaw size distribution from the current inspection results. The method of predicting flaw size distribution in the current *Operational Assessment* is revised to conform with the current inspection results. For flaws, either the acceptance criteria prohibiting leakage or acceptance criteria permitting leakage described in Part 5.1 of this paper must be satisfied.

5.3 Operational Assessment

A forward-looking, *Operational Assessment* of fitness-for-service of the entire population of tubes in the reactor unit is performed to demonstrate that the *Performance Criteria* are satisfied during the next evaluation period. The *Operational Assessment* considers the projected future condition of the tubes during the evaluation period based on the inspection results, the predicted flaw growth rates, and corrective actions implemented at that time. When a *Condition Monitoring Assessment* demonstrates compliance with the *Performance Criteria* for each degradation mechanism, the *Operational Assessment* must be performed within 90 days after unit restart. When a *Condition Monitoring Assessment* has not demonstrated compliance with the *Performance Criteria* for a degradation mechanism, the *Operational Assessment* must be performed prior to unit restart. The *Operation Assessment* is similar to the *Condition Monitoring Assessment*, except that the *Operational Assessment* is based on the projected condition of the population of tubes at the end of the evaluation period.

6. QUALITATIVE OVERVIEW OF AN EXAMPLE FFSG ASSESSMENT OF FRETTING FLAWS

The purpose of this qualitative overview of a fictitious example FFSG assessment is to illustrate how the elements of the FFSG are applied. The assessment is for fretting flaws at tube supports, which have been characterized with an axial orientation in the tubes. All of the steam generators were previously subjected to 100% inspection at the tube support locations of concern. The current inspection results are from 100% inspection of the same tube support locations in the steam generators with the most severe fretting. The purpose of the inspection is to confirm the predicted fretting wear rates, and to justify continued operation of all of the steam generators for the next evaluation period. In the previous inspection outage, an *Operational Assessment* was performed. The steps of the assessment are summarized in Table 1.

6.1 Evaluation of Detected Flaws

Since the fretting has already been characterized from the previous inspection, the tube plugging criterion is determined prior to the unit outage. From the previous *Operational Assessment*, the root cause is fretting at the tube supports due to flow-induced vibration. The flaw axial length is taken to be the width of the tube support. Flaw growth is predicted based on fretting rates from the previous *Operational Assessment*. For detected flaws, acceptance criteria prohibiting leakage are used to calculate the Maximum Tolerable Flaw Size (MTFS) of the part-through-wall flaw. The tube plugging criterion is equal to: MTFS minus (flaw sizing error + flaw growth). In addition, a through-wall flaw of the same length is postulated and evaluated to be stable.

6.2 Condition Monitoring Assessment

At this stage, the inspection results from the current outage are now available. The current inspection results are from eddy current with a limited amount of UT results to confirm the characteristic overall flaw dimensions. The nature and locations of fretting confirms that the root cause analysis from the previous *Operational Assessment* is valid. Probability distributions of current flaw depths in the inspected steam generators are compared with corresponding distributions from the previous *Operational Assessment* to confirm that the predictions of initial flaw size and flaw growth are conservative. Acceptance criteria permitting leakage are used for the assessment of the entire tube population. This includes a *Leak-Before-Break* assessment, and a *Consequential Leakage Assessment*. The assessment results demonstrate that the *Performance Criteria* were satisfied during the previous evaluation period.

6.3 Operational Assessment

Since the *Condition Monitoring Assessment* demonstrated compliance with the *Performance Criteria*, the reactor unit is permitted to restart. An *Operational Assessment* is performed within 90 days of restart. Based on the current flaw size probability distributions for inspected steam generators and the previous probability distributions for the un-inspected steam generators, probability distributions of predicted end-of-evaluation period flaw depths are developed. Based on service experience and previous *Operational Assessments* , crack initiation is not a concern. Acceptance criteria permitting leakage are used to demonstrate that the *Performance Criteria* will be satisfied during the next evaluation period.

