December 13, 2000

Mr. Valeri Tolstykh
Regulatory Activities Unit
Safety Assessment Section
Division of Nuclear Installation Safety
International Atomic Energy Agency
Wagramer Strasse 5
P.O. Box 100, A-1400
Vienna, Austria

Dear Mr. Tolstykh:

Enclosed are the following IRS reports:

- MANAGING REGULATORY COMMITMENTS MADE BY POWER REACTOR LICENSEES TO THE NRC STAFF (NRC Regulatory Issue Summary 2000-17).

- ISSUES STEMMING FROM NRC STAFF REVIEW OF RECENT DIFFICULTIES EXPERIENCED IN MAINTAINING STEAM GENERATOR TUBE INTEGRITY (NRC Regulatory Issue Summary 2000-22).

Each report is being submitted in the following two media: (1) a hard copy of the input file for the AIRS database; and (2) a 3.5-inch HD diskette containing the input file for the AIRS database in Microsoft Word 6.0 format.

If you have any questions regarding these reports, please call Eric J. Benner of my staff. He can be reached at (301) 415-1171.

Sincerely,

/RA John R. Tappert For/
Ledyard B. Marsh, Chief
Events Assessment, Generic Communications and
Non-Power Reactors Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Enclosures: as stated

cc w/enclosures 1 and 2:
Mr. Lennart Carlsson
Nuclear Safety Division
Nuclear Energy Agency
Organization for Economic
Cooperation and Development
Le Seine Saint Germain
12, Boulevard des Iles
92130, Issy-les-Moulineaux, France

Mr. Valeri Tolstykh December 13 2000
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INCIDENT REPORTING SYSTEM

IRS NO.	EVENT DATE	DATE RECEIVED
	2000/09/21	

EVENT TITLE

MANAGING REGULATORY COMMITMENTS MADE BY POWER REACTOR LICENSEES TO THE NRC STAFF (Regulatory Issue Summary 2000-17)

COUNTRY	PLANT AND UNIT	REACTOR TYPE
USA	Generic	(BWR or PWR)

INITIAL STATUS	RATED POWER (MWe NET)
N/A	N/A

DESIGNER	1st COMMERCIAL OPERATION
(WEST, GE, CE, B&W)	N/A

ABSTRACT

This IRS report discusses the Nuclear Energy Institute (NEI) guidance document, "Guidelines for Managing NRC Commitment Changes" (NEI-99-04)(ADAMS Accession No. ML003680088), describes an acceptable way for licensees to control regulatory commitments. The NRC encourages licensees to use the NEI guidance or similar administrative controls to ensure that regulatory commitments are implemented and that changes to the regulatory commitments are evaluated and, when appropriate, reported to the NRC.

MANAGING REGULATORY COMMITMENTS MADE BY POWER REACTOR LICENSEES TO THE NRC STAFF (Regulatory Issue Summary 2000-17)

Please refer to the dictionary of codes corresponding to each of the sections below and to the coding guidelines manual.

1.	**Reporting Categories:**	**1.4**		
2.	**Plant Status Prior to the Event:**	**2.0**		
3.	**Failed/Affected Systems:**	**3.Z**		
4.	**Failed/Affected Components:**	**4.0**		
5.	**Cause of the Event:**	**5.1.0**		
6.	**Effects on Operation:**	**6.0**		
7.	**Characteristics of the Incident:**	**7.0**		
8.	**Nature of Failure or Error:**	**8.0**		
9.	**Nature of Recovery Actions:**	**9.0**		

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

September 21, 2000

NRC REGULATORY ISSUE SUMMARY 2000-17
MANAGING REGULATORY COMMITMENTS MADE BY POWER REACTOR LICENSEES TO THE NRC STAFF

ADDRESSEES

All holders of operating licenses for nuclear power reactors.

INTENT

The U.S. Nuclear Regulatory Commission (NRC) is issuing this regulatory issue summary (RIS) to inform the addressees that the Nuclear Energy Institute (NEI) guidance document, "Guidelines for Managing NRC Commitment Changes" (NEI-99-04)(ADAMS Accession No. ML003680088), describes an acceptable way for licensees to control regulatory commitments. The NRC encourages licensees to use the NEI guidance or similar administrative controls to ensure that regulatory commitments are implemented and that changes to the regulatory commitments are evaluated and, when appropriate, reported to the NRC. This RIS does not transmit any new requirements or staff positions. No specific action or written response is required.

