**From:** Gregory Cranston
**To:** Wayne Schmidt
**Date:** Mon, Jul 17, 2000 4:29 PM
**Subject:** Enforcement Conf. First Draft

Comments?

I'll get rid of the second page number and the italics 'hints' later.

L/14

ITEM # 30

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**SDP/ENFORCEMENT PANEL WORKSHEET**
**7/17/2000**

**EA:**
**Date of Panel:** LATER

**Licensee:** Consolidated Edison Company of New York, Inc.
**Facility/Location:** Buchanan, New York

**Docket No(s).:** 50-247
**License No(s).:**
**Inspection/OI Report No(s):** 2000-010
**Date of Exit Meeting/OI Report Date:** July 18, 2000

**Panel Chairman (SES Sponsor):**
**Responsible Branch Chief/Lead Inspector:** David Lew/Wayne Schmidt
**Enforcement Representative:**
**Other regional attendees:** LATER
**Headquarters attendees:** LATER

1. **Brief Summary of Issues/Potential Violations:**

A Severity Level III, Red finding is proposed. The team concluded that during the 1997 steam generator inspection, Con Edison did not recognize and take corrective actions for significant conditions adverse to quality relating to eddy current data collection and analysis and specific steam generator conditions. These missed opportunities caused significant limitations and uncertainties, resulting in tubes with detectable flaws being left in service. Collectively, these opportunities, along with a new active degradation mechanism, increased the likelihood of tube integrity problems during the subsequent operating cycle.

A. Technical Specification 4.13 .B requires that steam generator tubes with defect depths of greater than 40% through wall are not considered acceptable for continued service and shall be plugged.

Contrary to the above, during the 1997 steam generator eddy current inspections, steam generator tubes that had depths of degradation greater than 40% through wall were considered acceptable for continued service and were not plugged (based on hindsight look at the 1997 eddy current data).

Specifically: (1) as document on Con Edison Condition Report (CR) 2000 - 1939,

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four tubes had U-bend indications with estimated depths greater than 40% through wall (SG21 tube R2C87 - 53%; SG24 R2C5 - 87%, R2C69 - 53% and R2C72 - 75%); (2) as documented in the Con Edison June 2, 2000, Condition Monitoring and Operational Assessment Report (CMOA), Attachment SG-00-05-010, Table 4.5-1, 'Indian Point 2 Sludge Pile ODSCC Growth Rates from Bobbin Coil Analysis', five tubes had sludge pile ODSCC indications with estimated maximum depths greater that 40% through wall (SG22 tubes R34C51, R35C51, R35C52, R34/C54and R33/C49); and, (3) Con Edison did not generate a CR nor submit an LER on this apparent TS violation.

B. 10 CFR 50, Appendix B Criterion XVI - Corrective Actions, requires, in part, that Con Edison promptly identify and take corrective actions for conditions adverse to quality.

Contrary to the above, Con Edison failed to promptly identify and plug six steam generator tubes with identifiable U-bend inside diameter primary water stress corrosion cracking (PWSCC) during the 1997 refueling outage. Con Edison did not adequately evaluate poor quality data (low signal to noise ratios) that was encountered during the eddy current inspections in 1997. Con Edison failed to evaluate the effect on the probability of detection of small radius U-bend tube indications. Consequently, these tubes were left in service after the 1997 refueling outage, eventually leading to the February 15, 2000, steam generator 24 tube row 2, column 5 tube failure.

C. 10 CFR 50, Appendix B Criterion IX - Control of Special Processes, requires, in part, that measures shall be established to assure those special processes, including nondestructive testing, are controlled and accomplished using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements.

Paragraph 4.3 of Specification No. NPE-72217, "Eddy Current Examination of Nuclear Steam Generator Tubes, Indian Point 2," Revision 10 dated December 17, 1996, states, in part, "The examination technique shall be performed using qualified methods that are capable of detecting axial, skew, and circumferential cracking. The techniques used shall be qualified to the EPRI Steam Generator Examination Guidelines, Appendix H."

The EPRI Steam Generator Examination Guidelines, Appendix H, Qualified Technique for Low Radius U-bends (96511Pwscc_ubend.doc), specified a phase rotation setting of 10° for a calibration standard and 40 percent inside diameter through-wall circumferential and axial notches.

Contrary to the above, the Indian Point 2 specific qualification sheet (Sheet IP2-

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97-E, Revision 0,) specified a phase rotation so that probe motion was horizontal and the calibration standard did not include 40 percent through-wall circumferential and axial inside diameter notches. As such, the plus Point probe technique used at Indian Point 2 in 1997 was not calibrated or set-up in accordance with the EPRI Steam Generator Examination Guidelines, Appendix H, qualified U-bend examination technique.

D. Technical Specification 4.13.C.3 requires, in part, the monitoring for significant hourglassing (flow slot closure) of the upper support plate flow slots to ensure the long term integrity of small radius U-bends beyond row 1.

Contrary to the above, Con Edison did not adequately monitor for significant hourglassing of the upper support plate flow slots. Specifically, Con Edison did not have a method to measure the flow slot hourglassing nor a criteria to determine when it was significant, with respect to long term integrity of small radius U-bends beyond row 1. Con Edison did not sufficiently assess eddy current probe restrictions in the upper support plate encountered during the 1997 steam generator inspections, with respect to the potential for flow slot hourglassing. Con Edison did not evaluate the potential for increased apex stresses and PWSCC.

E. CFR 50, Appendix B Criterion XVI - Corrective Actions, requires that Con Edison, determine the cause and take actions to prevent recurrence for significant conditions adverse to quality.

Contrary to the above, Con Edison did not adequately determine the cause for the failure of SG 24 tube R2C5, and as such, corrective actions may not have been taken for a significant condition adverse to quality. Specifically, the root cause analysis did not identify inadequacies in Con Edison's technical oversight and management of the 1997 steam generator inspections. It failed to address the lack of corrective action in response to a new SG degradation mechanism and did not identify the improper set-up of the eddy current probe, and the inadequate inspection and evaluation of the upper support plate denting and/or flow slot hourglassing. Con Edison did not have an accurate method of measuring, nor criteria for determining, when significant hourglassing of the upper tube support plates had taken place. Con Edison did not sufficiently assess eddy current probe restrictions in the upper support plate encountered during the 1997 steam generator inspections, with respect to the potential for flow slot hourglassing. Con Edison did not evaluate the potential for increased apex stresses and PWSCC. As such, no meaningful visual examination of the flow slots was conducted.

F. 10 CFR 50.73 requires that Con Edison submit a licensee event report within thirty day after the discovery of conditions prohibited by plant technical specifications.



