August 17, 2004

The Honorable Edward J. Markey
United States House of Representatives
Washington, D.C. 20515

Dear Congressman Markey:

I am responding on behalf of the U.S. Nuclear Regulatory Commission (NRC) to your letter dated May 28, 2004. Your letter expressed concerns that the NRC may not be doing enough to ensure that the steam generator tube integrity is maintained and requested responses to a number of specific questions.

The NRC staff has completed an assessment that concluded that current licensee programs for addressing steam generator issues are adequate and provide confidence that steam generator tube integrity will be maintained. This assessment is contained in SECY-03-0080, "Steam Generator Tube Integrity (SGTI) - Plans for Revising the Associated Regulatory Framework," dated May 16, 2003, which is available in the NRC Agencywide Documents Access and Management System under the accession number ML023540491.

As stated in SECY-03-0080, the NRC staff determined that the existing regulations provide an adequate regulatory basis for dealing with steam generator issues to ensure public health and safety. The staff also concluded that the NRC should take actions that would result in pressurized-water reactor (PWR) licensees upgrading the steam generator tube surveillance requirements in the technical specifications (TS). Due to the changing nature of the degradation being observed in the steam generators, the need to upgrade the steam generator tube surveillance requirements became more important. Under an industry initiative program, the PWR licensees have implemented the improved surveillance requirements developed by the industry. This initiative has resulted in improved program quality and a consistent, effective approach for maintaining steam generator tube integrity. The NRC staff has been working to review the adequacy of the improved steam generator surveillance requirements.

Additionally, improvements have been made to the NRC's regulatory oversight program, including a much improved procedure for inspecting the licensee's inservice examination of steam generator tubes, and more direct involvement in following the tube examination activities for licensees with degraded or potentially degraded tubes. Continued research has deepened the staff's understanding of steam generator behavior under normal and accident conditions and will provide improvements in the risk assessment methods in this area.

In your letter, you indicated that the NRC communicated the finding of cracks in the steam generator tubes at Seabrook in an Information Notice, but that the Information Notice did not require any specific action or impose any requirement. In this particular case, it is important to note that the cracks did not violate any applicable regulatory criteria, did not impact public safety, and were found during a routine inservice inspection. Further inspection results from

the most recent outage were consistent with the expected results from the licensee's steam generator performance assessments, and all applicable regulatory criteria continued to be met. As a result, the NRC concluded that an Information Notice was the appropriate regulatory vehicle to communicate this information to the industry. Although not explicitly required in the Information Notice, the NRC expects licensees to evaluate the information provided in an Information Notice for applicability to their plant and take the necessary corrective actions as required by Criterion XVI of the Appendix B to the Title 10 of the *Code of Federal Regulations*, Part 50. If the severity of an issue warrants action on a more urgent basis, other regulatory tools are used to require licensee action. In addition, although certain generic communications do not require action, the NRC periodically verifies (through a biennial inspection) that licensees' programs for addressing NRC's generic communications are adequate. Also, while doing inspections in response to identified deficiencies or weaknesses in a licensee's program or performance, the NRC may review the licensees' followup to applicable generic communications

The staff has prepared responses to your specific questions, which are enclosed. If you have any other questions, please contact me.

Sincerely,

/RA/

Nils J. Diaz

Enclosure: Questions and Answers

Questions and Answers

Question:

1. During November 2000, after the Indian Point 2 steam generator leakage event, the Nuclear Regulatory Commission (NRC) developed a Steam Generator Action Plan to consolidate activities related to steam generators. For each year since 2000, please describe all the activities of the Plan:

   a) How many full time employees were/are devoted to working on the Action Plan?

   b) What milestones were met? What milestone deadlines were missed?

   c) What milestones for the next five years exist?

   d) How is information regarding the activities of the Action Plan conveyed to the pressurized-water reactors (PWR) licensees?

   e) Have the activities of the Action Plan resulted in any recommendations? If so, please list each recommendation, as well as whether the recommendation has been communicated to all PWR licensees.

   f) Have any of the recommendations of the Action Plan resulted in any changes in NRC regulations or otherwise have been made mandatory? If so, please describe them.

Response:

After the February 2000 Indian Point 2 (IP2) steam generator tube leak event, the Steam Generator Action Plan (the plan), consisting of the IP2 lessons-learned items, was published in an internal NRC memorandum dated November 16, 2000 (Reference 1). On May 11, 2001, the plan was revised to incorporate the recommendations from the Advisory Committee on Reactor Safeguards's (ACRS's) review of a steam generator differing professional opinion (DPO) (Reference 2). These memoranda and the plan are available on the NRC's public Web site (http://www.nrc.gov/reactors/operating/ops-experience/steam-generator-tube.html). The NRC activities related to the plan are described in the plan with target dates for completion. Completed milestones are shown with the completion date and a reference to the completion document by its Agencywide Documents Access and Management System (ADAMS) accession number. There are three groups of activities. The first group of activities, consisting of 1.x numbers, is related to the steam generator inspection, oversight, and licensing program. Some of these activities resulted from the NRC's IP2 lessons-learned review team (LLT) recommendations. The second group, with 2.x numbers, consists of activities that also resulted from the IP2 LLT recommendations, but are not specific to steam generators. The third group, the 3.x milestones, describes the enhanced steam generator research activities in response to the ACRS recommendations, following the Committee's review of a DPO (documented in NUREG-1740, "Voltage-Based Alternative Repair Criteria," dated February 2001). These research activities include developing a better understanding of the steam

generator inservice inspection and nondestructive evaluation methods as well as validating and improving the current methods for maintaining tube integrity. Research activities also include thermal hydraulic and primary system component response (including the steam generators) during design-basis and severe accidents and integrating the improved understanding into a probabilistic risk assessment methodology to identify the risk significant configurations and accident conditions.