BACKGROUND INFORMATION

Various activities undertaken by the staff and the nuclear industry in the early 1990s culminated in the issuance of SECY-95-300, “Nuclear Energy Institute's Guidance Document, 'Guideline for Managing NRC Commitments,'" dated December 20, 1995. The industry document and related Commission paper contained guidance for handling licensing basis information that was not subject to controls defined in NRC regulations. The NEI guidance described a process that licensees can use to modify or delete regulatory commitments and provided criteria to decide if and when changes to regulatory commitments should be reported to the NRC. The use of this guidance was intended to clarify the standing of regulatory commitments and give licensees the confidence and flexibility to modify or delete regulatory commitments shown to be inefficient or ineffective.

In SECY-98-224, "Staff and Industry Activities Pertaining to the Management of Commitments Made by Power Reactor Licensees to the NRC," dated September 28, 1998, the staff described its activities related to commitment management strategies, audits of commitment management programs at power reactor facilities, and discussions with stakeholders. In SECY-98-224, the staff also (1) discussed its rationale for maintaining regulatory commitments as an element of the licensing bases for power reactors and (2) described the expected management of regulatory commitments by licensees' administrative processes and the proposed internal

ML003741774

guidance for the NRC staff. Following the plan described in SECY-98-224, the staff worked with NEI and licensees as they revised the industry guidance document. These efforts were reflected, along with the insights from participating licensees, in the development of NEI 99-04. The staff's review of NEI 99-04 and its finding that the revised guidance remained useful for controlling regulatory commitments are described in SECY-00-045, "Acceptance of NEI 99-04, 'Guidelines for Managing NRC Commitments,'" dated February 22, 2000, and in the letter from S. Collins (NRC) to R. Beedle (NEI) dated March 31, 2000 (ADAMS Accession No. ML003696998).

SUMMARY OF ISSUE

The NRC staff sees benefits in maintaining regulatory commitments as an integral part of control by licensees and the NRC staff of each facility's licensing basis information. The staff has described, in various Commission papers and internal guidance documents, a hierarchal structure for the various elements of a facility's licensing basis. The approach to the hierarchy is presented in terms of the change control, reporting requirements, and other attributes of the different elements of the licensing basis [see NRR Office Letter 807, "Control of Licensing Bases for Operating Reactors" (ADAMS Accession No. ML003693397)]. The levels of the hierarchy are (1) obligations or regulatory requirements that require prior NRC approval of proposed changes, (2) mandated licensing basis documents, such as the updated final safety analysis report, for which the NRC has established requirements for content, change control and reporting, and (3) regulatory commitments controlled by licensee and NRC administrative processes.

The guidance for licensees provided by NEI 99-04 and related guidance developed for the NRC staff [e.g., NRR Office Letter 900, "Managing Commitments Made by Licensees to the NRC" (ADAMS Accession No. ML003692416)] address the third level of the licensing bases hierarchy. The process and guidance provided in NEI 99-04 are a refinement of the process and guidance described in NEI's previous guidance document and SECY-95-300. The revised guidance clarifies that not all corrective actions described in correspondence with the NRC staff are regulatory commitments. The guidance in NEI 99-04 also suggests that licensees use information management systems, annotations to procedures, or other methods to ensure the traceability of regulatory commitments after implementation.

The staff has reviewed NEI 99-04 and finds that it offers an acceptable way to manage regulatory commitments. Definitions and other guidance in NEI 99-04 are consistent with the principles described in Commission papers and the staff's internal guidance. The NRC encourages licensees to use the NEI guidance or similar administrative controls to ensure that regulatory commitments are implemented and that changes to the regulatory commitments are evaluated and, when appropriate, reported to the NRC. The value of maintaining a working commitment management program is that it supports a common understanding by licensees, the staff, and other stakeholders of how a licensing issue is being resolved and how the matter will be controlled in the future. The NRC staff will continue to assess how the industry and individual licensees are managing regulatory commitments to determine if changes in policy or additional regulatory actions are called for.

BACKFIT DISCUSSION

This RIS does not require any action or written response; therefore, the staff did not perform a backfit analysis.