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Contrary to the above, as of July 7, 2000, Con Edison did not submit an LER within thirty days after discovery of conditions prohibited by plant Technical Specification 4.13. Technical specification 4.13 requires that steam generator tubes with defects greater than 40% through wall be removed from service prior to returning the unit to operation.

Specifically, (1) on March 3, 2000, Con Edison documented in CR 2000-1936 that four steam generator tubes had U-bend indications greater than 40% through wall. (2) on June 4, 2000, Con Edison documented in their steam generator Condition Monitoring and Operational Assessment (CMOA), Attachment SG-00-05-010, Table 4.5-1, 'Indian Point 2 Sludge Pile ODSCC Growth Rates from Bobbin Coil Analysis', that six tubes had sludge pile ODSCC indications with estimated maximum depths greater that 40% through wall.

2. **Purpose of Panel:**

A Severity Level III, Red finding is proposed.

The Probabilistic Safety Assessment Branch of NRR reviewed the information associated with the Indian Point, Unit 2, Steam Generator Tube Leak of February 15, 2000, and performed a risk assessment (Memo Richard J. Barrett, Chief Probabilistic Risk Assessment Branch, Division of Systems Safety and Analysis, Office of NRR, dated May 4, 2000 to A. Randolph Blough, Director, DRP, Region I). The risk associated with the condition of the SG tubes during Cycle 14 comes from several potential accident sequences:

A. Spontaneous rupture of a tube, not successfully mitigated by plant operators, causing core damage and bypass of the containment by large radioactive releases.
B. Rupture of one or more tubes induced by a steam system depressurization event, not successfully mitigated by plant operators, causing core damage and bypass of the containment by large radioactive releases.
C. Rupture of one or more tubes induced by a reactor system over pressurization event, causing core damage and bypass of the containment by large radioactive releases.
D. A core damage event that occurs with the reactor system at normal operating pressure, inducing tube rupture by increasing tube temperature and/or tube differential pressure, causing bypass of the containment by large radioactive releases.

Of these, the first two increase both the core damage frequency (CDF) and the

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frequency of large radioactive releases bypassing the containment and reaching the environment (assumed to be a "large early release"). The later two sequences are already included in the plant's core damage frequency estimate, but would not normally be included in its large early release frequency (LERF). The induced tube ruptures cause them to make contributions to LERF.

The sum of these tube degradation related risk contributions for Indian Point Unit 2, during Cycle 14 is estimated to be a probability of core damage accident with a large release at approximately 10E-4. This risk occurred mostly during the latter year of the operational cycle. The risk from spontaneous rupture is the dominant contributor to the increases in both the core damage and the large release probabilities. The risk contribution from ruptures induced by steam system depressurizations adds about 10% to these totals, and the risk contribution from other core damage sequences that induce tube failure adds perhaps another 10% to the probability of large release, without increasing the core damage probability.

The Significance Determination Process (SDP) for the New Reactor Oversight Process is based on changes to core damage frequency associated with a condition at a power reactor unit. For conditions that increase the frequency of a LERF the threshold significance determination criteria are reduced by a factor of 10, compared to the criteria used for core damage sequences that do not produce a large, early release. The guidance for core damage sequences involving steam generator tube rupture is to consider them as LERF sequences.

The current guidance for assigning risk significance is contained in a draft NUREG/CR "Basis Document for Large Early Release Frequency (LERF) Significance Determination Process (SDP) - Inspection Findings That May Affect LERF." The guidance is summarized in Table 1 below.

**Table 1** - **Risk Significance Based on LERF and CDF**

| Frequency Range/RY | SDP Based on CDF | SDP Based on LERF |
|---|---|---|
| > 10E-4 | Red | Red |
| < 10E-4 - 10E-5 | Yellow | Red |
| < 10E-5 - 10E-6 | White | Yellow |
| < 10E-6 - 10E-7 | Green | White |
| < 10E-7 | Green | Green |

The conceptual question is how to assign a frequency to an accident initiating event that has happened once as the consequence of a condition that has developed over a period of time. The following discussion is considered sufficiently quantitative to establish the risk input fo determining the "color" of the situation that occurred at Indian Point 2.

Indian Point 2 was returned to service in 1997 in a condition that deteriorated with time to



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the point that a steam generator tube rupture occurred within approximately 23 months of operation from the 1997 inspection. The risk assessment indicates that the reactor was susceptible to the various accident sequences primarily during the last year of this period. If the licensee's tube inspection and operational assessment processes that led to this event were repeated without improvement, it is expected that a similar result would occur. This is used to establish an average frequency for the steam generator tube rupture initiating event of about 0.5/reactor year (RY). Because the condition deteriorated with time, it can also be argued the initiating event frequency had zero increase over the first year and was increased about 1.0/RY during the second year. Multiplying these two estimates of the initiating event frequency by the probability that core damage would not be averted (about 1 x 10E-4) results in estimates for the incremental CDF of 5 x 10E-5/RY and 1 x 10E-4/RY, respectively. Consideration of the other pertinent sequences (where tube rupture is induced instead of initiating the sequence) is expected to add an additional increase on the order of 10E-5/RY. Therefore, the CDF/LERF increment associated with this event is considered to be clearly above the 10E-5/RY criterion for a 'Red" significance determination.

**Note: The following disclaimer was stated in the NRR memo referenced at the beginning of this section (from which the above risk analysis and assessment was taken).**

> "It should be noted that, if this risk analysis had been formally unitized as part of the revised reactor oversight program, it would have been subjected to additional review and discussion with the licensee and with the SDP and Enforcement Review Panel during the process for finalizing a significance determination. In addition, the assignment of a color in the significance determination process would depend upon a determination that the action or inaction that created the risk increment constituted inadequate performance by the licencee. Because, the agency has decided not to apply the revised program to this event at Indian Point, these steps were not taken, Therefore, this analysis should not be construed as the NRC's significance determination or the final establishment of a "color" for this event".

**3. Regional Recommended Strategy:**

A. The proposed enforcement action is a Severity Level III Violation and a Red finding based on the Significance Determination Process.
B. A regulatory conference is recommended.
C. No action is warranted against any individual.

