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U. S. Nuclear Regulatory Commission
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Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Subject: Response to Request for Additional Information for Technical Specifications Change to Revise Steam Generator Inspection Frequency for the Fall 2001 Refueling Outage for Braidwood Station, Unit 1

References:

(1) Letter from R. M. Krich (Exelon Generation Company, LLC) to US NRC, "Request for Technical Specifications Change Braidwood Station, Unit 1, Steam Generator Inspection Frequency Revision for the Fall 2001 Refueling Outage," dated February 9, 2001

(2) Letter from M. Chawla (US NRC) to O. D. Kingsley (Exelon Generation Company, LLC), "Request for Technical Specifications Change - Braidwood Station, Unit 1, Steam Generator Inspection Frequency Revision for the Fall 2001 Refueling," dated May 4, 2001

(3) Letter from R. M. Krich (Exelon Generation Company, LLC) to US NRC, "Response to Request for Additional Information for Technical Specifications Change to Revise Steam Generator Inspection Frequency for the Fall 2001 Refueling Outage for Braidwood Station, Unit 1," dated May 18, 2001

(4) Letter from M. Chawla (US NRC) to O. D. Kingsley (Exelon Generation Company, LLC), "Request for Technical Specifications Change - Braidwood Station, Unit 1, Steam Generator Inspection Frequency Revision for the Fall 2001 Refueling," dated June 1, 2001

In the Reference 1 letter, in accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company (EGC), LLC requested a change to the Technical Specifications (TS) of Facility Operating License Nos. NPF-72 and NPF-77 for the Braidwood Station, Units 1 and 2. The proposed one-time change revises the Steam Generator (SG) inspection frequency requirements in TS 5.5.9.d.2, "Steam Generator (SG) Tube

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Surveillance Program, Inspection Frequencies," for the Braidwood Station, Unit 1 fall 2001 refueling outage to allow a 40 month inspection interval after one SG inspection, rather than after two consecutive inspections resulting in C-1 classification.

The NRC subsequently issued a Request for Additional Information (RAI) in the Reference 2 letter and we provided our response to the RAI in the Reference 3 letter. The NRC issued a second RAI in the Reference 4 letter. The RAI letter requested that additional information be provided within 30 days after receipt of the letter (i.e., by July 2, 2001). The requested additional information is provided in the Attachment and Enclosures to this letter.

Should you have any questions concerning this letter, please contact Ms. Kelly M. Root at (630) 657-2820.

Respectfully,

R. M. Krich
Director - Licensing
Mid-West Regional Operating Group

Attachment: Response to Request for Additional Information for Technical Specifications Change to Revise Steam Generator Inspection Frequency for the Fall 2001 Refueling Outage for Braidwood Station, Unit 1

Enclosures:
1. Technical Justification for the Braidwood Station, Unit 1 and the Byron Station, Unit 1 Replacement Steam Generators Plus-Point Eddy Current Inspection Scope
2. Technical Justification for using Dual Automated Eddy Current Systems for the Braidwood Station, Unit 1 and the Byron Station, Unit 1 Replacement Steam Generators

cc: Regional Administrator - NRC Region III
NRC Senior Resident Inspector - Braidwood Station
Office of Nuclear Facility Safety - Illinois Department of Nuclear Safety

Attachment

Response to Request for Additional Information for Technical Specifications Change to Revise Steam Generator Inspection Frequency for the Fall 2001 Refueling Outage for Braidwood Station, Unit 1

QUESTION 1

"Please provide the scope of MRPC [multiple rotating pancake coil] examinations and a summary [and a summary (sic)] of results of previous inspections (including pre-service inspections) of the replacement SGs during which MRPC had been performed."

RESPONSE TO QUESTION 1

During the baseline pre-service inspection (PSI) of the Braidwood Station, Unit 1 Steam Generators (SGs), Plus Point inspection was performed on 100% of the tightest radius (i.e., row 3) U-bends in all four SGs. No defects were identified.

During the baseline PSI, 100% of the tubing in all four SGs received full-length bobbin coil inspection. In accordance with the Exelon Generation Company, (EGC) LLC data analysis guidelines and the Electric Power Research Institute (EPRI) Pressurized Water Reactor (PWR) SG Examination Guidelines, bobbin coil signals which were not clearly discernable required additional inspection using the Plus Point probe to provide a basis for signal characterization. A total of 54 bobbin signals were inspected with the Plus Point probe. The Plus Point probe either classified all 54 locations as no defect found (NDF) or manufactures buff mark (MBM) and no further actions were required.

As part of the baseline PSI bobbin coil inspection, tubesheet profilometry was performed on all hot leg and cold leg tubesheet expansions. Tubesheet profilometry provides a method to measure the tube expansion within the tubesheet and also determine the roll transition region relative to the top of the secondary face of the tubesheet. No tubes were identified as having conditions requiring repair, and no over expansions beyond the secondary face of the tubesheet were identified.