FEDERAL REGISTER NOTIFICATION

A notice of opportunity for public comment on this RIS was not published in the *Federal Register* because it is informational and pertains to a staff position that does not represent a departure from current regulatory requirements and practice. This RIS requires no action or written response on the part of an addressee.

PAPERWORK REDUCTION ACT STATEMENT

This RIS does not request any information collection.

If there is any question about this matter, please contact the person listed below or the appropriate Office of Nuclear Reactor Regulation project manager for a specific nuclear power plant.

/RA/

David B. Matthews, Director
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Technical contact: William D. Reckley, NRR
301-415-1323
E-mail: wdr@nrc.gov

Attachment: List of Recently Issued NRC Regulatory Issue Summaries

INCIDENT REPORTING SYSTEM

IRS NO. | **EVENT DATE** 2000/11/03 | **DATE RECEIVED**

EVENT TITLE

ISSUES STEMMING FROM NRC STAFF REVIEW OF RECENT DIFFICULTIES EXPERIENCED IN MAINTAINING STEAM GENERATOR TUBE INTEGRITY
(NRC Regulatory Issue Summary 2000-22)

COUNTRY
USA

PLANT AND UNIT
Generic

REACTOR TYPE
(BWR or PWR)

INITIAL STATUS
N/A

RATED POWER (MWe NET)
N/A

DESIGNER
(WEST, GE, CE, B&W)

1st COMMERCIAL OPERATION
N/A

ABSTRACT

This IRS report discusses issues stemming from the staff's review of (1) the circumstances of the steam generator (SG) tube failure at Indian Point Unit 2 (IP-2), (2) the inability to maintain SG tube integrity at Arkansas Nuclear One Unit 2 (ANO-2) and (3) the analyses done to demonstrate that SG tube integrity at these facilities would be maintained during subsequent operation.

ISSUES STEMMING FROM NRC STAFF REVIEW OF RECENT DIFFICULTIES EXPERIENCED IN MAINTAINING STEAM GENERATOR TUBE INTEGRITY
(NRC Regulatory Issue Summary 2000-22)

Please refer to the dictionary of codes corresponding to each of the sections below and to the coding guidelines manual.

1.	**Reporting Categories:**	1.2.2		
2.	**Plant Status Prior to the Event:**	2.0		
3.	**Failed/Affected Systems:**	3.AH		
4.	**Failed/Affected Components:**	4.2.6		
5.	**Cause of the Event:**	5.1.1.1		
6.	**Effects on Operation:**	6.0		
7.	**Characteristics of the Incident:**	7.2		
8.	**Nature of Failure or Error:**	8.2		
9.	**Nature of Recovery Actions:**	9.0		

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

November 3, 2000

NRC REGULATORY ISSUE SUMMARY 2000-22 ISSUES STEMMING FROM NRC STAFF REVIEW OF RECENT DIFFICULTIES EXPERIENCED IN MAINTAINING STEAM GENERATOR TUBE INTEGRITY

ADDRESSEES

All holders of operating licenses for pressurized-water reactors (PWRs), except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

INTENT

The U.S. Nuclear Regulatory Commission (NRC) is issuing this regulatory issue summary to inform pressurized-water reactor licensees of issues stemming from the staff’s review of (1) the circumstances of the steam generator (SG) tube failure at Indian Point Unit 2 (IP-2), (2) the inability to maintain SG tube integrity at Arkansas Nuclear One Unit 2 (ANO-2) and (3) the analyses done to demonstrate that SG tube integrity at these facilities would be maintained during subsequent operation. This RIS does not transmit any new requirements or staff positions. No specific action or written response is required.

BACKGROUND INFORMATION

Indian Point Unit 2

IP-2 experienced a steam generator tube failure event on February 15, 2000. The event involved the failure of a row 2 tube due to primary water stress corrosion cracking (PWSCC) at the apex of the small radius U-bend. All row 1 tubes had been plugged prior to initial plant operation. The dominant contributor to the crack was hourglass deformation of upper support plate flow slots producing abnormal stresses at the apex of the U-bend. The hourglass deformation of the upper support plate flow slots was associated with in-plane deformation of the support plate caused by denting at the tube-to-support plate intersections. The last inspection of the small radius U-bends before the failure event was in 1997. A reexamination of the 1997 inspection data revealed flaw indications in four small radius U-bends, including the one which failed on February 15, 2000. The flaws had been missed because appropriate corrective action had not been taken to address the poor quality of the data.