a) The NRC has spent approximately 11.3 full-time equivalent (FTE) on the plan activities since November 2000 (the FTE breakdown per year being 0.4 in 2000, 4.2 in 2001, 3.1 in 2002, 1.4 in 2003 and 2.2 in 2004 to date). While the plan is an integral part of NRC's steam generator program, it does not reflect all the steam generator work (e.g., the NRC regional baseline steam generator inspections).

b) Most of the plan actions from the IP2 lessons learned were completed on or close to the target dates. In some cases, delays resulted from a need to expand the scope of the activity. One exception is the NRC staff activities related to revising the regulatory framework that began under an industry initiative program before the February 2000 IP2 steam generator tube leak event. The purpose of this effort is to revise the steam generator tube inspection and repair requirements contained in the technical specifications (TS). The review of the industry initiative was delayed as a result of regulatory, policy, and technical issues. This included delaying staff review to incorporate the IP2 lessons learned, resolution of the policy issue related to the regulatory process to be used by individual licensees in implementing the industry initiative, and resolution of various emergent technical issues. The NRC is currently reviewing an application to revise the TS. Once this application is approved, the rest of the PWR licensees are expected to follow with similar applications to revise their TS.

c) The milestones for the next five years relate to revising the regulatory framework described above (milestones 1.14, 1.16, and 1.17) and the research activities that resulted from the ACRS recommendations on the steam generator DPO (the 3.x milestones in the plan). The research activities did not result from NRC's IP2 LLT recommendations.

d) The plan milestones are implemented through established NRC programs and processes. Communication with the licensees is assessed and appropriate methods are used. During February 2001, a steam generator workshop was held with the PWR licensees and other stakeholders to discuss the new milestones added to the plan. The plan and the closeout documents, including the inspection procedures, are placed on NRC's public Web site, generic communications are issued to the licensees containing information and recommendations, and public meetings are held with the licensees to resolve technical issues. For example, a risk-informed significance determination process (SDP) for steam generator tube integrity was developed under the NRC's reactor oversight program with PWR licensees' participation through periodic public meetings during each phase of the development. Public meetings are held with the PWR licensees to resolve technical issues related to

the development of a steam generator regulatory framework and the TS under an industry initiative. This initiative has already improved licensees' tube integrity program. Additionally, research developments are communicated to the licensees and other stakeholders through annual meetings on reactor safety, the nuclear safety research conference, and publicly available topical reports.

e) Numerous recommendations to improve the NRC inspection, oversight, and license amendment processes are contained in the closeout documents associated with the activities of the plan. These recommendations address both NRC and licensee activities and are listed in the attached Table (Attachment). The closeout documents are referenced in the plan posted on the NRC public Web site. Availability of these documents on the NRC public Web site and in ADAMS informs the licensees and stakeholders. In addition, as noted in the Table, the staff has used generic communications and held public meetings and workshops to inform the licensees and stakeholders about recommendations.

f) Recommendations of the plan did not require or result in any changes to NRC regulations. As stated in SECY-03-0080 (Reference 3), the NRC staff determined that the existing regulations in 10 CFR Part 50 provide an adequate regulatory basis to ensure protection of public health and safety with respect to steam generator operation. Nonetheless, ongoing work to revise the steam generator tube surveillance requirements in the TS to ensure that the condition of the tubes continues to remain adequate for the period of time between inspections will result in mandatory requirements. This will make the TS more reflective of the knowledge gained from the years of operation, including new information on degradation mechanisms, better understanding of steam generator material behavior, and improved technology and assessment methods.

Question:

2. My understanding is that not all steam generator tube inspections are conducted using the best available technologies, and that some of the technologies used have only 70 percent detection rate. Why does not the NRC require the best available technologies for all of these inspections?

Response:

The fundamental purpose of steam generator tube inspections is to ensure the integrity of the tubes. The regulatory requirements pertaining to steam generator tube inspections do not specify a particular technology (e.g., ultrasonic, eddy current, radiography), nor do they specify a particular method (e.g., probe) for these inspections. The choice of technology and method is left to the plant owner. However, the technology and methods chosen must have the objective and capability of detecting flaws of any type that may be present along the length of the tube required to be inspected and that may meet or exceed the applicable tube repair criteria. The NRC staff is in the final stages of developing a generic letter which reiterates the NRC position on this issue.

As you indicate, not all flaws are detected during these inspections. The ability to detect a flaw depends on many factors, including the type of inspection probe, noise (including interfering signals), orientation, and the size of the flaw. Smaller flaws are detected less frequently than the deeper flaws. In general, flaws that could challenge steam generator tube integrity are detected with almost 100 percent effectiveness. The NRC has evaluated, and continues to evaluate, the capabilities of inspection techniques used by plant operators for detecting flaws in steam generator tubes. For example, the NRC issued NUREG/CR-6791, "Eddy Current Reliability Results From the Steam Generator Mockup Analysis Round-Robin," which provides an assessment of the reliability of steam generator tube inspections. These results and operating experience indicate that techniques currently in use to inspect the steam generator tubes are effective at ensuring steam generator tube integrity.

During inspections, licensees monitor the condition of their steam generator tubes to ensure the tube condition is consistent with their expectations. If a plant cannot ensure tube integrity for the period of time between inspections with the inspection techniques it is using (for instance, if the inspection data for a certain technique was of inadequate quality), licensees are required to take corrective action, which may include using the best available technology or performing more frequent inspections.

Question:

3. How many steam generator tubes currently in use are made of Alloy 600 rather than Alloy 690? Please provide a list of all PWRs, indicating for each whether (and if so, when, and with which alloy) the steam generators have been replaced. If some PWR licensees have chosen to replace their steam generator tubes with Alloy 600, please explain why, since Alloy 690 has been found to be less likely to degrade.