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**4. Analysis of Significance/Root Cause:**

**a. Actual Consequence(s):**

The event had moderate risk significance. It involved a steam generator tube failure that resulted in an initial primary-to-secondary leak of reactor coolant of approximately 146 gallons per minute, and required an "Alert" declaration (the second level of emergency action in the NRC required emergency response plan). The event resulted in a minor radiological release to the environment that was well within regulatory limits. No radioactivity was measured off-site above normal background levels and, consequently, the event did not impact the public health and safety. The licensee's staff acted to protect the health and safety of the public. Specifically, the operators promptly and appropriately took those actions in the emergency operating procedures to trip the reactor, isolate the affected steam generator, and depressurize the reactor coolant system. Additionally, the necessary event mitigation systems worked properly. Notwithstanding the above, the NRC identified problems in several areas including operator performance, procedure quality, equipment performance, technical support, and emergency response. These problems challenged the operators, complicated the event response, and delayed the plant cooldown.

Several of the identified equipment problems such as a degraded steam jet air ejector steam supply valve, and an isolation valve seal water system design deficiency were long standing. The failure to correct these problems reflected weaknesses in engineering, corrective action processes, and operational support at the Station.

**b. Potential Consequence(s):**

See copy of the Memo from Richard J. Barrett, Chief Probabilistic Risk Assessment Branch, Division of Systems Safety and Analysis, Office of NRR, dated May 4, 2000 to A. Randolph Blough, Director, DRP, Region I, attached. The Probabilistic Safety Assessment Branch of NRR reviewed the information associated with the Indian Point, Unit 2, Steam Generator Tube Leak of February 15, 2000, and performed a risk assessment. The overall risk increase caused by the degradation of the tubes during operational Cycle 14 was estimated at approximately a 10E-4 increase in core damage probability and a similar magnitude increase in large release probability. The risk from spontaneous rupture is the dominant contributor to the increases in both the core damage and the large release probabilities. The risk contribution from ruptures induced by steam system depressurizations adds about 10% to these totals, and the risk contribution from other core damage sequences that induce tube failure adds perhaps

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another 10% to the probability of large release, without increasing the core damage probability.

c. **Potential for Impacting Regulatory Process**: LATER

d. **Willful Aspects**: None

e. **Root Causes:**

Con Edison returned Indian Point, Unit 2, to service in 1997 with the steam generator tubes in a condition that deteriorated with time to the point that a steam generator tube failure occurred within approximately 23 months of operation. Con Edison did not adequately evaluate poor quality data (low signal to noise ratios) that was encountered during the eddy current inspections in 1997. Con Edison failed to evaluate the effect on the probability of detection of small radius U-bend tube indications.
Though the cause of the failure of the 24 steam generator R2C5 tube was due to primary water stress corrosion cracking, there were other significant root causes that contributed to the steam generator tube failure event that were not addressed by Con Edison. Con Edison's root cause analysis did not adequately address the failure to identify the tube flaws in the low radius U-bend region during the 1997 outage. While the root cause analysis attributed the tube failure to a flaw that was obscured by eddy current signal noise, the adequacy in Con Edison's technical oversight of the 1997 steam generator inspections was not addressed. The root cause analysis conducted by Con Edison also did not address the adequacy of the corrective actions taken in response to a new SG degradation mechanism.

From a broad perspective the team found that Con Edison returned Indian Point, Unit 2, to service in 1997 in a condition that deteriorated with time to the point that a steam generator tube failure occurred. This resulted from Con Edison's weak technical oversight of this program and led to an inadequate, integrated technical understanding of the steam generator conditions during the 1997 inspection. The team identified the following significant performance issues:

(1) Based on an independent NRC review of the eight U-bend PWSCC indications that were identified in 2000 through review of existing 1997 inspection data, the NRC determined that Con Edison should have identified six of these defects including R2C5 and removed the associated tubes from service in 1997. The team found that several issues decreased the probability of defect detection and increased the likelihood of apex flaws in the small radius U-bend steam generator tubes. With respect to the U-bend indications, Con Edison failed to identify several factors that caused significant limitations and uncertainties in data collection and analyses, this increased the likelihood that steam generator tubes



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with detectable flaws would have been left in service. Overall, Con Edison did not ensure an adequate, integrated technical understanding of the steam generator conditions. Specifically:

(A) Con Edison did not evaluate and take necessary actions to compensate for equipment and technique challenges to flaw detection or to consolidate steam generator condition information to assess the significance of the new ID PWSCC degradation mechanism.

(B) Con Edison did not recognize the significance of and evaluate the flaw masking effects of the high noise encountered in the eddy current signal (low signal to noise ratios). In the case of SG 24, tube R2C5, the magnitude of the signal noise was estimated to equate to a 70-100% through-wall tube defect. The data analysis techniques were not adjusted to compensate for the noise to allow identification of flaw signal and ensure the appropriate probability of detection.

(C) Con Edison did not adequately responded to a PWSCC indication in the U-bend area of tube R2C67 in SG 24. This indication, which was located in the apex of this small radius tube, was a new and significant degradation mechanism at Indian Point 2. Apex cracking is more likely to burst than other u-bend cracks. Con Edison did not enter this significant issue into the corrective action program to ensure that this new degradation mechanism and the associated root cause were fully understood.

(D) Con Edison did not sufficiently assess eddy current probe restrictions in the upper support plate with respect to the upper flow slot hourglassing that increased the likelihood of increased apex stresses and PWSCC.

(E) Con Edison did not properly set-up the U-bend plus-Point eddy current probe, which negatively affected the probability of detection of U-bend indications. The probe was not set-up with the proper calibration standard or with the phase rotation specified by the EPRI qualified technique sheet.

(F) Con Edison did not have an accurate method of measuring nor some criteria for determining when significant hourglassing of the upper tube support plates had taken place. As such, no meaningful visual examination of the upper flow slots was conducted. Hourglassing in the lower tube support plates was significant in the 1980's, then quieted down

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until 1997 when 19 upper support plate restrictions where identified. These were the first indications of significant hourglassing at the upper support plate (which is the prime contributor to the stress in the low bend radius tubes that leads to PWSCC).

(2) Con Edison also failed to identify outside diameter stress corrosion cracking (ODSCC) in five tubes in the sludge pile area, just above the tube sheet.

**5. Apparent Severity Level(s)/Color and Basis:**

See Section 2. above.

**6. Application of Enforcement Policy**

**a. Enforcement/Performance History:**

(1) NRC INSPECTION REPORT 50-247/98-01 AND NOTICE OF VIOLATION - On February 9, 1998, the NRC completed an inspection at Indian Point 2. Several issues discussed concern observations by the inspectors, and by Con Edison through the site problem identification system, involving problems with the work control and the plant modification process that resulted in unnecessary challenges to control room operators, plant equipment and personnel safety. Several instances were noted where maintenance and surveillance test activities were adversely impacted by poor quality procedures.