7. CURRENT ISSUES WITH FFSG

The development of the FFSG, and experience gained from actual steam generator tube assessments, leads to the identification of the following issues:

(a) Improved Probability of Detection (POD) and flaw sizing capability is required, along with improved quantification of current POD and flaw sizing error.

(b) Additional information on flaw growth rates, including laboratory results and in-service data, is required.

(c) Additional experimental results on tube burst and leak rates are required.

(d) An improved understanding of crack initiation at blunt flaws such as pits, and at localized tube deformation, such as dents, is required. The latter includes the role of residual stress.

8. FUTURE DEVELOPMENTS

Currently, additional work on the flaw evaluation procedures in Section I of the FFSG, as well as work on Section II, needs to be completed. Future R&D results need to be incorporated to improve the evaluation procedures. The technical basis document also needs to be developed. Canadian nuclear industry consensus on the FFSG still needs to be obtained. The FFSG must also be submitted to the Canadian regulatory authorities for review.

9. CONCLUSIONS

(a) The proposed Fitness-for-Service Guidelines for steam generator tubes in CANDU reactors are based on safety-related structural integrity and leakage *Performance Criteria*. Procedures to evaluate individual detected flaws are provided. Backward-looking *Condition Monitoring Assessments* of the tube population to validate and or adjust predictive methods, and forward-looking, *Operational Assessments*, are also included.

(b) Improvements to the inspection capability, as well as improved quantification of the current inspection capability, are required. This includes flaw POD and flaw sizing error. Additional R&D results in topics related to FFSG assessments, such as tube burst and leak rates, are also required.

(c) Canadian nuclear industry consensus on the FFSG needs to be obtained. The FFSG must also be submitted to the regulatory authorities for review.

10. ACKNOWLEDGEMENTS

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TABLE 1 EXAMPLE FFSG ASSESSMENT OF FRETTING FLAWS

TASK SCHEDULE	ELEMENT OF PERIODIC ASSESSMENT	TASK DESCRIPTION
Prior to outage	*Evaluation of Detected Flaws*	Input flaw characterization from previous *Operational Assessment*, input applied stresses Use root cause analysis from previous *Operational Assessment* -> fretting at tube supports Predict flaw growth based on fretting rates from previous *Operational Assessment* Use acceptance criteria prohibiting leakage to calculate MTFS -> part-thru-wall flaw stability, postulated thru-wall flaw stability Use MTFS to determine plugging criterion
During outage	*Condition Monitoring Assessment*	Input current inspection results, confirm flaw characterization including flaw length, input applied stresses Confirm root cause analysis from previous *Operational Assessment* Develop probability distribution of current flaw depths, compare with distribution from previous *Operational Assessment* Confirm flaw growth rates from previous *Operational Assessment* Use acceptance criteria permitting leakage -> Leak-Before-Break, Consequential Leakage Assessment Assess whether the *Performance Criteria* were satisfied during the previous evaluation period
After restart but within 90 days	*Operational Assessment*	Predict flaw growth rates for the next evaluation period Develop probability distribution of predicted end-of-evaluation-period flaw depths Evaluate for crack initiation Use acceptance criteria permitting leakage -> Leak-Before-Break, Consequential Leakage Assessment Assess whether the *Performance Criteria* will be satisfied during the next evaluation period