Response:

Two of the most important factors affecting tube vulnerability to degradation are the tube material and the tube heat treatment. The two types of tube material being used in the U.S. are Alloy 600 and Alloy 690. The two types of heat treatment applied to these materials for improving their mechanical and corrosion properties are mill annealing (MA) and thermal treatment (TT). In the U.S., steam generator tubes are MA Alloy 600, TT Alloy 600, or TT Alloy 690. At a given plant, all of the tubes are usually fabricated from the same material and with the same heat treatment.

As MA Alloy 600 steam generator tubes began exhibiting degradation in the early 1970s, the industry pursued improvements in the design of future steam generators to reduce the likelihood of corrosion. In the late 1970s, Alloy 600 tubes were subjected to a high-temperature TT to improve the tubes' resistance to corrosion. This TT process was first used on tubes installed in replacement steam generators put into service in the early 1980s. In the 1980s, no significant degradation was observed in steam generator tubes made from TT Alloy 600. Nonetheless, plants began using TT Alloy 690 steam generator tubes in their replacement steam generators in 1989 since it was considered more corrosion resistant than TT Alloy 600. Degradation has recently been observed in

TT Alloy 600 steam generator tubes. The number of tubes affected is small and the severity of the degradation is minor.

Of the 69 operating pressurized water reactors, 24 plants have steam generators with MA Alloy 600 tubing, 17 have TT Alloy 600 tubing, and 28 have TT Alloy 690 tubing. In the operating PWRs, there are approximately 440,000 MA Alloy 600 steam generator tubes and 280,000 TT Alloy 600 steam generator tubes. The NRC staff estimates that approximately 10 percent of the MA Alloy 600 steam generator tubes have been removed from service by plugging. By contrast, approximately 0.6 percent of the TT Alloy 600 steam generator tubes have been removed from service by plugging. The operating experience with TT Alloy 600 is documented in NUREG-1771, "U.S. Operating Experience With TT Alloy 600 Steam Generator Tubes." There are approximately a half million TT Alloy 690 steam generator tubes in the United States. No corrosion-related degradation has been identified in the TT Alloy 690 tubes.

The table on the next page lists each operating PWR, the date of steam generator replacement (as applicable), and the tube material used in the original and replacement steam generators.

A review of the following list indicates that up until 1989, all replacement steam generators used TT Alloy 600 tubes (since this was considered the material of choice at that time). As additional data on the corrosion resistance of TT Alloy 690 became available, plants elected to use TT Alloy 690 in their replacement steam generators.

A further review of the list indicates that since 1989, only three plants have used a tube material other than Alloy 690 in their replacement steam generators. These three plants are Palisades, Salem 1, and Indian Point 2 (IP2). In 1991, the Palisades steam generators were replaced with steam generators that had MA Alloy 600 tubes. The replacement steam generators were fabricated for Palisades in the 1970s when MA Alloy 600 was widely used, but were not installed in the plant until 1991. In 1997, the Salem 1 steam generators were replaced with steam generators that had TT Alloy 600 tubes. The Salem 1 licensee had identified many flawed tubes during an inspection. Rather than repair (or plug) these flawed tubes, the licensee purchased replacement steam generators that were fabricated for a nuclear power plant that was never completed and installed them in the plant. In 2000, IP2 steam generators were replaced with steam generators that had TT Alloy 600 tubes. The replacement steam generators were fabricated in the 1980s when TT Alloy 600 was the material of choice and were stored at the plant, but were not installed in the plant until 2000.

Table: Steam Generator Tube Material

Plant	Replaced	Original Tube Material	Replacement Tube Material	SG Replacement Date
Arkansas Nuclear One 1	N	MA-600		
Beaver Valley 1	N	MA-600		
Beaver Valley 2	N	MA-600		
Braidwood 2	N	TT-600		
Byron 2	N	TT-600		
Callaway	N	MA-600		
Catawba 2	N	TT-600		
Comanche Peak 1	N	MA-600		
Comanche Peak 2	N	TT-600		
Crystal River 3	N	MA-600		
Davis Besse	N	MA-600		
Diablo Canyon 1	N	MA-600		
Diablo Canyon 2	N	MA-600		
Fort Calhoun	N	MA-600		
Millstone 3	N	TT-600		
Oconee 3	N	MA-600		
Palo Verde 3	N	MA-600		
Palo Verde 1	N	MA-600		
Prairie Island 2	N	MA-600		
Prairie Island 1	N	MA-600		
Salem 2	N	MA-600		
San Onofre 2	N	MA-600		
San Onofre 3	N	MA-600		
Seabrook 1	N	TT-600		
Sequoyah 2	N	MA-600		
St. Lucie 2	N	MA-600		
Three Mile Island 1	N	MA-600		
Vogtle 2	N	TT-600		
Vogtle 1	N	TT-600		
Waterford 3	N	MA-600		
Watts Bar 1	N	MA-600		
Wolf Creek 1	N	TT-600		
Surry 2	Y	MA-600	TT-600	Sep 1980
Surry 1	Y	MA-600	TT-600	Jul 1981
Turkey Point 3	Y	MA-600	TT-600	Apr 1982
Turkey Point 4	Y	MA-600	TT-600	May 1983
Point Beach 1	Y	MA-600	TT-600	Mar 1984
Robinson 2	Y	MA-600	TT-600	Oct 1984
D.C. Cook 2	Y	MA-600	TT-690	Mar 1989
Indian Point 3	Y	MA-600	TT-690	Jun 1989
Palisades	Y	MA-600	MA-600	Mar 1991
Millstone 2	Y	MA-600	TT-690	Jan 1993
North Anna 1	Y	MA-600	TT-690	Apr 1993
Summer	Y	MA-600	TT-690	Dec 1994
North Anna 2	Y	MA-600	TT-690	May 1995
Ginna	Y	MA-600	TT-690	Jun 1996
Catawba 1	Y	MA-600	TT-690	Sep 1996
Point Beach 2	Y	MA-600	TT-690	Dec 1996
McGuire 1	Y	MA-600	TT-690	May 1997
Salem 1	Y	MA-600	TT-600	Jul 1997
McGuire 2	Y	MA-600	TT-690	Dec 1997
Byron 1	Y	MA-600	TT-690	Jan 1998
St. Lucie 1	Y	MA-600	TT-690	Jan 1998
Braidwood 1	Y	MA-600	TT-690	Nov 1998
Farley 1	Y	MA-600	TT-690	May 2000
South Texas Project 1	Y	MA-600	TT-690	May 2000
Arkansas Nuclear One 2	Y	MA-600	TT-690	Dec 2000
D.C. Cook 1	Y	MA-600	TT-690	Dec 2000
Indian Point 2	Y	MA-600	TT-600	Dec 2000
Farley 2	Y	MA-600	TT-690	May 2001
Harris 1	Y	MA-600	TT-690	Dec 2001
Kewaunee	Y	MA-600	TT-690	Dec 2001
Calvert Cliffs 1	Y	MA-600	TT-690	Apr 2002
South Texas Project 2	Y	MA-600	TT-690	Dec 2002
Calvert Cliffs 2	Y	MA-600	TT-690	May 2003
Sequoyah 1	Y	MA-600	TT-690	May 2003
Palo Verde 2	Y	MA-600	TT-690	Dec 2003
Oconee 1	Y	MA-600	TT-690	Jan 2004
Oconee 2	Y	MA-600	TT-690	Jun 2004