Additionally, the NRC determined that violations of NRC requirements occurred. One violation concerned individuals who violated Con Edison's procedures for hot work activity when a worker continued grinding while his firewatch had left his assigned post. This violation was of particular concern to the NRC since it was similar to concerns previously expressed by the NRC regarding problems at Indian Point 2 with lack of procedure adherence, poor procedures, poor supervisory oversight, and plant management expectations not being met or enforced.

The second violation involved a modification to the control circuitry for the 23 emergency diesel generator that was improperly controlled in that an incorrectly sized relay was installed in the circuitry and was damaged during subsequent testing of the diesel. This issue represents an additional example of configuration control and drawing discrepancies adversely impacting plant operators and equipment as previously noted in NRC Inspection Report 50-247/97-15, Section



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(2) NRC INTEGRATED INSPECTION REPORT 50-247/98-02, NOTICE OF VIOLATION - On March 23, 1998, the NRC completed an inspection of Indian Point 2. Several issues discussed concerned observations by the inspectors of specific plant restart criteria. In several cases, substantial work was required to implement the necessary corrective actions. Based on the results of the inspection, the NRC determined that three violations of NRC requirements occurred.

The first violation is of particular concern because it involved four errors in a one month period involving improper tagging of equipment by operations personnel. While there were no personnel injuries or adverse equipment effects that resulted from the improper application of the tagouts, there was a significant potential to affect personnel safety. Fortunately, the tagging errors were either identified by the work group prior to the maintenance activity, and in one case, by an operations supervisor in response to locating another tag for a clearance. However, even though corrective actions were taken following three of the errors, their lack of effectiveness became apparent following a fourth event.

A second violation was issued for failing to have a Station Nuclear Safety Committee (SNSC) approved procedure for a surveillance test required by plant TSs. In reviewing this violation, the NRC noted that in Inspection Report 50-247/96-080, a violation (50-247/96-080-05) was also issued for failure to use a SNSC approved procedure during a radwaste processing activity. Together these violations indicated that a weakness exists in ensuring that relevant departmental procedures receive review by the Station Nuclear Safety Committee.

The third violation, involving a failure to meet 10 CFR 50.73 reporting requirements, was issued to reinforce the NRC expectations in meeting reporting timeliness following discovery of an event.

(3) NRC INTEGRATED INSPECTION REPORT 50-247/98-03, NOTICE OF VIOLATION, AND NOTICE OF ENFORCEMENT DISCRETION - **Excerpt:** "We note, with some concern, an increase in the number of errors during maintenance activities in which the independent verification process failed to prevent problems. In two separate instances safety-related electrical components were miswired, and in a third instance, the bolts for a pressurizer safety valve were over torqued during reinstallation. Further, the same bolt over torquing error occurred during the valve reinstallation in the 1997 refueling outage. Previous NRC reports have documented issues where the independent verification process was ineffective. For these events, it is our assessment, that while the processes for performing independent verification are adequate, they are not being effectively utilized by



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your staff, which again raises fundamental questions related to human performance at the site and requires your prompt attention in correcting.

Additionally, based on the results of this inspection, the NRC has determined that violations of NRC requirements occurred. The violations are cited in the enclosed Notice of Violation and the circumstances surrounding them are described in detail in the subject inspection report. One violation involving the wiring errors is discussed above. Another violation involved the isolation of the 21 emergency diesel generator air start motors during troubleshooting without proper procedural controls. This is of particular concern due to the potential for loss of configuration control through the manipulation of components outside of the approved methods which are procedures, tagouts, or check-off lists. Also, this is another example of a continuing concern the NRC has with procedure adherence and maintaining positive controls on equipment configuration by operations personnel. Two other violations were issued that involve informality in adherence to station processes, in particular, the modification process."

(4) NRC INTEGRATED INSPECTION REPORT 50-247/98-04 AND NOTICE OF VIOLATION - **Excerpt:** "In the area of radiological controls, it was determined that a violation of NRC requirements occurred. The violation is cited in the enclosed Notice of Violation and the circumstances surrounding it are described in detail in the subject inspection report. This violation is of concern because initial corrective actions to address a quality assurance finding that quality control checks of laundered protective clothing required strengthened implementation and management oversight were not effective in precluding a subsequent failure to implement the procedural requirements."

(5) NRC EVALUATION TEAM REPORT 50-247/98-005 - **Excerpt:** "This letter provides you with the results of an NRC Evaluation Team (NET) that observed your Independent Safety Assessment (ISA) of Indian Point Unit 2 (IP) during the period of March 30-May 7, 1998. The NET observed and evaluated the ISA in order to assess the validity of the ISA conclusions and to provide a recommendation to me as to whether the ISA had fulfilled the NRC's intent to obtain a current performance assessment through an Operational Safety Team Inspection (OSTI). In addition to observing the ISA, the NET (1) conducted an historical review of facility performance in order to place ISA findings in context, and (2) conducted plant tours and made a limited number of direct observations of licensee activities."

Overall, the ISA identified that some important deficiencies and weaknesses exist at the facility, particularly in the areas of management and operations. In

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highlighting one deficiency and five weaknesses in management and leadership areas, the ISA commented that the chain of command is not clearly defined, that direct supervision in the control room is weak, and that middle management has low visibility. Further, the ISA found that the site has been isolated with respect to learning from the rest of industry in that IP2 has been generally ineffective in observing and adopting industry best practices. Additionally, the ISA concluded that the site urgently needs an effective work management process; the current process is fragmented. The ISA found that Quality Assurance (QA) has been ineffectively used to improve plant performance due in part to a lack of line management support. Further, QA has become too involved in activities that more properly belong in line organizations. The ISA found that the problem identification process was receiving beneficial emphasis; however, the site corrective action processes were cumbersome and inefficient, many corrective actions were untimely, and completed actions were typically not revisited to see whether they had achieved the intended impact. The ISA found that surveillance program deficiencies, as well as noncompliances which your staff have been identifying, could challenge or prevent adherence to Technical Specifications. In the operations area, the ISA identified concerns with control of plant status by operators. The ISA directly observed a variety of examples of this long-standing configuration control problem, and listed several contributing causes, including reliance on informal controls and insufficient adherence to administrative procedures."

(6) NRC INSPECTION REPORT NO. 50-247/98-06 AND NOTICE OF VIOLATION - **Excerpt**: "Except for notable progress observed regarding the resolution of the DB-50 and DB-75 breaker problems, it was evident that your corrective actions for other major restart issues, such as the setpoint control program, were incomplete. As discussed in the enclosed inspection report, we have assigned unresolved items to several issues to track the final reviews, including consideration for any appropriate enforcement action.