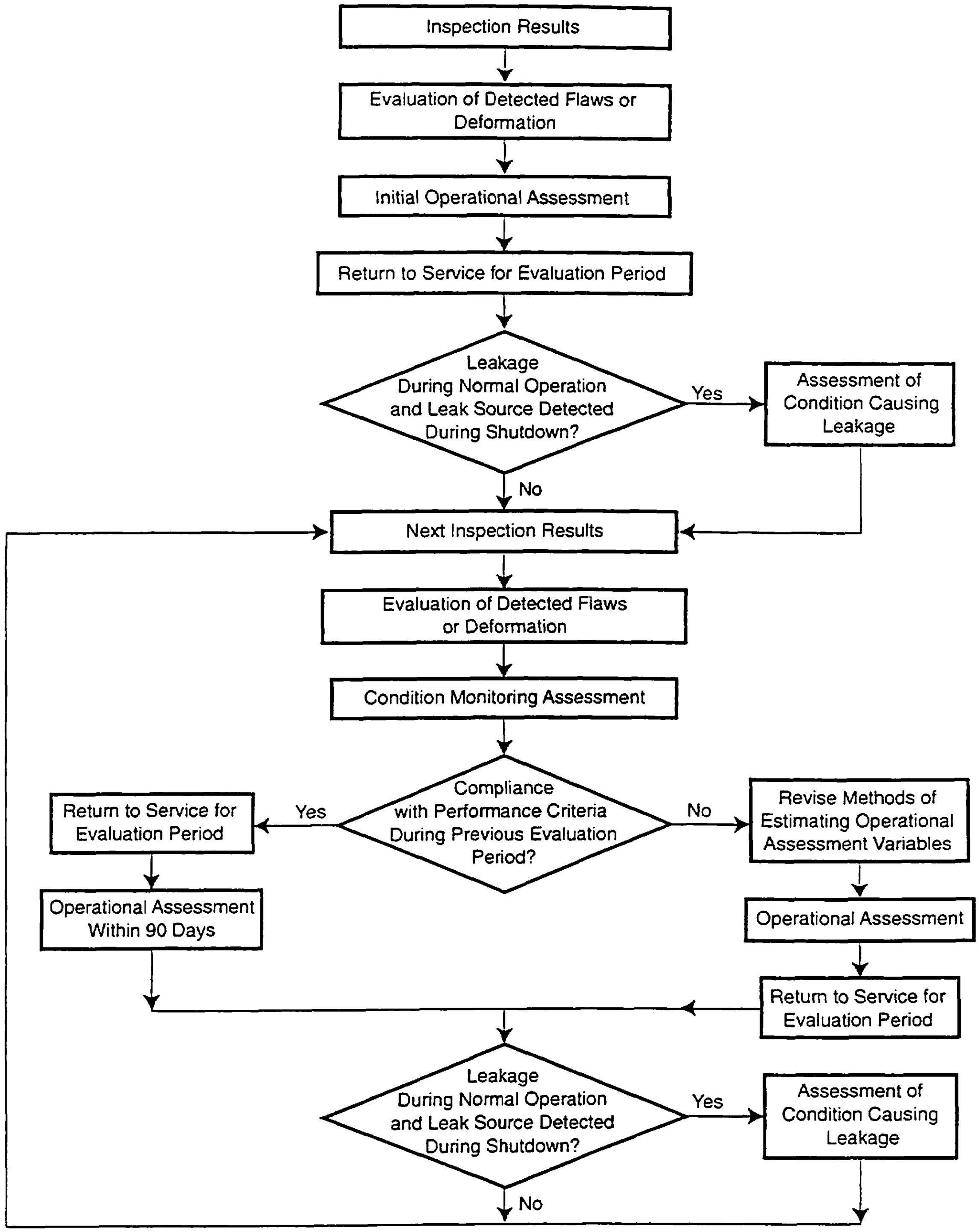

FIGURE 1 FLOWCHART OF STEAM GENERATOR TUBE PERIODIC ASSESSMENTS

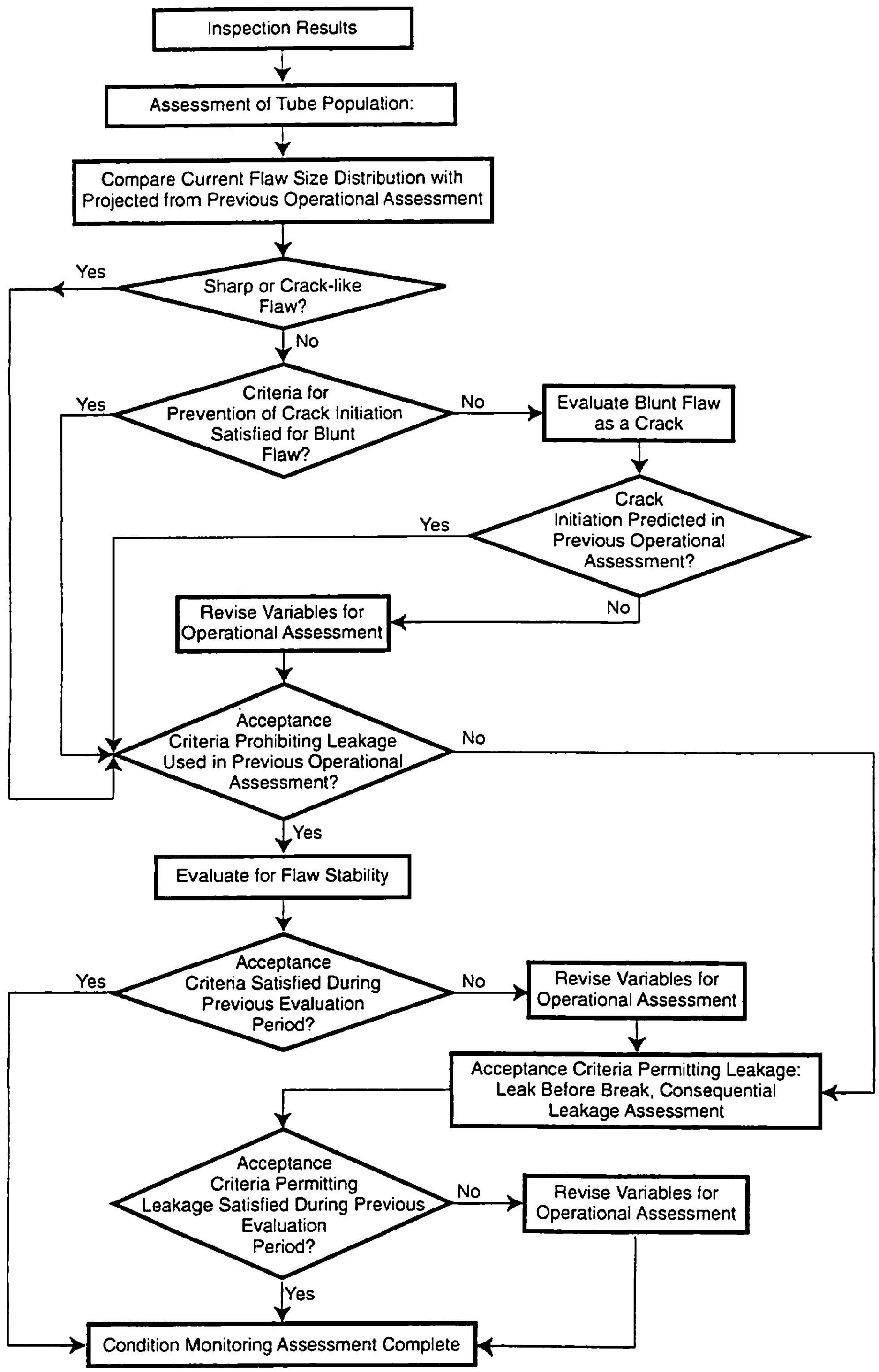

FIGURE 2 FLOW CHART OF CONDITION MONITORING ASSESSMENT OF FLAWS

DISCUSSION

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Paper: CANDU Steam Generator Fitness-for-Service Guidelines

Questioner: E.G. Price

Question/Comment:

What data exist to justify the distribution of defect sizes remains similar after a further time period? In particular for the SCC cracking phenomena?

Response:

The flaw size frequency distribution at the end of an evaluation period is predicted using a frequency distribution for the current flaw size, and a frequency distribution for flaw growth rate. When these two distributions are used, the predicted flaw size frequency distribution for the end of the evaluation period will be different in terms of both the mean of the distribution and the variance, or shape, of the distribution. It is not assumed that the shape of the flaw size distribution remains the same over the evaluation period. When the steam generator tubes are inspected at the end of the evaluation period, a new current flaw size distribution is fitted to the in-service inspection data. This new current flaw size distribution will have a different mean and variance, or shape, from the flaw size distribution from the previous inspection. The new current flaw size distribution is then compared against the previously predicted flaw size distribution to validate or correct the flaw growth rate distribution.