Question:

4. For the past 10 years, please provide a list of all PWRs that have requested deferrals of their required steam generator (tube) inspections. For each such request, please list the dates upon which it was requested, whether or not the request was granted (and if so, on what date) and when the inspection was completed.

Response:

A search of the NRC database provided the following information related to deferral of the required steam generator tube inspection. The "Date Requested" column reflects the date of application for the deferral request. The actual dates when inspection was due can be found in the application.

Plant	Date Requested	Date Approved	Date of Inspection
North Anna 1[1]	July 2, 1993	withdrawn 03/01/94	September 1994
North Anna 2[1]	October 17, 1995	withdrawn 02/19/96	October 1996
Millstone 2	October 24, 1995	withdrawn 02/04/97	June 1997
Indian Point 3	March 14, 1996	06/19/96	May 1997
Cook 1	August 28, 1998	12/30/98	March 2000
Beaver Valley 1	November 11, 1998	04/16/99	June 2000
Indian Point 2	December 7, 1998	06/09/99	June 2000
Crystal River	January 27, 1999	05/05/99	October 1999
Farley 1[2]	April 30, 1999	08/17/99	March 2000
Beaver Valley 2	November 29, 1999	08/04/00	October 2000
Arkansas Nuclear One 2[3]	March 9, 2000	denied 07/21/00	September 2000
Cook 2	September 30, 2000	01/30/01	February 2002
Braidwood 1	February 9, 2001	08/09/01	April 2003
Farley 1	March 4, 2002	09/20/02	planned fall 2004
South Texas 1	June 20, 2002	07/31/02	planned fall 2004
Arkansas Nuclear One 2	November 22, 2002	05/28/03	planned spring 2005
Summer	January 14, 2003	10/29/03	planned spring 2005
Farley 2	February 11, 2003	07/14/03	planned fall 2005
Kewaunee	October 8, 2003	under review	planned spring 2006[4]
Indian Point 2	October 21, 2003	under review	planned June 2006[4]
Harris	December 8, 2003	to be withdrawn	May 2004

Notes for Response to Question 4:

1. The North Anna Units 1 and 2 request was to inspect only one steam generator rather than the two the licensee would normally be required to inspect. The licensee subsequently withdrew the request.

2. The Farley Unit 1 licensee submitted a request to operate for a full cycle without performing a midcycle inspection. The midcycle inspection was required due to suspected degradation of the steam generator tubes. The licensee provided a detailed risk-informed analysis that demonstrated that the likelihood the tubes would not meet the acceptance criteria at the end of the full cycle was acceptably low.

3. Arkansas Nuclear One Unit 2 submitted a request to operate a full cycle without performing a midcycle inspection. The midcycle inspection was necessary due to suspected degradation of steam generator tubes. The request was denied by the NRC staff based on failure to meet acceptance criteria.

4. The date of the inspection is contingent on the NRC staff's approval of the amendment.

Question:

5. For the past 10 years, please provide a list of all instances (including the name of the reactor and the date of the instance) in which the Commission discovered that a PWR licensee was operating with steam generator tubes that are either a) out of compliance with the 40 percent limit on cracking and thinning, b) out of compliance with the 10-15 percent plugging limit or c) out of compliance with regulations mandating that inspections be conducted. Please include complete information on any enforcement actions that may have been taken, penalties that may have been imposed, the date on which the reactor came back into compliance, and what NRC did to ensure that the licensee did take the required corrective measures.

Response:

a,c) When a licensee performs a steam generator inspection, it is recognized that flaws exceeding the tube plugging limit (e.g., 40 percent through-wall) may be detected. This is because a flaw whose depth is less than the plugging limit may be left in service. This flaw may continue to grow such that it exceeds the plugging limit at the time of the next inspection. Flaws that exceed the plugging limit do not necessarily compromise tube integrity since the plugging limit takes into account continued flaw growth until the next inspection, as well as the uncertainty in the measurement of the size of the degradation and the type of degradation affecting the tubes. Therefore, the fact that flaws are found during an inspection that exceed the plugging limit does not necessarily mean that a violation of NRC requirements has occurred.