Based on the results of this inspection, the NRC has determined that three violations of NRC requirements occurred. These violations are cited in the enclosed Notice of Violation (Notice)and the circumstances surrounding them are described in detail in the subject inspection report. The first violation involved a failure in your internal process which allowed a nonconforming mechanical part (fuel tube) to be installed in emergency diesel generator #22. The need to install qualified parts in the diesel generator is important to ensure that its performance at design conditions is maintained after necessary repairs. The other two violations involved failures concerning your service water (SW) system activities. Specifically, the findings in response to your SW self assessment performed in 1994 have not been promptly corrected and your staff failed to record the as-found conditions of both component cooling water heat exchangers during inspections in mid-1997 as required by your procedures. These violations are of



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concern since your system readiness reviews have been ongoing and should have identified and corrected these problems sooner."

(7) NRC INSPECTION REPORT NO. 50-247/98-08 AND NOTICES OF VIOLATION - **Excerpt:** "Your corrective actions regarding the various issues reviewed by the NRC were found to be acceptable. The findings and conclusions contained herein represent the culmination of over three months of NRC inspection to assess broad restart commitments involving setpoint and drawing control, as-built electrical wiring configuration, as well as follow-up to a number of technically involved design-related findings from the February 1998 NRC Architect Engineering team. Two findings were developed into issues of significance to emergency core cooling system (ECCS) performance, and as such were addressed in a comprehensive ECCS Design Bases Action Plan which was also evaluated as part of this inspection."

"Inadequate translation of design basis information into plant procedures and unauthorized changes to test procedures were cited in the first enclosed Notice as violations of NRC requirements for design control and changes to the licensed design of the facility. However, information regarding the reasons for those two violations, including corrective actions, is already adequately understood as addressed in this inspection report. Therefore, you are not required to respond to the first enclosed Notice unless the description therein does not accurately reflect your corrective actions or your position.

The second Notice describes two violations of NRC requirements which require your written response as set forth below. The first violation concerns your letter of January 6, 1998, in response to NRC Generic Letter (GL) 97-04, "Assurance of Sufficient Net Positive Suction Head (NPSH) for Emergency Core Cooling and Containment Heat Removal Pumps". The NRC concluded that information contained in your letter was incomplete in certain material respects, in violation of 10 CFR 50.9. The second violation involves five examples of deficiencies in electrical calculations; while individually the deficiencies were appropriately addressed, the findings collectively indicate a quality issue associated with electrical design activities."

(8) INDIAN POINT 2 MAINTENANCE RULE BASELINE TEAM INSPECTION REPORT 50-247/98-10 - **Excerpt:** "Overall, we concluded that your facility had implemented the maintenance rule acceptably at the time of the inspection, with the exception of the apparent violations identified below. We are concerned, however, that before January 1998 significant performance weaknesses and inadequacies existed that indicated an apparent organizational ineffectiveness in

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initial implementation of the maintenance rule. We noted that a number of revisions and substantial improvements to your program were implemented just prior to this inspection.

Based on the results of this inspection, the NRC has determined that apparent violations of NRC requirements occurred. The violations were identified by the NRC, in part, and included: 1) failure to place several systems into an (a)(1) status in a timely manner commensurate with safety, and 2) failure to include in the scope of the maintenance rule program as required by 10 CFR 50.65(b) the emergency lighting system, the communications system, and several annunciators."

(9) NRC MOTOR OPERATED VALVE INSPECTION 50-247/98-11 AND NOTICE OF VIOLATION - **Excerpt:** "Based on the results of this inspection, the NRC has determined that a violation of NRC requirements occurred. The violation involved inadequate review of vendor-performed calculations, and is cited in the enclosed Notice of Violation (Notice) because the deficiencies (if not known) could have resulted in a failure to perform additional actions needed to validate certain design assumptions. However, the deficiencies cited were isolated and did not represent a more broad concern with respect to review and acceptance of vendor-performed calculations. The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and to prevent recurrence are already adequately addressed in this inspection report."

(10) NRC SPECIAL INSPECTION REPORT 50-247/98-16 AND NOTICE OF VIOLATION - **Excerpt**: "A special inspection was conducted following the failure of a High Efficiency Particulate Air (HEPA) filter in the 25 containment fan cooler unit (FCU) on September 13, 1998, and the identification of HEPA filter deterioration in the 22 FCU on September 16, 1998."

"Based on the results of this inspection, the NRC determined that a violation of NRC requirements occurred. The violation involved two examples were procedures and instructions for activities affecting quality were not established. The first example involved the lack of administrative controls for the installation of roughing filters on the FCUs during outages as described in the UFSAR; this contributed to lapses in roughing filter installation which occurred during the last outage. The second example involved the lack of inspection and preventive maintenance of the drains for the safety-related FCU demisters designed to remove entrained water from the air flow stream, resulting in the partial or complete clogging the drains from the long-term accumulation of rust and debris. The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and to prevent recurrence are already adequately addressed in this inspection report."



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(11) NRC INTEGRATED INSPECTION REPORT 50-247/99-02 - **Excerpt:** "The NRC identified that radioactive waste processing performed at Unit 1 did not meet station requirements, a recurring problem from the last inspection period. Specifically, the inspectors identified a failure to follow procedures governing a temporary facility change. Subsequently, your staff performed a review of waste processing activities at Unit 1 and issued a stop-work order. Limited waste processing was resumed after implementation of short-term corrective actions."

"Based on the results of this inspection, the NRC determined that six violations of NRC requirements occurred. Five of the six violations were identified during NRC review of Licensee Event Reports (LERs) which you submitted from mid-1998 to early 1999. These violations are being treated as Non-Cited Violations (NCVs), consistent with Appendix C of the Enforcement Policy. These NCVs involve: (1) the failure to implement a temporary facility charge for processing radioactive water at Unit 1; (2) the failure to notify a licensed operator upon actuation of the Unit 1 spent fuel pool portable radiation monitor; (3) a missed Unit 2 TS surveillance for various radiation monitors; (4) a missed TS surveillance on sampling of the Unit 1 sphere foundation drain sump and north curtain drain; (5) information provided in the last three annual effluent and waste disposal reports for Units 1 and 2 that was not accurate in all material respects; and (6) inadequate surveillance procedures to test various Unit 2 safety-related logic circuits."

(12) NRC INSPECTION REPORT 50-247/99-03 - **Excerpt:** "Our evaluation of selected portions of the post-accident sampling system found several equipment deficiencies, plant drawing deficiencies and component labeling deficiencies which would challenge a chemistry technician's ability to acquire a sample during a postulated emergency. Your short-term corrective actions appropriately considered extent of condition for the sampling systems and each NRC observation was appropriately placed into the problem identification process.