By letter dated August 31, 2000 (Accession No. ML003746339), the NRC staff transmitted to the IP-2 licensee the results of a special NRC team inspection addressing the causes of the tube failure event at IP-2 and the licensee's performance during the 1997 SG inspection (NRC Special Inspection Report, Indian Point Unit 2 Steam Generator Tube Failure, Report No. 05000247/2000-010). The staff found that overall the licensee's 1997 SG inspection was deficient. Despite opportunities, the licensee did not recognize and take corrective action for significant conditions adverse to quality in the SG inspection program. The licensee did not adequately account for conditions that adversely affected the detectibility of, or increased susceptibility to, tube flaws. As a result, tubes with PWSCC flaws in the small radius U-bends were left in service after the 1997 inspection, and one of those tubes subsequently failed on February 15, 2000.

On October 11, 2000, the NRC staff issued a technical evaluation report (TER) (Accession No. ML003759418) documenting the NRC staff's review of the February 15, 2000, tube failure, its root causes, the inspections and other actions taken by the licensee to prevent recurrence, and the licensee's operational assessment justifying continued operation. In accordance with the IP-2 technical specifications, plant restart following the post-event outage was subject to NRC approval because the SG tube inspections performed during the outage produced category C-3 results in two of four steam generators (i.e., greater than 1% of the tubes inspected had pluggable indications). NRC had not yet decided on whether to allow IP-2 restart when the licensee decided to install replacement steam generators rather than restart with the existing steam generators. Nevertheless, the TER contains a number of staff conclusions about specific technical aspects of the licensee's corrective actions and its operational assessment to support continued operation with the original steam generators.

Arkansas Nuclear One Unit 2

A tube exhibiting outside diameter stress corrosion cracking (ODSCC) near an egg crate support plate failed to satisfy the applicable burst pressure performance criterion (i.e., the three times normal operating pressure (3 delta P) criterion) in an in situ pressure test during a refueling outage inspection in January 1999. The licensee performed an operational assessment and concluded the unit could be operated until mid cycle and still maintain adequate tube integrity. In situ pressure testing during the mid cycle inspection in November 1999 was terminated when the leakage for a tube with an axial ODSCC indication exceeded the makeup capacity of the test system at a pressure below the 3 delta P criterion. At first the licensee assumed the peak pressure reached during the test to be the burst pressure. The licensee further examined the circumstances of the test and concluded that the test was terminated when ligaments of material in the crack tore and that the likely burst pressure exceeded the 3 delta P criterion. Using this finding as a benchmark, the licensee submitted its operational assessment to the NRC staff, for information, as technical justification that the plant could be operated until the scheduled refueling outage in September 2000 and still maintain tube structural and leakage integrity in accordance with applicable performance criteria. The staff disagreed with this justification and concluded that the licensee had not demonstrated that the applicable performance criteria would continue to be met until the September 2000 refueling outage (NRC letters dated May 2 and June 23, 2000 (Accession Nos. ML003710343 and ML003726321, respectively)).

Accordingly, the licensee submitted a request to change the plant's licensing basis to allow operation of ANO-2 until the September 2000 refueling outage on the basis of a risk-informed demonstration that steam generator tube integrity would meet the acceptance criteria in NRC

Regulatory Guide 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis.” In a letter dated July 21, 2000 (Accession No. ML003734450), the staff told the licensee that the requested license amendment could not be approved. The letter enclosed a safety evaluation documenting the basis for this finding. ANO-2 began shutting down for a mid cycle inspection on the same day, i.e., July 21, 2000.

SUMMARY OF ISSUES

The NRC special inspection report and technical evaluation report for IP-2, and the safety evaluation for ANO-2 (including NRC letters dated May 2 and June 23, 2000, referenced in the safety evaluation) identify a number of technical issues involving degradation management, tube inspections, in situ pressure testing, and operational assessment. The staff believes this information will be useful to the rest of the industry in managing SG tube integrity, and may be of interest to other stakeholders as well. By letter dated October 6, 2000, the Nuclear Energy Institute provided the NRC with an industry evaluation of recent operating experience on many of the same topics discussed below.