However, the licensees have occasionally failed to do the appropriate inspection or failed to plug indications that exceeded the plugging limit. These instances were violations of NRC requirements. Based on a search of the NRC database, the following violations were identified:

Plant	Date	Violation	Issue and Corrective Action
Palo Verde	04/2004	green (non-cited violation)	A non-cited violation of corrective action was generated when the licensee failed to correct tube damage from a packing crate screw, resulting in a tube leak. The licensee in-situ tested the tube, with satisfactory results, removed the tube from service, and entered the issue in its corrective action program.
San Onofre	01/2003	green (non-cited violation)	A non-cited violation of Appendix B, Criterion V, resulted when the licensee found in January 2003 that the technicians failed to identify an eddy current indication that required further examination and evaluation during the previous outage in January 2001. Further evaluation in 2003 revealed that the flaw did not exceed the TS repair limit of 44 percent through-wall, and the tube passed the in-situ pressure test. The tube was taken out of service.
Comanche Peak 1	09/2002	white (violation)	The licensee failed to identify a flawed tube and remove it from service before the flaw developed into a leak. Subsequent NRC inspection determined licensee's corrective action to be adequate. See further discussion below.
Indian Point 2	02/2000	red (violation)	Failure to identify tube flaw resulted in leakage. See further discussion below.

St. Lucie	03/1999	non-cited violation	In March 1999, the licensee identified that a tube in steam generator 2B operated during two fuel cycles with a 47 percent through-wall indication. The tube was plugged in November 1998. The licensee determined that deficiencies in the data analysis guidelines and inadequate analyst training were the causes. The corrective actions were completed in March.
St. Lucie	11/1998	non-cited violation	During a November 1998 Unit 2 refueling outage, the licensee determined that three tubes had not received the TS-required inspection during the previous outage due to encoding errors associated with remote positioning equipment. Subsequent inspection revealed that these tubes remained operable during the missed surveillance period. This violation was entered in the licensee's corrective action program.
Arkansas Nuclear One	04/1998	non-cited violation	A non-cited violation of the TS resulted from licensee's identification in April 1998 that tubes, with flaws exceeding the TS limit, had been left in service following the previous refueling outage (1R13). The licensee completed the corrective actions.
Oconee	1998	non-cited violation	Inadequate guidance for the eddy current analysts and the failure to correctly analyze eddy current data resulted in defective steam generator tubes being left in service. Repairs were made during the 1998 outages.
Arkansas Nuclear One	1997	violation	A Severity Level III violation was issued for failure to conduct properly and evaluate the results of a steam generator tube inspection in 1996. See detailed discussion below.

The following addresses escalated enforcement cases listed above:

Since 1994, enforcement actions involving three escalated enforcement cases have been taken that pertained to violations of steam generator tube inspection requirements or other NRC regulations associated with steam generator tube inspections.

- Arkansas Nuclear One - A Severity Level III Notice of Violation was issued on November 18, 1997, for the licensee's failure to conduct properly and evaluate the results of steam generator tube inspections in 1996 (this enforcement action predated implementation of the revised Reactor Oversight Process (ROP) in 2000). No civil penalties were assessed because the licensee was credited for identification of the violations and comprehensive corrective actions, in accordance with the NRC Enforcement Policy. In this case, no tube leakage occurred. The licensee identified errors in 1996 inspections conducted on Unit 1 through destructive examination of removed tubes on April 8, 1997. Unit 1 was operating at the time, and the licensee correctly concluded that their steam generator tubes did not meet TS requirements. The licensee sought and was granted enforcement discretion until a license amendment was granted on May 7, 1997, to authorize operation to the next refueling outage with tubes not verified to be within plugging limits. This licensing action restored compliance for Unit 1. Unit 2 was in an outage at that time, and the licensee was using improved tube inspection techniques. Appropriate tubes were plugged before the plant was returned to service, including some that should have been identified and plugged in the previous outage in 1996. These repairs restored compliance for Unit 2. The NRC conducted inspections concurrently with licensee corrective measures, observing the improved tube inspection techniques. The NRC also verified corrective actions and observed more steam generator tube inspections in March 1998.

- Indian Point 2 - A red[1] SDP finding and associated notice of violation were issued on November 20, 2000, for the licensee's failure to evaluate properly the results of steam generator tube inspections in 1997. That failure led to the steam generator tube failure event that occurred on February 15, 2000. The actual consequences of the event in terms of radiological release were insignificant. No civil penalties were assessed because civil penalties are not normally issued in the ROP for findings evaluated by the SDP. The plant remained shut down for steam generator replacement, which restored compliance. The plant restarted in December 2000. Throughout the extended shutdown and following the plant restart, the NRC conducted many inspections as the plant was in the Multiple/Repetitive Degraded Cornerstone column of the NRC's ROP Action Matrix. The increased inspections continued until August 28, 2002, when the facility was removed from the Multiple/Repetitive Degraded Cornerstone column of the Action Matrix.

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[1]Under the ROP, SDP findings are graded in accordance with escalating significance, i.e., green, white, yellow, and red.

- Comanche Peak - A white SDP finding and associated notice of violation were issued on February 13, 2004, for the licensee's failure to properly evaluate the results of steam generator tube inspections in April 2001. That failure led to the steam generator tube failure event on September 26, 2002, and the resulting plant shutdown that occurred on September 28, 2002. No civil penalties were assessed because civil penalties are not normally issued in the revised ROP for findings evaluated by the SDP. The steam generator was repaired prior to plant restart, restoring compliance. The restart took place on November 18, 2002. The NRC conducted inspections concurrently with the repairs and the licensee's corrective measures prior to restart. In March 2004, the NRC conducted supplemental inspections associated with the white finding, and concluded that the licensee had taken appropriate actions.

b) In the design of a facility, a certain percentage of the steam generator tubes are assumed to be removed from service. This percentage varies from plant to plant. Some plants can only tolerate a small percentage of tubes being removed from service (e.g., 10 percent) without derating the maximum reactor power level, while other plants can tolerate a much larger percentage (e.g., 30 percent). When a steam generator approaches its limit on the percentage of tubes removed from service, a plant operator can consider several courses of action, including reanalyzing the percentage of tubes that can be removed from service, derating the facility (reducing the power level), and replacing the steam generators. To the NRC staff's knowledge, licensees have never operated their steam generators outside these analyzed limits.