Based on the results of this inspection, the NRC has determined that two Severity Level IV violations of NRC requirements occurred. These violations are being treated as Non-Cited Violations (NCVs), consistent with Appendix C of the Enforcement Policy. These NCVs involve untimely reporting of licensee event report (LER) 98-020 and a long-standing design deficiency for the containment pressure relief system as documented in LER 99-002."

(13) NRC TEAM INSPECTION REPORT 50-247/99-04 AND EXERCISE OF ENFORCEMENT DISCRETION - **Excerpt**: "The team focused on your engineering activities and included an in-depth review of the safety injection system and its use with the pressurizer power operated relief valves in the "primary bleed" cooling function. In addition, the team reviewed the setpoint



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verification program, several open issues regarding Maintenance Rule and 10 CFR 50 Appendix R issues, and a number of engineering activities involving plant safety and risk significant systems."

"Based on the results of this inspection, the NRC determined two Severity Level IV violations of NRC requirements occurred. The violations are being treated as Non-Cited Violations (NCVs) consistent with Appendix C of the Enforcement Policy. The NCVs involved the failure to validate set point tolerances for the toxic gas monitor, and the failure to maintain environmental qualification (EQ) for acoustic monitors and limit switches in the pressurizer relief system."

(14) NRC INTEGRATED INSPECTION REPORT 50-247/99-06 AND NOTICE OF VIOLATION - **Excerpt**:"We are concerned with instances of ineffective planning and execution of maintenance activities during the period. Examples of ineffective maintenance planning and execution involved inadequate contractor oversight during gas turbine operations, poor planning and awareness of plant equipment as it relates to central control room ventilation, ineffective planning of leakrate surveillances, inadequate pre-job briefings for Unit 1 ventilation work, and poor coordination between operations and maintenance on containment isolation valve work. The consequences of the ineffective maintenance included in an entry into an abnormal operating instruction for an unexpected temperature rise in the central control room, the spread of contamination in Unit 1, and inoperability of a containment isolation valve."

"Based on the results of this inspection, the NRC has determined that two Severity Level IV violations of NRC requirements occurred. One violation is cited in the enclosed Notice of Violation (Notice) and the circumstances surrounding it are described in detail in the subject inspection report. The NRC identified two recent examples in which you failed to implement procedures and instructions associated with temporary facility changes. This violation is of concern because it was repetitive as a result of ineffective implementation of corrective actions for previous violations involving temporary facility changes."

"The remaining Severity Level IV violation is being treated as a Non-Cited Violation (NCV), consistent with Appendix C of the Enforcement Policy. It involved an untimely submittal of Licensee Event Report (LER) 99-011-00 that documented a loss of individual rod position indication."

(15) NRC AUGMENTED INSPECTION TEAM - REACTOR TRIP WITH COMPLICATIONS - REPORT NO. 50-247/99-08 - **Excerpt**: "On September 27, 1999, the NRC completed an Augmented Inspection Team (AIT) at the Indian

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Point Unit 2 (IP2) Station."

"The AIT was chartered (Enclosure 2) to review the causes, safety implications, and your staff's actions involving the reactor trip with complications at IP2 on August 31, 1999. The NRC noted that the event was complicated by unexpected system interactions that involved safety-related equipment. The team reviewed the record of activities that occurred, interviewed plant personnel, and conducted plant walkdowns. The team developed a sequence of events, determined the causes and risk significance of the event, and assessed the quality of response by the plant staff and management. A summary of the team's findings was presented at a public exit meeting on September 27, 1999. The NRC briefing slides from that meeting are provided in Enclosure 3.

Although there was no immediate threat to public health and safety, the event was risk significant. The event involved a loss-of-offsite power to all four of the 480 volt vital buses, the additional loss of the emergency diesel generator supplying one of those buses (along with some other risk-significant equipment), and the depletion of one of the four safety-related batteries. Other than one cell of the depleted battery needing replacement, there was no damage to plant equipment. Additionally, there was no radiological release due to the event.

The team determined that the event was preventable and was caused primarily by problems in plant configuration control. Contributing to these were some notable weaknesses in the corrective actions and technical support areas. In addition, weaknesses in management oversight during the event contributed to the delay in restoring normal electrical power supplies.

Configuration control problems included the station auxiliary transformer load tap changer being left in a position contrary to licensing bases. This led to a loss of offsite power to the vital buses following the plant trip. Poor control of emergency diesel generator output breaker short time over-current trip settings, compounded by a deficiency with the timing of the sequencing relays for some safety-related loads, caused the loss of emergency power to one of the vital buses.

Weaknesses were also noted in management oversight of the station's response to the event. Management did not promptly recognize the significance of the degrading conditions associated with the event. Managers appeared to focus primarily on developing shutdown work plans and schedules instead of establishing and prioritizing activities to restore plant equipment and to limit further risk. As a result of these weaknesses, station personnel provided poorly coordinated and untimely support to plant operators in restoring normal electrical power. Likewise, the post-trip response organization did not provide support to operations in the review of plant conditions relative to the emergency plan. As such, station personnel did not recognize that the declaration of an Unusual



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Event was missed when offsite power was lost to all 480 volt vital buses."

(16) NRC INTEGRATED INSPECTION REPORT 50-247/99-09 - **Excerpt**: "Our inspectors noted a number of corrective action deficiencies during the period. The deficiencies involved improper closure of a gas turbine condition report, insufficient causal analysis for the 24 reactor coolant pump motor oil leakage, and ineffective corrective actions to address long-standing oil leakage from the 22 charging pump. We note that Inspection 99-08 also highlighted deficiencies in the quality of causal analyses and corrective actions.

Further, the failure to incorporate industry operating experience related to sample coolers in the condensate and feedwater systems was a missed opportunity to prevent the October 14, 1999, chemistry excursion in the steam generators. Our inspectors noted other performance deficiencies contributed to problems controlling secondary chemistry, including the injection of boron into the secondary system without procedural controls, the lack of a preventive maintenance program for secondary system sample coolers, and the failure to conduct timely extent of conditions reviews following past sample cooler tube leaks. The consequence of these deficiencies was an event that challenged the steam generator tube integrity.

Based on the results of this inspection, the NRC has determined that two Severity Level IV violations of NRC requirements occurred. These violations are being treated as Non-Cited Violations (NCVs), consistent with Section VII.B.1.a of the Enforcement Policy (November 9, 1999; 64 FR 61142). The NCVs involve the failure to follow procedures per Technical Specification 6.8.1 during operation of the reactor coolant pumps and when adding chemicals to the steam generators."

