"SDP risk assess"

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Memorandum to: A. Randolph Blough, Deputy Director
Division of Reactor Projects
Region I

From: Richard J. Barrett, Chief
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Subject: **Risk Assessment and Input to Significance Determination Process for Condition of Indian Point Unit 2 Steam Generator Tubes During Operational Cycle 14**

As you requested, the Probabilistic Safety Assessment Branch has reviewed the information available and performed a risk assessment for the recent findings at Indian Point unit 2.

During operation cycle 14, Indian Point unit 2 experienced degradation of steam generator tubes that culminated in failure of a flaw in the U-bend of tube R2C5 in steam generator 24. In addition, inspection following the tube failure event revealed 5 additional tubes with defects in the same region of steam generator A, plus other defects in other regions and other generators. However, none of these other defects appears to have become susceptible to induced rupture by the time tube R2C5 ruptured spontaneously.

The risk associated with the condition of the tubes during cycle 14 comes from several potential accident sequences:

1. Spontaneous rupture of a tube, not successfully mitigated by plant operators, causing core damage and bypass of the containment by large radioactive releases.

2. 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.

3. 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.

4. 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.

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Of these, the first two increase both the core damage frequency and the frequency of large radioactive releases bypassing the containment and reaching the environment (hereafter assumed to be a "large early release"). The latter two sequences are already included in the plant's core damage frequency (CDF) 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 10^{-4}. This risk occurred mostly during the latter year of the operational cycle.

The basis for this estimate is discussed below for each potential accident sequence, individually.

Spontaneous Tube Rupture:

The Indian Point unit 2 probabilistic risk assessment (PRA) includes this sequence. The probability of the initiating event, spontaneous tube rupture, was assumed to be 1.3 x 10^{-2} per reactor-year of operation (RY) and the resulting core damage frequency was estimated as 1.0 x 10^{-6} / RY. From this, the conditional probability for failing to mitigate a rupture after it occurs is inferred to be 7.7 x 10^{-5}. This number is comparable to the conditional probability values obtained from the NUREG-1150 model for Surry, 1.4 x 10^{-4}, and from the NRC's Rev.2QA SPAR model for Indian Point unit 2, 3.3 x 10^{-4}. So, given that the spontaneous rupture initiating event did occur at Indian Point unit 2, the conditional probability of core damage is estimated to be about 1 x 10^{-4}. Because most of the core damage sequences resulting from spontaneous tube rupture involve loss of steam system integrity, approximately the same conditional probability applies to the occurrence of a large early release of radioactive material to the environment.

The most probable reasons for a spontaneous rupture event to cause core damage involve human errors while attempting to cool down the unit. The probability of the operators making (and not correcting) these errors depends on the amount of time available to them, which depends on the leak rate through the ruptured tube. The PRAs assume that the rupture is as large as can occur with one tube, which creates a leak flow of several hundred gallons per minute (gpm). The rupture that actually occurred at Indian Point unit 2 resulted in only about 150 gpm of leakage. So, the operators had much more time to correct the situation than is assumed in the PRA models that were used above to estimate the conditional probability of core damage. Thus, it can be argued that the probability of the Indian Point operators failing to mitigate this particular rupture was much lower than 10^{-4}. However, the flaw that failed in the Indian Point tube was about 2 inches long, and a flaw this long is capable of bursting to the extent assumed in the PRAs. The fact that the tube flaw was held partially closed by several ligaments across the flaw is the reason that it did not open completely and leak much more. Experience has shown that the probability is about 0.5 that tubes with large flaws will leak substantially or only partially break open before they fail completely, allowing operators an opportunity to intercede before complete failure occurs. Thus, the fact that the type of degradation that occurred can result in large flaws and that the flaw that failed was indeed large indicates that the risk associated with the degradation at Indian Point unit 2 is best estimated as having about 10^{-4} conditional probability of core damage and large release from the spontaneous rupture sequence.

Ruptures Induced by Steam System Depressurization:

Core damage sequences of this type are not generally included in licensees' PRAs, but have been evaluated by the NRC in NUREGs 0844, 1477 and 1570. They are similar to the spontaneous rupture sequences in licensees' PRAs except that the loss of steam system integrity comes first and causes the tube rupture instead of *vice versa*. As in the spontaneous rupture sequences, the most probable path to core damage involves errors in the operators' response to the conditions that occur. For a tube rupture induced by a steam system depressurization, the errors are estimated to be more probable because the events are more complicated and the operators do not normally drill on this type of sequence.