Issue 1: Consideration of relevant operating experience and appropriate diagnostic, corrective, or compensatory measures to ensure tube integrity.

Issue 2: Assessment of the root causes of all degradation mechanisms at a plant and appropriate diagnostic, corrective, or compensatory measures to ensure tube integrity.

Issues 1 and 2 derive from two performance issues identified in the IP-2 1997 inspections (and documented in the NRC special inspection report dated August 31, 2000).

The failure mechanism that led to the February 15, 2000 SG tube failure at IP-2 was essentially the same mechanism that caused a tube rupture at Surry 2 in 1976. This failure mechanism was PWSCC as a result of abnormally high stress at the apex of the affected small radius U-bend. The abnormal stress was associated with an ovalized condition at the apex caused by the inward displacement of the legs of the U-bend as a result of hourglass deformation of the uppermost tube support plate flow slots which was induced by denting in the tube-to-tube support plates. The amount of hourglass deformation adjacent to the failed tube at IP-2 was measured as 0.47 inches, after the failure.

During the 1997 inspection, a PWSCC indication was detected for the first time in the apex of a small radius U-bend. One performance issue cited in the special inspection report is that the licensee and its contractor did not identify or evaluate the potential tube integrity implications of this type of indication. Another performance issue cited in the report is that the licensee did not have a procedure, a method, or criteria for determining if significant hourglass deformation of the uppermost support plate had occurred, even though the technical specifications required the reporting of significant hourglass deformation of the uppermost support plates. The licensee did not assess hourglass deformation as a potential causal factor for the apex indication found during the inspection, despite evidence of active denting in the uppermost support plates.

Issue 3: Data quality.

Data quality depends on the degree to which the eddy current signal from a flaw can be masked or distorted by signals from sources other than the flaw. Data quality directly affects the ability to detect and size flaws. The signals from sources other than the flaw are often called “noise”. The amplitude of the noise signal and signal-to-noise ratio are important measures of data quality.

Issue 3 derives from a performance issue identified during the IP-2 1997 inspection (and documented in the NRC special inspection report dated August 31, 2000). A hindsight analysis, completed after the tube failure on February 15, 2000, of eddy current data collected during the 1997 SG inspection with a midrange plus-point coil revealed four indications were missed during the 1997 data analysis. One of the missed indications was in the tube that failed on February 15, 2000.

The NRC special inspection report found that the 1997 eddy current data was noisy making detection difficult. However, in 1997, the licensee did not identify the potential masking effect of noise as a significant condition adverse to quality. The licensee did not increase the level of review or more carefully examine existing data. The special inspection report noted that these omissions were important since conditions indicating an increased susceptibility to PWSCC were identified during the 1997 inspection (see issues 1 and 2). In addition, the licensee did not take steps to minimize the effects of noise on data quality (e.g., use of high frequency plus-point coil instead of midrange plus-point coil) or establish data quality acceptance criteria.

The industry has developed draft guidelines for data quality. When finalized, these guidelines will be incorporated into Revision 6 of the Electric Power Research Institute (EPRI) Steam Generator Examination Guidelines.

Issue 4: Non-destructive examination (NDE) qualification programs that include tube samples with flaws that truly represent flaws in the field.

Issue 4 derives from the NRC staff’s review of actions taken by the IP-2 licensee to justify restart with the existing steam generators following the February 15, 2000 tube failure event, as documented in the TER dated October 11, 2000.

The EPRI Steam Generator Examination Guidelines Revision 5 state that flaws in qualification data sets should produce signals similar to those observed in the field in terms of signal characteristics, signal amplitude, and signal-to-noise level. In practice, however, the technique qualifications for various stress corrosion cracking applications continue to rely on data sets consisting of specimens notched by electro-discharge machining (EDM). For example, technique qualification data sets for PWSCC and ODSCC in small radius U-bends consist largely of EDM notched specimens. EDM notched specimens produce much larger amplitude signals and have better signal-to-noise levels than comparably sized cracks. NDE detection and sizing performance for actual cracks will not be as good as that observed for EDM notches. Technique qualifications based on EDM notched specimens do not provide an adequate basis for evaluating the technique capability parameters (probability of detection (POD), sizing accuracy).

Chemical methods to induce stress corrosion cracks can potentially produce cracks which exhibit higher amplitude signal responses than comparably sized cracks in the field. Data from pulled tube specimens can be useful for demonstrating that the fabricated flaws and comparably sized cracks in the field produce signals of similar amplitude.