Question:

6. How many inspection hours are allocated to steam generators within the NRC's baseline module of the reactor oversight program? For all reactors receiving more than the baseline inspection effort, please list the reactor, the reason for the increased inspection effort, and the results from the inspections.

Response:

The steam generator inspections are conducted in accordance with NRC baseline Inspection Procedure 71111.08, "Inservice Inspection Activities." The steam generator inspection is estimated to take about 30 direct inspection hours per PWR unit, every refueling outage, for plants that have not yet replaced the steam generators. Fewer inspection hours are expended for plants with newer steam generators. It is expected that the actual hours required to complete an individual inspection procedure at a particular plant will vary from the estimate based on the risk-significant issues identified by the inspectors. Direct inspection hours do not include inspector preparation (which includes some review of the licensee's procedures) and documentation of the inspection findings. Additional inspection hours are expended on steam generator replacement activities.

In recent years, several reactors received inspections beyond the baseline inspection program. These inspections are discussed below.

- Following the February 15, 2000 steam generator tube failure event at IP2, the NRC inspection teams observed that the licensee had failed to identify and correct a significant condition adverse to quality involving the presence of primary water stress corrosion cracking (PWSCC) flaws in row 2 steam generator tubes in the small radius, low-row U-bend apex area. Specifically, during the 1997 steam generator eddy current test and secondary side visual examination, the licensee did not adequately account for conditions which adversely affected the detectability of, and increased the susceptibility to, tube flaws. As a result, tubes with PWSCC flaws in their small radius U-bends were left in service following the 1997 inspection until the tube failure on February 15, 2000, while the reactor was at 100-percent power. The NRC put IP2 in the multiple/repetitive degraded cornerstone column of the ROP action matrix, in accordance with the guidance in NRC Inspection Manual Chapter 0305, "Operating Reactor Assessment Program." This action followed the red inspection finding concerning performance issues that led to the February 2000 steam generator tube failure.

  The NRC completed an extensive supplemental team inspection in February 2001 in accordance with the NRC Supplemental Inspection Procedure 95003, “Supplemental Inspection For Repetitive Degraded Cornerstones, Multiple Degraded Cornerstones, Multiple Yellow Inputs, Or One Red Input.” The 95003 inspection focused on performance problems associated with the red inspection finding. Based on the outcome of the 95003 inspection, the NRC concluded that continued heightened oversight of IP2 was necessary until the NRC gained confidence in the ability of the performance improvement program to address the performance issues that underlie the degraded cornerstones. Subsequent NRC followup inspections found that the licensee had adequately addressed all performance issues.

- During November 2001, the NRC completed a special team inspection at Three Mile Island 1 to examine licensee’s activities related to the discovery of a previously plugged tube that had severed. Based on the results of this inspection, the NRC team found no licensee performance issues associated with the severed plugged tube and concluded that the licensee’s corrective actions were appropriate. The NRC issued an Information Notice 2002-02, “Recent Experience with Plugged Steam Generator Tubes,” dated January 8, 2002.

- In April 2002, the NRC conducted a special team inspection at Oconee Nuclear Station to review the licensee’s activities related to a severed plugged tube discovered in Unit 1. The tube was discovered by the licensee’s augmented inspection of previously plugged steam generator tubes following the discovery of a severed tube in Three Mile Island 1 during the fall of 2001. The NRC inspection concluded there were no performance deficiencies.

- In November 2002, the NRC performed a special team inspection at Comanche Peak Steam Electric Station in response to a steam generator tube leak. The NRC findings are included in response to question 5 (Table).

- In February 2003, the NRC conducted a special team inspection at Diablo Canyon Power Plant, following a report by the licensee of unexpected steam generator eddy current test results. The staff determined that the licensee's actions were appropriate. This inspection resulted in the staff issuing an Information Notice 2003-13, "Steam Generator Tube Degradation at Diablo Canyon," dated August 28, 2003.

- During February and March of 2004, the NRC conducted a special inspection at Palo Verde Nuclear Generating Station which reviewed the causes and corrective actions taken by the licensee following a steam generator tube leak. The NRC findings are discussed in response to question 5.

References:

1. NRC Memorandum from B. Sheron and J. Johnson to S. Collins, Steam Generator Action Plan, November 16, 2000 (ML003770259).

2. NRC Memorandum from S. Collins and A. Thadani to W. Travers, Steam Generator Action Plan Revisions to Address Differing Professional Opinion on Steam Generator Tube Integrity, May 11, 2001 (ML011300073).

3. SECY-03-0080, "Steam Generator Tube Integrity (SGTI) - Plans for Revising the Associated Regulatory Framework," dated May 16, 2003 (ML023540491).

List of Recommendations from Steam Generator Action Plan

Item No.	Milestone	Completion Date/ Document	Recommendations for NRC	Recommendations for Licensees
1.1	Issue Regulatory Information Summary on SG Lessons Learned - Regulatory Issues Summary RIS 2000-22 was issued.	11/03/00 (C) ML010820457 RIS 2000-022		Consider relevant operating experience and appropriate diagnostic, corrective, or compensatory measures to ensure tube integrity. Assess the root causes of all degradation mechanisms at a plant and use appropriate diagnostic, corrective, or compensatory measures to ensure tube integrity. Minimize the effects of noise on data quality. Non-destructive examination (NDE) qualification programs should include tube samples with flaws that truly represent flaws in the field. Site-specific qualifications of generically qualified techniques is needed to ensure an application is consistent with site-specific conditions and that appropriate NDE performance capabilities are considered in operational assessments (e.g., probability of detection (POD) of flaws and flaw size measurement error).