In the case of Indian Point unit 2, it is clear that a secondary depressurization event would have caused tube R2C5 to rupture when it was in the weakened condition that just preceded its spontaneous rupture. During that period, the core damage frequency (and large release frequency) is estimated using a steam system depressurization frequency of 7.6×10^{-3} / RY, the assumption that only one of four steam generators was susceptible, a conditional rupture probability of 1.0, and a human error probability of 10^{-2}. The result is an increase in both the core damage frequency and the large release frequency of about 1.9×10^{-5}/ RY.

However, in order to estimate the increase in probability of core damage and large release, it is necessary to consider the length of time that this increase in frequency is applicable. Based on the currently available information, the period of time the tube was susceptible to this accident sequence is estimated in Appendix A as approximately four to eleven months or 0.3 to 0.9 year. Thus, the number of ruptures that would be mathematically "expected" for this frequency over this period is 6×10^{-6} to 1.7×10^{-5}. For such small expectation values, the probability of occurrence of a single event is numerically indistinguishable, so the increase in the probability of core damage and large release from this sequence for this condition is estimated to be about 1×10^{-5}.

Ruptures Induced by Reactor System Over-Pressurization Events:

Tube ruptures that are induced by the normal operational occurrences that involve slight elevations in reactor system pressure are considered to be captured by the value used for the frequency of spontaneous ruptures. The additional sequences considered here are those involving gross over-pressure events that, by themselves, would produce core damage. These result from failure of the reactor control system to shut down the nuclear chain reaction when required by a design basis transient, such as loss of feed water to the steam generators. These events are called anticipated transients without scram (ATWS) events. Most licensees' PRAs include core damage sequences due to ATWS events, but do not consider the probability that such an event could also rupture a steam generator tube, causing containment bypass by the radioactive material it would release from the damaged reactor core.

The PRA for Indian Point unit 2 estimates a core damage frequency contribution of 1.81×10^{-6} / RY due to ATWS events. ATWS events that create a reactor coolant system pressure above 3,200 psi are assumed to lead to core damage. During the period of extreme reactor system pressure, the steam system pressure is expected to be at the steam system safety valve setpoint, producing a pressure differential across the steam generator tube walls of at least 2100 psid. Based on the rate of degradation estimated in Appendix A, we estimate that an ATWS event would have induced tube R2C5 to rupture for a period greater than 3 months. In the same manner described above for steam system depressurization sequences, this results in an estimated increase in the large early release probability that is $> 4 \times 10^{-7}$, perhaps by a factor > 3. There is no increase in the core damage probability because the ATWS sequences that would induce the tube rupture are already part of the core damage frequency estimate, and the addition of the tube rupture potential is not assumed to change the frequency with which ATWS would cause core damage.

Tube Ruptures Induced by Other Core Damage Sequences:

Other core damage sequences that are included in licensees' and NRC's PRAs may also cause large releases by inducing steam generator tube ruptures, but this effect is rarely included in the results of current PRAs. The studies documented in NUREG-1150 and particularly NUREG-1570 do address this potential for large releases to bypass containment due to tube failures. For accident sequences in which the reactor coolant system (RCS) remains at high pressure, the failures of flawed tubes may be caused by steam system depressurization that sometimes occurs as an essential or incidental part of the event sequence that leads to core damage. Also, for sequences with high RCS pressure and dry steam generators (hi/dry sequences), tube failure may be induced when the overheating reactor core causes the tube temperatures to rise so high that their metal weakens. Tubes with flaws that would not fail upon steam system depressurization may still fail when the tube temperatures increase, later in the accident sequence. This is clearly the case for the Indian Point tube for some period during the last cycle, before it was susceptible to failure by steam system depressurization, alone. It also is clear that, for some shorter period of time, tube R2C5 would have failed if dry and overheated by a high pressure core damage accident, even if the steam system remained pressurized.

To accurately estimate the additional probability of a large release due to a core damage accident during the last cycle, it is necessary to separately identify the hi/dry core damage sequence frequency and subdivide it into cases with and without steam system depressurization. It also is necessary to estimate the time periods during which tube R2C5 was susceptible to rupture 1) from steam system depressurization, alone, 2) from high temperature without steam system depressurization, and 3) from the combination of high temperatures and steam system depressurization.

However, without expending the effort to perform this detailed analysis, it can be seen that the result would not substantially change the overall risk estimate for the situation at Indian Point unit 2 during cycle 14. This is based on the fact that the total core damage frequency is estimated to be 2.6×10^{-5} / RY. Although the majority of this frequency is expected to be hi/dry sequences, and about half of those sequences may involve steam system depressurization, the contribution to the total increase in the large release probability would still be about an order of magnitude less than the dominant contribution from spontaneous tube rupture, even if tube R2C5 was susceptible for about a year.