(17) NRC INTEGRATED INSPECTION REPORT 50-247/99-09 - **Excerpt**: "Our inspectors noted a number of corrective action deficiencies during the period. The deficiencies involved improper closure of a gas turbine condition report, insufficient causal analysis for the 24 reactor coolant pump motor oil leakage, and ineffective corrective actions to address long-standing oil leakage from the 22 charging pump. We note that Inspection 99-08 also highlighted deficiencies in the quality of causal analyses and corrective actions.

Further, the failure to incorporate industry operating experience related to sample coolers in the condensate and feedwater systems was a missed opportunity to prevent the October 14, 1999, chemistry excursion in the steam generators. Our inspectors noted other performance deficiencies contributed to problems controlling secondary chemistry, including the injection of boron into the secondary system without procedural controls, the lack of a preventive maintenance program for secondary system sample coolers, and the failure to conduct timely extent of conditions reviews following past sample cooler tube



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leaks. The consequence of these deficiencies was an event that challenged the steam generator tube integrity.

Based on the results of this inspection, the NRC has determined that two Severity Level IV violations of NRC requirements occurred. These violations are being treated as Non-Cited Violations (NCVs), consistent with Section VII.B.1.a of the Enforcement Policy (November 9, 1999; 64 FR 61142). The NCVs involve the failure to follow procedures per Technical Specification 6.8.1 during operation of the reactor coolant pumps and when adding chemicals to the steam generators."

(18) NRC INTEGRATED INSPECTION REPORT 05000247/1999011 - **Excerpt:** "Our inspectors noted that degraded equipment conditions and ineffectiveness of the work control process continued to challenge operators. These conditions include freezing of safety-related instrumentation, too many concurrent test or maintenance activities in the control room, and the untimely conduct of corrective maintenance on the station auxiliary transformer."

"Based on the results of this inspection, the NRC has determined that four Severity Level IV violations of NRC requirements occurred. These violations are being treated as Non-Cited Violations (NCVs), consistent with Section VII.B.1.a of the Enforcement Policy (November 9, 1999, 64 FR 61142). The NCVs involve the failure to maintain fire protection equipment operable due to inadequate implementation and testing of a plant modification, and the failure to adequately test fire protection and reactor protection system equipment."

(19) RESULTS FROM THE FOLLOW UP INSPECTION TO THE AUGMENTED INSPECTION TEAM, NRC INSPECTION REPORT 05000247/99013 - **Excerpt:** "The inspection was focused on your short term corrective actions and other self-assessment activities as a result of the August 31, 1999, reactor trip with complications. This inspection followed both our Augmented Inspection Team (AIT) review of the event and your initial recovery efforts as described to us at a September 14, 1999, meeting in our King of Prussia office."

"Our inspectors noted many examples of mixed performance in the your recovery efforts. For example, our probing led to the identification that a safety related breaker had been returned to service even though data obtained during testing was out of your pre-approved tolerances. Also, during training of your plant operators on your newly issued procedure for recovery of a 480 volt safety bus following loss of power, a number of discrepancies were identified that necessitated a procedure revision."

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"We also found that there were other significant areas of weak performance needing your added attention and longer term followup to assure that the improvements you initiated as a result of the August 31 event are effective. A significant finding during the recovery effort was that your plant work backlog contained issues of safety significance that persisted without timely corrective action or performing a safety evaluation. For example, the station load tap changer remained in the manual position for approximately one year prior to the event without adequate compensatory measures to assure that the function of the tap changer would be retained following a unit trip. As a result of this finding, you corrected the deficient condition that had necessitated keeping the tap changer in the manual position, and revised station procedures to limit the time that the tap changer could remain in manual to coincide with the technical specification limitations on operability of the offsite power supplies. You also initiated a broad review of station work backlogs to verify that other degraded conditions were adequately evaluated for safety impact on operations or were corrected prior to unit restart. Providing assurance that future degraded conditions are addressed in a timely manner is another area that requires further action on your part. You informed us that you intend to complete an effectiveness review of corrective actions resulting from this event.

"Based on the results of this inspection, the NRC has determined that a number of violations of NRC requirements occurred. The violations described in the attached report are being treated as a non-cited violations (NCV), consistent with Section VII.B.1.a of the Enforcement Policy (November 9, 1999; 64 FR 61142). All of the violations relate to your activities taken following the August 31,1999 event and your subsequent corrective activities."

**b. Is Credit Warranted for Identification? Explain:**

Credit is not warranted for identification. The problem was revealed through the steam generator tube failure event of February 15, 2000. Missed opportunities are described in Section 4.e. above.

**c. Is Credit Warranted for Corrective Actions? Explain:**

Credit is not warranted for Corrective Actions. Though the actions to correct the problem of stopping the primary to secondary leakage and associated release of radioactivity to the environment, the corrective actions are not comprehensive and are still being reviewed by NRR. Additional corrective actions, such as plugging all row 3 steam generator tubes is under discussion. Issues related to Con Edison's Condition Monitoring and Operational Assessment of the event are not yet resolved.



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The root cause provided by the licensee was inadequate as described in Section 4.e above.

**d. Should Discretion Be Exercised to Mitigate or Escalate Sanction?**

There are three issues on the 'List of Issues That May Warrant Discretion' for consideration.

(1) Case involves particularly poor licensee performance. From a broad perspective the team found that Con Edison returned Indian Point, Unit 2, to service in 1997 in a condition that deteriorated with time to the point that a steam generator tube failure occurred. This resulted from Con Edison's weak technical oversight of this program and led to an inadequate, integrated technical understanding of the steam generator conditions during the 1997 inspection. The team concluded that during the 1997 steam generator inspection, Con Edison did not recognize and take corrective actions for significant conditions adverse to quality relating to eddy current data collection and analysis and specific steam generator conditions. These missed opportunities caused significant limitations and uncertainties, resulting in tubes with detectable flaws being left in service. Collectively, these opportunities, along with a new active degradation mechanism, increased the likelihood of tube integrity problems during the subsequent operating cycle. The team identified significant performance issues that are described in Section 4.e.

(2) Excessive duration of the problem resulted in a substantial increase in risk. The team found that Con Edison returned Indian Point, Unit 2, to service in 1997 in a condition that deteriorated with time to the point that a steam generator tube failure occurred.

(3) Discretion should be exercised by escalating or mitigating to ensure that any proposed civil penalty reflects the NRC's concern regarding the violation at issue and that it conveys the appropriate message to the licensee. The team concluded that during the 1997 steam generator inspection, Con Edison did not recognize and take corrective actions for significant conditions adverse to quality relating to eddy current data collection and analysis and specific steam generator conditions. These missed opportunities caused significant limitations and uncertainties, resulting in tubes with detectable flaws being left in service. Collectively, these opportunities, along with a new active degradation mechanism, increased the likelihood of tube integrity problems during the subsequent operating cycle. Since the plant is already an Agency Focus Plant escalation may not be needed.