Issue 5: Site-specific qualifications of generically qualified techniques ensuring an application is consistent with site-specific conditions and that appropriate NDE performance capabilities are considered in operational assessments (e.g., POD of flaws and flaw size measurement error).

Issue 5 derives from the NRC staff's review of actions taken by the IP-2 licensee to justify restart with the existing steam generators following the February 15, 2000 failure event, as documented in the TER dated October 11, 2000.

The EPRI Steam Generator Examination Guidelines state that site-specific qualification of NDE techniques is necessary to ensure that the technique performance capabilities (POD and sizing accuracy) obtained from the generic technique qualification are applicable to site-specific conditions. This is accomplished by a documented review of a qualified technique's tubing essential variables (e.g., denting, deposits, tube geometry changes, and signal characteristics) to ensure the application is consistent with site-specific steam generator conditions. The guidelines state that the review shall establish that tubing essential variables of the flawed tubes in the generic qualification data set are similar in terms of voltage and signal-to-noise ratio to those expected from actual steam generator signals.

The IP-2 licensee was unable to demonstrate to the staff's satisfaction that the NDE technique performance capabilities assumed for the small radius U-bends during the post-failure event inspections performed in 2000 were applicable for IP-2 conditions. The assumed technique performance capabilities were important input parameters in the licensee's operational assessment intended to support plant restart. Difficulties were encountered in the review because the site-specific qualification process described in the EPRI guidelines was not performed rigorously.

After the tube failure at IP-2 on February 15, 2000, the licensee used a high frequency plus-point probe for detecting PWSCC in the small radius U-bends and a midrange plus-point probe for sizing the flaws. Each of these probes were Appendix H qualified in accordance with the EPRI guidelines for detection of small radius U-bend cracks. However, the qualification flaw data set consisted largely of EDM notches rather than actual cracks. Therefore, as previously discussed, the technique capability parameters derived from this generic qualification were not representative of IP-2.

Under these circumstances, the EPRI guidelines state that a technique qualification meeting the statistical requirements of Appendix H should be performed on a data set having tubing essential variables similar to the site-specific conditions. The guidelines further state that this may warrant pulling one or more tubes. An alternative approach using in situ test results did not apply to IP-2 since the IP-2 tubing was in an unacceptable condition at the conclusion of the previous cycle.

Instead of doing such a technique qualification, the IP-2 licensee used NDE performance capability parameter values (POD, flaw size measurement errors) in its operational assessment that were based on a performance demonstration with the midrange plus point probe on a data set consisting of actual PWSCC flaws in simulated dents at egg crate support intersections. However, the licensee was unable to demonstrate to the staff's satisfaction that the tubing essential variables of the performance demonstration data set were representative of the condition of the U-bends at IP-2. A major consideration for the staff was that the licensee was unable to demonstrate quantitatively to the staff's satisfaction that noise levels in the performance demonstration data set were comparable to noise levels for the IP-2 small radius U-bends.

Issue 6: Consideration of flaw size measurement error when applying the threshold screening criteria for selection of in situ pressure test results.

Issue 6 derives from phone conversations in November 1999 between the NRC staff and the ANO-2 licensee concerning the licensee's plans for in situ pressure testing during the November 1999 mid cycle SG inspection outage.

At ANO-2 six tubes were found to exceed the screening criteria for in situ pressure testing. The licensee initially determined that four of the six tubes did not need to be tested since the NDE measured size of the respective flaws were bounded by the size of flaws pressure tested in situ during a previous inspection outage. After discussions with the NRC staff, the licensee decided to test these tubes. The staff noted that the screening criteria, in accordance with the EPRI guidelines, are intended to account for NDE test flaw size measurement uncertainty.

Testing on one of the four tubes was terminated at a pressure below the 3 delta P criterion when leakage through the flaw exceeded the capacity of the system. After reviewing the circumstances of the test, the staff concluded that the tube was about to burst when the test was terminated. This conclusion is discussed under issue 7 and in more detail in the NRC letter dated May 2, 2000 (Accession No. ML003710343). This experience underscores the importance of allowing for flaw size measurement errors (in accordance with the EPRI guidelines) when selecting tubes for in situ pressure testing.