During the Braidwood Station, Unit 1 spring 2000 refueling outage, 100% of the tubing in all four SGs received full-length bobbin coil inspection. In accordance with the EGC data analysis guidelines and the EPRI PWR SG Examination Guidelines, bobbin coil signals which were not clearly discernable required additional inspection using the Plus Point probe. A total of eight bobbin signals were inspected with the Plus Point probe. Of the eight signals inspected with the Plus Point probe, seven were classified as NDF or MBM, and one was classified as volumetric wear < 10% through-wall at a fan bar location. The tube with fan bar wear indication was removed from service by mechanical plugging and is discussed in detail in the Reference 1 letter, Attachment A, "Braidwood Station, Unit 1, Description and Safety Analysis of the Proposed Change," Section F, "Safety Analysis of the Proposed Changes."

QUESTION 2

"Page 1, Paragraph 3, of the response referenced a "technical justification" developed by Exelon for the exception taken to the EPRI SG Examination Guidelines. Please provide the staff a copy of the 'technical justification.'"

RESPONSE TO QUESTION 2

In our response to question 1 in the Reference 2 letter, we provided a summary of the technical justification to support not performing inspections in the hot leg top of tubesheet (TTS) and the tightest radius (i.e., row 3) U-bend regions using a technique qualified to detect stress corrosion cracking (SCC) within the first 60 effective full power months (EFPMs) of replacement SG operation. Enclosure 1 contains a copy of our technical justification as written and approved in December 1999. The enclosed technical justification contains a list of industry experience for SGs containing thermally treated Inconel-690 tubes, which was current at the time the technical justification was written. As part of our response to question 2 in the Reference 2 letter, an updated list of industry experience for SGs containing thermally treated Inconel-690 tubes was provided. The enclosed technical justification also refers to the Braidwood Station, Unit 2 and the Byron Station, Unit 2 operating experience at the time the technical justification was written. Since that time, additional operating experience has occurred and is reflected in our response to question 1 in the Reference 2 letter.

QUESTION 3

"The response stated that only bobbin coils will be used during its eddy current inspections. (a) For the low-row U-bends in the replacement SGs, specify what probes are qualified to detect specific types of degradation. (b) If MRPCs were not used for the TTS and low row U-bends during the previous outage, please explain how the inspection met 10 CFR50, Appendix B, requirements. 10 CFR50 Appendix B requires non-destructive testing to be controlled by "qualified personnel using qualified procedures". (c) How would Exelon acquire operating experience concerning TTS and low-row U-bends if other plants are taking the same exception? Please elaborate on data available for your SGs. If operating experience for 600TT tubes is relied upon, how are differences in design and operation accounted for. (d) If these MRPC exams are not scheduled in the first 60 EFPMs, when would they be performed?"

RESPONSE TO QUESTION 3a

In accordance with a teleconference we had with representatives of the NRC on June 19, 2001, it was agreed that no response is necessary.

RESPONSE TO QUESTION 3b

Prior to the Braidwood Station, Unit 1 spring 2000 refueling outage, a degradation assessment was performed in accordance with the EPRI Steam Generator Integrity Assessment Guidelines and EGC SG Program procedures. The degradation assessment takes into consideration SG design, operating conditions, and operating experience of similarly designed SGs throughout the industry. The degradation assessment identifies active and potential degradation mechanisms and determines inspection scope and techniques required to be performed during the upcoming outage. Inspection techniques qualified in accordance with the EPRI PWR SG Examination Guidelines, Appendix H, "Performance Demonstration for Eddy Current Examination," are then chosen to detect active and potential degradation mechanisms identified in the degradation assessment.

The degradation assessment performed prior to the Braidwood Station, Unit 1 spring 2000 refueling outage classified SCC of the TTS roll transition region and the tightest radius (i.e., row 3) U-bend region as neither an active or potential degradation mechanism. In addition, a technical justification that supports the basis that SCC will not develop within the first 60 EFPMs of operation was written and is provided in Enclosure 1.

During the Braidwood Station, Unit 1 spring 2000 refueling outage, EPRI PWR SG Examination Guidelines, Appendix H, qualified techniques were used to inspect for all potential damage mechanisms as identified in the degradation assessment. All data acquisition and analysis personnel were certified in accordance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components." In addition, all data analysis personnel were qualified in accordance with EPRI PWR SG Examination Guidelines, Appendix G, "Qualification of Nondestructive Examination Personnel for Analysis of NDE Data." Therefore, all 10 CRF 50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," requirements were met.

RESPONSE TO QUESTION 3c

In accordance with a teleconference we had with representatives of the NRC on June 19, 2001, it was agreed that no response is necessary for the tightest radius (i.e., row 3) U-bend regions. Therefore, the following response applies to the TTS region only.