ATTACHMENT

1.1 (cont'd)				Flaw size measurement error should be considered when applying the threshold screening criteria for selection of in-situ pressure test results. Rigorous analyses of the results are needed when in-situ pressure tests are terminated with leakage exceeding the capacity of the test system. Laboratory and in situ pressure test procedures must utilize pressurization rates that do not influence burst pressure results. 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. Benchmark operational assessment methodologies against actual operating experience to ensure realistic results.

1.2	Discuss steam generator action plan and Indian Point 2 lessons learned with industry and other external stakeholders	12/20/00 (C) ML010820457 Public meeting summary ML010510389		Establish NDE data quality acceptance criteria. Update Technical Specifications including operational leakage limits and surveillance requirements to reflect current knowledge.
1.5	Staff review of ACRS recommendations on differing professional opinion (DPO) and develop detailed milestones and evaluate impact on other action plan milestones.	05/11/01 (C) ML011720125 ML013300073	While pursuing resolution to alternate repair criteria (ARC) outside the scope of Generic Letter 95-05, any newly proposed ARC will be evaluated with due consideration of the issues raised by the ACRS, in response to the DPO, to ensure that unacceptable risks are not introduced. The Office of Nuclear Reactor Regulation (NRR) and the Office of Nuclear Regulatory Research (RES) staff will continue to work together and interact with other stakeholders as needed (e.g., the Regions, Nuclear Energy Institute (NEI), Electric Power Research Institute (EPRI), the public) on the issues addressed in the steam generator action plan. Steam generator action plan milestones 3.1 through 3.10 were developed.	Provide the data and analyses necessary to show that risk would remain at acceptable levels under any new ARC proposal. Referenced documents may be accessed via the NRC public Web site.
1.6	Determine Generic Safety Issue (GSI)-163 resolution strategy and revise steam generator action plan milestones, as appropriate.	05/11/01 (C)	Steam generator action plan milestone 3.11 was developed which recommends development of detailed milestones after completion of action plan milestones 3.1 through 3.5 and 3.7 through 3.9.	

1.7	Determine need to incorporate new steam generator performance indicators into Reactor Oversight Process (ROP).	01/24/01 (C) ML010820457	While the ROP needs to support the appropriate identification of and reaction to potential licensee performance problems related to maintaining steam generator tube integrity, performance indicators were determined not to be the appropriate vehicle for meeting these needs. However, applicable revisions to the baseline inspection program for licensee steam generator tube inspections and an associated Significance Determination Process (SDP) have been made to better identify and assess steam generator issues. (ML041340207 and SGAP milestone 1.11 below - ML020730318/ML041340004).	None. Availability of the closeout document on the NRC's public Web site and in ADAMS informs the licensee about the changes in NRC's program and processes.
1.9	Review NRC inspection program and, if necessary, revise guidance to inspectors on overseeing facilities with known steam generator tube leakage.	03/30/01 (C) ML010920112 Inspection Manual Chapter (IMC) Part 9900	Provided guidance to inspectors on overseeing PWRs with known primary-to-secondary leakage in order to allow for enhanced opportunities for corrective action when leakage occurs.	
1.10	Reassess the NRC treatment of licensee steam generator inspection results summary reports and conference calls during outages. Evaluate need for review guidance.	04/30/01 (C) ML011220621 ML013020093	Formally review and document steam generator inspection results. Developed steam generator review guidance for consistency in reviews, and developed process for conducting outage conference calls.	

1.11	a.	Review and revise the baseline inspection program.	04/30/01 (C) ML011210293	Increased the frequency and level of effort for NRC oversight of licensee steam generator tube inspections. Expanded the guidance provided to NRC inspectors who oversee licensee steam generator tube inspections.	
1.11	b.2)	Develop and issue draft revision of risk-informed SDP.	02/21/02 (C) ML020730318	Draft SDP was issued for internal and external comment, comments were resolved and the final SDP was issued on May 6, 2004 for use in the inspection program. (ML041340004)	None. The SDP is an internal NRC staff process to categorize the significance of NRC inspection findings. The SDP and information related to the ROP are available on the NRC public Web site and in ADAMS. Additionally, several public meetings were held with the licensees and other stakeholders on the SDP.

1.11	c) Review and revise the training program for inspectors. c.1) Provide training material to Regions. c.2) Formal training to inspectors.	ML020560366 ML012970361 10/11/01 (C) 02/01/02 (C)	The IMC 1245, “Inspector Qualification,” Working Group is developing advanced technical training for inspectors in the areas of metallurgy, inservice inspections, mechanical, and electrical engineering. This advanced training would be completed after, and supplement, the Full Inspector Qualification requirements. c.1) Training materials were distributed to Regions in October of 2001. This material consisted of industry standards pertinent to the SG inspections. (ML020560366) c.2) The Division of Engineering conducted a 3-day classroom training session for regional inspectors during January 22-24, 2002. (ML020560366)	
1.12	Provide formal written guidance for technical reviewers to utilize when performing steam generator tube integrity license amendment reviews.	04/30/01 (C) ML011220621	Technical reviewers to utilize the formal written guidance when performing steam generator tube integrity license amendment reviews.	
1.18	Staff briefs the Commission on regulatory framework.	5/29/03 (C) ML031610897	Complete work associated with the regulatory framework revision and implementation of technical specification changes.	PWR licensees participate in the initiative with NEI lead and keep informed via public meetings, NRC public Web page and through their membership to NEI.