Issue 7: Rigorous analyses of the results of in situ pressure tests that are terminated when leakage exceeds the capacity of the test system.

Issue 7 derives from the staff's review of in situ pressure test results for a tube tested during the November 1997 mid cycle inspection outage at ANO-2. The staff's review was documented in the letter dated May 2, 2000.

At ANO-2 in situ pressure testing was conducted for a tube identified during the November 1999 inspection as having an axial ODSCC flaw. This test involved a pressurization rate of about 1000 psi/minute with periodic hold points. Initial low level leakage was observed at a test pressure of about 3800 psi. This leakage increased to about 1 gpm at a test pressure of 4025 psi and to 3.7 gpm at 4147 psi, whereupon the leakage exceeded the capacity of the test system and the pressure dropped suddenly to 600 psi. The maximum test pressure reached during the test was less than the applicable structural performance criterion of 4369 psi, which corresponds to three times normal operating pressure adjusted to room temperature conditions.

The licensee did not repeat the test with a bladder to determine whether the burst pressure was actually higher than the maximum pressure reached during the test. Instead, on the basis of an assessment of the pre- and post-test eddy current flaw size measurements and in situ test results, the licensee concluded that the test was terminated when ligaments of material in the crack tore and that the actual burst pressure of the tube was at least 500 psi higher than the maximum pressure reached during the test. Using this finding as a benchmark, the licensees did an operational assessment to demonstrate that ANO-2 could be operated without additional SG inspections until the next scheduled refueling outage in September 2000 and could maintain a margin against tube rupture consistent with the plant's licensing basis.

The staff reviewed the licensee's assessment and concluded that the licensee did not have an adequate basis for concluding that the burst pressure was higher than the maximum pressure reached during the test. The staff, therefore, concluded that the licensee had not satisfactorily demonstrated that the applicable performance criteria would be maintained until the scheduled September 2000 refueling outage. The staff's evaluation was documented in the letter dated May 2, 2000.

The EPRI in situ pressure test guidelines provide guidance for assessing burst pressure when the test is terminated because of excessive leakage. The preferred approach is to retest with a bladder installed. Where this is not possible, the guidelines suggest that margin against burst may be verified via visual or eddy current testing (ECT) examination, or by extrapolation of leakage data obtained during the test. There is little specific guidance on how to use visual or NDE results for this purpose. Licensees need to be aware that using the leakage extrapolation methods suggested in the EPRI guidelines can lead to nonconservative assessment results because leak rates for actual cracks can vary by orders of magnitude from the rates indicated by predictive models.

Issue 8: Laboratory and in situ pressure test procedures utilizing pressurization rates that do not influence burst pressure results.

This issue derives from laboratory burst and leak tests performed to support the ANO-2 licensee's assessment of the in situ pressure test results discussed under issue 7. The laboratory test results and their implications have been the subject of public meetings with industry representatives on July 6 and September 28, 2000. The accession number for the July 6, 2000 meeting summary is ML003763234, and for the September 28, 2000 meeting it is ML003760794.

The ANO-2 licensee fabricated a number of EDM notches in laboratory tube specimens in an attempt to simulate the actual flaw geometry in the tube which was the subject of the in-situ pressure test discussed under issue 7. How closely these notches approximated the actual crack is under investigation by the industry and not relevant to this discussion. The laboratory tube samples were leak-tested and burst-tested. Pressurization rates ranged from essentially quasi-static to 2000 psi/second. The tests showed that burst pressure was strongly affected by the pressurization rates used during the tests. This was an unexpected finding in view of prior evidence from laboratories around the world.

The industry is investigating the causes of this apparent pressurization rate effect and any generic implications for pressure test procedures and existing burst pressure databases. Industry representatives discussed the status of this effort with the NRC staff during the July 6 and September 28, 2000, public meetings. They said their preliminary finding was that the pressurization rate effect may be limited to planar, part-through-wall axial flaws with maximum depths greater than 90% through-wall. They also said that changes to industry pressure test procedures may be needed to ensure that burst pressure data is not influenced by pressurization rate. They said that interim industry guidance for in situ pressure testing will be issued to the industry in October 2000. This guidance is expected to recommend limiting pressurization rates to no more than 200 psi/second, with periodic pressure hold points of 2-minute duration. The pressure hold points include main steam line break pressure and three times normal operating pressure.