Summarization of Overall Risk Increase:

On the basis of the foregoing discussions, it is estimated that the risk increase caused by the degradation of the tubes at Indian Point unit 2 during operational cycle 14 is approximately 10^{-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% of 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. More detailed analysis is not expected to change the magnitude of this estimate.

The risk input for use in a Significance Determination in accordance with the new Reactor Oversight Process is provided in Appendix B.

If you or your staff would like to discuss this assessment in further detail, please feel free to contact me or Steve Long.

cc: William M. Dean

Appendix A

Flawed Tube Strength as A Function of Time

Based on the license's reanalysis of their eddy current results from 1997, it appears that an inside diameter flaw approximately 2.4 inches long and averaging approximately 72% through wall was present in steam generator A tube R2C5 when the plant was returned to service.

Based on these flaw size measurements, NRC staff in the Division of Engineering performed burst pressure estimates for the subject tube at the time it was returned to service. Available burst pressure prediction models apply specifically to straight tubes rather than to u-bend geometries. These straight tube models indicate a burst pressure in the range of 3200 to 3620 psi. Westinghouse work in the early 1980's indicates that tubes exhibit higher burst strengths in the u-bends for a given size flaw than in the straight length portions due to the cold-worked state of the material in the u-bends. This Westinghouse work is not well documented nor is there much corroborating evidence for this work. The best that can be drawn from this information at this time is that burst pressures are somewhere between zero and 58% higher in the u-bend than the straight length regions for given size flaws. Thus, the staff concludes that the subject tube had a burst capability in the range of 3200 to 5700 psi at the time the plant was returned to service in 1997.

When the tube burst during operation, it's burst pressure had decreased to the plant's normal operating pressure differential, 1600 psid. The period of power operation that elapsed between these times was 22.5 months.

Assuming that the growth in the flaw created a decrease in strength that was linear with time, the following table was constructed for the duration of the periods that the flawed tube was susceptible to rupture at various pressure levels that are important thresholds for the risk assessment process.

Initial strength	= 3,200 - 5,700 psid	at	23 months
TI-SGTR threshold	< 2,800 psid*	for	7 - 17 months
PI-SGTR threshold	< 2,350 psid	for	4 - 11 months
Spontaneous rupture	= 1,600 psid		(instantaneous)

* This value is an approximation, based on the stress magnification factor that resulted in a 50% failure probability in the analysis previously performed for the Farley unit 1 license amendment application review. Of the analyses currently available to the staff, that one is the most similar to the Indian Point unit 2 reactor. However, that analysis contained many assumptions about the location of the flaw and the spatial distribution of tube temperatures that are not identical to the situation at Indian Point unit 2. In addition, these two reactors have not been verified to produce the same thermal-hydraulic conditions for severe accident sequences. However, because the value is not crucial to the conclusion, it is considered sufficient and useful to indicate the nature of the situation.

Significance Determination Input

The draft 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 large, early release (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 titled "Basis Document for Large Early Release Frequency (LERF) Significance Determination Process (SDP) - Inspection Findings That May Affect LERF." The Office of Research is sponsoring the project at Brookhaven National Laboratory that is developing this guidance. The guidance is summarized in Table 1 of that document as shown here.

Table 1 Risk Significance Based on LERF and CDF

Frequency Range/ry	SDP Based on CDF	SDP Based on LERF
$\geq 10^{-4}$	Red	Red
$< 10^{-4} - 10^{-5}$	Yellow	Red
$<10^{-5} - 10^{-6}$	White	Yellow
$<10^{-6} - 10^{-7}$	Green	White
$<10^{-7}$	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 sufficient to establish the risk input for determining the "color" of the situation that occurred at Indian Point unit 2.

Indian Point unit 2 was returned to service in 1997 in a condition that deteriorated with time to the point that at steam generator tube rupture occurred within approximately 23 months of operation. 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 / 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 10^{-4}) results in estimates for the incremental CDF of 5 x 10^{-5}/ RY and 1 x 10^{-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 10^{-5}/ RY. Therefore, the CDF/LERF increment associated this event is considered to be clearly above the 10^{-5}/ RY criterion for a "red" significance determination.

However, the assignment of a color in the significance determination process also depends upon a determination that the action or inaction that created the risk increment constitutes inadequate performance by the licensee. The Division of Engineering is responsible for making the input for that part of the assessment.