1.19	Issue generic communication related to steam generator operating experience and status of steam generator issues.	10/31/01 (C) ML020230299 Information Notice 2001-16	Continue to assess operating experience and events and issue periodic communication as appropriate and consistent with the generic communication process. Licensees are informed via the generic communication process.	Effectively monitor the tubes and their support structures, regardless of steam generator design or materials, to ensure tube structural and leakage integrity are maintained. Perform comprehensive inspections subsequent to chemical and/or mechanical cleaning of steam generators to ensure that degradation induced (or exacerbated) by secondary-side activities is detected to prevent a loss of tube integrity. Monitor secondary side structures/components to prevent degradation that may result in tube damage through loss of support to the tube and/or through mechanical damage by introduction of loose material into the steam generator. Perform comprehensive inspections of steam generator tubes throughout the lifetime of the steam generator regardless of the tube material.
1.20	Staff issues a Commission Paper on regulatory framework.	05/16/03 (C) SECY-03-0080 ML023540491	Evaluate the lead plant submittal using the license amendment process and evaluate the generic submittal using the Technical Specification Task Force process.	

2.2	Establish NRC web site for Steam Generator Action Plan.	01/16/01 (C) ML010820457	Periodically update the steam generator action plan on the NRC public Web site, and include relevant publically-available documents that are added to ADAMS.	
2.3	Review and revise, as appropriate, the policy for project manager involvement with the morning call between resident inspectors and the region.	03/23/01 (C) ML011020026	The Project Manager and Resident Inspector should exchange phone calls at least several times each week in order to discuss plant status and other routine matters.	

2.4	Review program requirements for routine communications between the resident inspectors and local officials based on public interest. Based on weighing current resident inspector responsibilities (e.g., inspection requirements, following up on plant events) against this review, revise program requirements if needed.	04/03/01 (C) ML010890426	This milestone stems from a comment made by local public officials regarding their lack of routine communications with NRC and with the Indian Point NRC resident inspectors to obtain information about plant status during events. The staff evaluated existing policies regarding press releases with contact information for newly assigned resident inspectors, inspector identification in publicly available inspection reports, plant performance information posted quarterly on NRC Web sites, resident participation with local officials during emergency planning exercises, periodic public meetings for the ROP and other topics of public interest, the NRC's 'open door' policy for State and local officials to meet with NRC staff, and the then new ROP timeliness of routine plant performance assessments. This evaluation resulted in a change to program guidance to encourage region management to consider using resident inspectors when appropriate to improve public communication, on a not-to-interfere basis with their primary duties (ROP Feedback 2515-336).	

2.5	Develop, revise, and implement, as appropriate, a process for the timely dissemination of technical information to inspectors for inclusion in the inspection program.	04/03/01 (C) ML010890426	The Reactor Operations Branch of NRR electronically disseminate technical information (e.g., NRC generic correspondence) to the regions. Each regional office was confirmed to have a process for handling and appropriately disseminating information to inspectors on an as-needed basis, therefore, no specific changes were made as a result of this evaluation. However staff work has began on the development of an Inspector Electronic Support System (IESS) as a Web-based Knowledge Capture/Management/Transfer tool to provide a library of information directly useful to inspectors.	
2.6	Develop communication plan for risk communication and outreach to the public regarding steam generator events.	12/24/02 (C) ML023440202	The NRC staff should follow the communication plan to clearly and accurately convey the actual safety significance of steam generator events, if and when they occur, to the public and other stakeholders.	
2.8	a. Issue OI LIC-101. b. Issue procedure for NRR and RES interactions.	08/31/01 (C) 02/27/02 (C) ML020580484	The NRC staff is required to follow these procedures during preparation of license amendments and interactions with RES.	

3.1	In Order to address ACRS comments on current risk assessments, develop a better understanding of the potential for damage progression of multiple steam generator (SG) tubes due to depressurization of the SG (e.g., during a main steamline break (MSLB) or other type of secondary side design basis accident).	3.1a)-3.1b) 12/31/02 (C) ML023650132 3.1c) 06/30/03 (C) 3.1d)-3.1h) 12/31/02 (C) ML030230822 3.1i) 06/03/03 (C) ML032080002 3.1j) 06/30/04 (C)	The calculated force loads from a MSLB are low and the structural analyses conducted indicated that existing flaws would not grow. Therefore, damage progression during depressurization events need not be considered in risk assessments, and is not an issue.	A non-proprietary NUREG/CR is in preparation.
3.2	Confirm that damage progression via jet cutting is of low enough probability that it can be neglected in accident analyses.	12/311/01 ML021910311	Analytical models were developed and tests of jet impingement under severe accident and MSLB conditions have been conducted showing that jet cutting would not be likely to occur. Based on the results, we have concluded that degradation from jet impingement on adjacent tubes is small or non-existent and can be neglected in accident analysis. ACRS has concurred in this conclusion.	NUREG/CR-6756 and NUREG/CR-6774 have been issued containing the results.

3.6	Address an ACRS report conclusion that improvements can be made over the current use of a constant probability of detection (POD) during eddy current (EC) for flaws in SG tubes.	12/31/01 ML021910311	Research results using the Argonne National Lab (ANL)/NRC steam generator mockup and commercial inspection teams and procedures were used to develop POD curves as a function of crack size and other flaw severity parameters. Therefore, results are available that can be used by the staff to evaluate industry proposals for using POD correlations other than a constant POD.	NUREG/CR-6791 has been issued containing the results.
3.7	Assess the need for better leakage correlations as a function of voltage for 7/8" steam generator tubes.	04/26/03 (C) ML031150674	Continue to assess the leakage correlation as more data are added to the database.	
3.8	Develop a program to monitor the prediction of flaw growth for systematic deviations from expectations.	01/03/02 (C) ML020070081	Continue to review inspection reports and if the projected tube conditions are not conservative when compared to the actual values, request licensees to evaluate the root cause and ensure appropriate corrective actions are taken.	