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7. **Is action being considered against individuals?**

No.

8. **Non-Routine Issues/Additional Information/Relevant Precedent/Lessons Learned:**

]

a. Generic communication may be needed for this issue regarding NRC expectations and observations related to the use of the EPRI Guidelines on steam generator eddy current testing, poor signal to noise ratios (high noise levels), the significance of top tube support plant hourglassing and U-bend/top support plate restrictions, and contractor oversight. NRR would provide any programmatic guidance deemed necessary.

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**Attachment 1**

**SDP/Enforcement Panel Disposition Record**

Licensee : _____
EA No. _____
Panel Date: _____
Issue: _____

**Attendees**

Chair - _____ Branch Chief - _____ Enf Reps _____
OI Rep. - _____ RI Counsel - _____ Others - _____
HQ Reps _____

**Required Actions** (Preliminary Proposed Actions - See OE Strategy Form for official record of panel decision.)

1)

Responsible Person: _____ ECD: _____

2)

Responsible Person: _____ ECD: _____

3)

Responsible Person: _____ ECD: _____

4)

Responsible Person: _____ ECD: _____

**Examples of Specific Actions To Be Documented**

- Call Licensee and Schedule Conf or give heads up on choice letter
- Prepare summary of OI findings as attachments to choice letter or conf letter
- Issue letters scheduling conference or providing choice
- Gather additional information and repanel
- Prepare the draft enforcement action
- Finalize the enforcement action
- Forward Package to OE



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Attachment 2

ISSUES TO CONSIDER FOR DISCRETION

Problems categorized at Severity Level I or II.

Case involves overexposure or release of radiological material in excess of NRC requirements.

Case involves particularly poor licensee performance.

Current violation is directly repetitive of an earlier violation.

Excessive duration of the problem resulted in a substantial increase in risk.

Case may involve willfulness. Include information to address whether or not the region has had discussions with OI regarding the case, whether or not the matter has been entered into the allegation system as staff suspected wrongdoing, and whether or not OI intends to initiate an investigation. Include a description, as applicable, of the facts and circumstances that address the aspects of negligence, careless disregard, willfulness, and/or management involvement.

Licensee made a conscious decision to be in noncompliance in order to obtain an economic benefit.

Case involves the loss of a source. (Note whether the licensee self-identified and reported the loss to the NRC.)

Licensee's sustained performance has been particularly good.

Discretion should be exercised by escalating or mitigating to ensure that any

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**proposed civil penalty reflects the NRC's concern regarding the violation at issue and that it conveys the appropriate message to the licensee.**

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**Attachment 3**

**FACTORS FOR THE SANCTION IN ACTIONS AGAINST INDIVIDUALS**

1. **The level of the individual within the organization**
   Corporate executive in large organization
   RSO, SRO or manager above first line supervisor (e.g., President of small business, plant manager)
   First line supervisor or other licensee official (e.g., authorized user, chief technologist, RO, radiographer)
   User (e.g., AO, assistant radiographer, technologist, technician, QA)
   Not normally involved in NRC-Licensed activities (e.g., laborer, carpenter, millwright etc.)
   Other, Explain: _____

2. **Culpability, the individual's training and experience as well as knowledge of the potential consequences of the wrongdoing**
   Prior individual action against individual by NRC or significant discipline to individual for similar wrongdoing by licensee
   Well-trained, experienced, no excuse for not appreciating the significance of wrongdoing, or management told individual not to do the wrongdoing
   Knows it is wrong but does not appreciate the significance of the wrongdoing (does not care)
   Newly hired, little or no experience, Knows it is wrong but does not appreciate the significance of wrongdoing; following culture of the organization
   Deliberate  Careless disregard  No prior nuclear employment  Not likely to work nuclear in the future
   Other, Explain: _____

3. **The safety consequences of the misconduct**
   Overexposure to individual(s)  Loss of redundancy or inoperable safety system
   Misadministration to individual(s)  Low consequences
   Release of radiation or radioactive material  No potential consequences
   Affects public health and safety  No consequences
   Other, Explain: _____

4. **The benefit to the wrongdoer**
   Significant tangible gain (e.g., Monetary, financial decision, promotion, clear motive)
   Tangible gain (e.g., avoidance of discipline, concerned about NRC inspection or licensee audit, clear motive)
   No real benefit (e.g., leave early, get job done more quickly)

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Other, Explain: _____
Benefit to Licensee, Explain: _____

5. **The degree of supervision of the individual**
   Close supervision (e.g., supervisor in area most of the time)
   Moderate supervision (e.g., supervised occasionally or audited occasionally)
   No supervision
   Other, Explain: _____

6. **The employer's response**
   Very significant impact to individual (e.g., dismissal, denied unescorted access, placed in PADS etc.)
   Substantial discipline (e.g., fine, demotion, probation, additional oversight of individual, removal from licensed activities)
   Some discipline (e.g., counseling)
   None
   Other, Explain: _____

7. **The attitude of the wrongdoer**
   Significant interference with investigation (e.g., actions such as destroying records, persuading others to lie)
   Interference with investigation (e.g., affirmative lying)
   Does not accept responsibility during investigation, exculpatory "no," does not provide testimony (e.g., exercising the Fifth Amendment privilege is neutral under this element)
   Admits to wrongdoing and acceptance of responsibility
   Cooperates during inspection and/or investigation
   Voluntarily identified and self reported the wrongdoing with minimal expectation that it would be discovered
   Other, Explain: _____

8. **The degree of management responsibility or culpability**
   Management directed and employee complains
   Management directed; however, employee does not question even though employee knows it is wrong
   Not directed by management but management does not provide resources to get the job done such that management is implicitly inviting cutting of corners, and individual does not complain
   Management Knew of questionable conduct and took no action to correct conduct
   No management involvement
   Other, Explain: _____



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9. **Who identified the misconduct**
   Individual
   Licensee (through audit, LER, and/or investigation)
   Third party (e..g., alleger, union, newspaper, etc.)
   NRC (through inspection, LER, and/or investigation)
   Other, Explain:_____

10. **Duration of violation**
    Repetitive or continues over time; How long_____
    Isolated or relatively isolated

11. **Other**
    The individual directed or coerced others to engage in the wrongdoing at issue
    Unusual event with significant health and safety consequences such as death or serious injury

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