Issue 9: Use of a “fractional flaw” method or other similar methods for determining a beginning-of-cycle flaw distribution may lead to nonconservative results when used in conjunction with a POD parameter that varies as a function of flaw size or voltage.

Issue 9 derives from the staff’s review (documented in the staff’s TER dated October 11, 2000) of the licensee’s operational assessment to support IP-2 restart after the SG tube failure event on February 15, 2000.

The fractional flaw method is based on the assumption that for each flaw found by inspection, there are flaws of the same size which were not detected by the inspection. For each detected flaw of a given size, the number of undetected flaws of that size is assumed equal to 1/POD - 1. This methodology has been approved by the staff for implementing voltage-based alternate repair criteria in accordance with NRC Generic Letter 95-05 and for implementing an alternate repair criterion for PWSCC at dented tube support plates. For these applications, licensees are currently assuming a constant POD of 0.6.

The fractional flaw method was recently used in an operational assessment of small radius U-bend PWSCC at IP-2. However, the operational assessment used a POD parameter that varied with crack size. The effect of the variable POD function in conjunction with the fractional flaw method make the results of the analysis insensitive to the size of the indications found by inspection. The staff considered this finding unrealistic. The TER contains more details on this issue at IP-2.

Issue 10: Benchmarking operational assessment methodologies against actual operating experience to ensure realistic results.

Issue 10 derives from the staff's review (documented in the staff's safety evaluation dated July 21, 2000) of the ANO-2 licensee's operational assessment in support of its risk-informed application to change the plant licensing basis to permit operation until the September 2000 inspection.

More and bigger flaws were found during the November 1999 mid cycle inspection than were predicted in the previous operational assessment. As discussed under issue 7, one tube apparently failed to meet the 3 delta P criterion. This made the POD and growth rate assumptions used in the licensee's assessment seem dubious, and the staff questioned the use of the same approach in the operational assessment supporting operation until September 2000.

The licensee responded that the November 1999 bobbin coil inspection was more sensitive than the previous inspection and thus had a higher POD. In both inspections a bobbin coil inspection was followed by a rotating pancake coil (RPC) inspection at locations where the bobbin coil identified possible indications. The licensee stated that a different calibration standard had been used during the November 1999 inspection resulting in an improved average flaw signal voltage amplitude and thus a more sensitive inspection. The licensee cited a number of lesser factors also contributing to a more sensitive inspection.

The staff reviewed independently the eddy current data and concluded that although signal amplitude had in fact increased, noise levels had also increased, so that there was no improvement in the signal-to-noise ratio. Based on this and the staff's review of the other factors cited by the licensee, the staff concluded that the inspection in November 1999 was no more sensitive than earlier inspections.

In summary, the licensee was unable to convincingly benchmark its operational assessment methodology against actual operating experience. This created doubt about the POD and growth rate parameters assumed in the assessment and thus about the tube integrity margins projected to exist at the scheduled September 2000 refueling outage. Accordingly, the staff was unable to approve the licensee's risk-informed application to change the plant licensing basis to permit operation to the September 2000 inspection.

BACKFIT DISCUSSION

This RIS does not require any action or written response. Therefore, the staff did not perform a backfit analysis.

FEDERAL REGISTER NOTIFICATION

A notice of opportunity for public comment on this RIS was not published in the *Federal Register* because it is informational, and the public was afforded numerous opportunities to comment as the IP-2 and ANO-2 matters were being studied. Furthermore, this RIS requires no action or written response on the part of an addressee.

PAPERWORK REDUCTION ACT STATEMENT

This RIS does not request any information collection.

If there are any questions concerning this RIS, please call the contact listed below.

/RA/
David B. Matthews, Director
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Technical Contact: Emmett L. Murphy, NRR
(301) 415-2710
E-mail: elm@nrc.gov

Attachment: List of Recently Issued NRC Regulatory Issue Summaries

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DOCUMENT NAME: G:\REXB\ejb1\irs_0010.wpd
SUBJECT: Enclosed are the following IRS reports IRS_0010
ORIGINATOR: E. Benner
SECRETARY: Violet Bowden
DATE: December 7, 2000

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ADAMS ACCESSION #:ML003775171 **TEMPLATE #: NRR-106**

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