As stated in our response to question 2 in the Reference 2 letter, in a comparison of the Braidwood Station, Unit 1 replacement SG hot leg operating temperature to other units containing thermally treated Inconel-690 tubing, 34 of the 54 units have a higher hot leg temperature than Braidwood Station, Unit 1 and the secondary water chemistry program at the Braidwood Station is similar to that of other units containing thermally treated Inconel-690 tubing. In addition, in a comparison of the Braidwood Station, Unit 1 replacement SG operating time in effective full power years (EFPYs) to other units containing thermally treated Inconel-690 tubing, 38 of the 54 units have more operating time than Braidwood Station, Unit 1and would be expected to experience SG degradation prior to the Braidwood Station, Unit 1 SGs.

In order to supplement the data available for plants operating with thermally treated Inconel-690 tubing, the Braidwood Station technical justification considers the inspection results from the Braidwood Station, Unit 2 SGs that contain thermally treated Inconel-600 tubing. A comparison of the thermally treated Inconel-690 tubing to the thermally treated Inconel-600 tubing is provided in our technical justification contained in Enclosure 1. In all aspects of corrosion resistance, the thermally treated Inconel-690 tubing is superior to the thermally treated Inconel-600 tubing. As detailed in our technical justification, performance of the Braidwood Station, Unit 2 thermally treated tubing provides reasonable assurance that the Braidwood Station, Unit 1 thermally treated Inconel-690 tubing will not develop SCC early in the life of the SGs.

The current Braidwood Station, Units 1 and 2 operating conditions are essentially identical. Both units operate at a hot leg temperature of 610°F. The secondary water chemistry programs are essentially identical, with the exception that Braidwood Station, Unit 2 began Molar Ratio Control in February 2001.

In addition to the information supplied in our response to question 1 in the Reference 2 letter, the following table provides multiple rotating pancake coil (MRPC) and Plus Point (i.e., +PT) inspection data for the Braidwood Station, Unit 2 SGs.

Braidwood Station, Unit 2
Hot Leg Top of Tubesheet Inspection Data

Outage	Inspection Scope / Probe	Comments
Spring 1996 outage (5.8 EFPYs / 70 EFPMs)	25% / MRPC	
Fall 1997 outage (7.1 EFPYs / 85 EFPMs)	100% / +PT	15 TTS indications removed from service. Determined to be irrelevant based on Byron Unit 2 tube pull*
Spring 1999 outage (8.5 EFPYs / 102 EFPMs)	25% / +PT	
Fall 2000 outage (9.9 EFPYs / 118 EFPMs)	50% / +PT	

(*) Byron Station pulled three tubes during the spring 1998 outage (8.6 EFPYs / 103 EFPMs). Metallurgical analysis showed no signs of SCC or intergranular attack (IGA).

As can be seen from the above data, the Braidwood Station, Units 1 and 2 operating conditions are very similar. There is a significant amount of inspection data available for the Braidwood Station, Unit 2 SGs that support the conclusion that SCC will not develop early in the life of the Braidwood Station, Unit 1 SGs.

RESPONSE TO QUESTION 3d

The technical justification provided in Enclosure 1 provides the basis for not performing TTS or U-bend inspections with a technique qualified to detect SCC for the first 60 EFPMs of replacement SG operation. When the current technical justification expires, i.e. after the first 60 EFPMs of operation, EGC will evaluate the available industry data, including performance of thermally treated Inconel-600 tubing, along with available industry guidance, and develop an inspection program for these regions of the replacements SGs.

QUESTION 4

"Regarding page 4 of the response, Analysis Quality Checks Section, what is the technical justification for only performing 20% of the manual review instead of 100% as required by the Guidelines? If there was a technical justification performed for the previous outage, please provide the staff a copy."

RESPONSE TO QUESTION 4

In our response to question 1 in the Reference 2 letter, we provided a summary of the technical justification for the use of both primary and secondary automated data analysis for inspection of the replacement SGs. Enclosure 2 contains a copy of our technical justification as written and approved in March 2000.

In addition to the data contained in the enclosed technical justification, the following information is being provided.

- Both automated systems were used in the interactive mode in which both the primary and secondary analysts reviewed 100% of the calls identified by the computer and validated the computer call with their own analysis. This meets the requirement as defined in the EPRI guidelines, "Both teams may use automated analysis. However, the automated analysis results must be verified by at least one of the two teams."

- The fact that Braidwood Station, Unit 1 replacement SG tubing has a bobbin coil average signal to noise ratio of greater than 35 to one along with the SCC resistance inherent with thermally treated Inconel-690 tubing provide reasonable assurance that automated data analysis systems would detect all potential degradation.

The possibility of a missed indication has been reduced through qualification, demonstration, and manual random data review. The intent of the PWR SG Examination Guidelines is also met and exceeded in some instances. Through implementation of the actions identified above it was determined that manual review of all data during the Braidwood Station, Unit 1 spring 2000 outage was not required and is documented as such in our technical justification.

REFERENCES

1. Letter from R. M. Krich (Exelon Generation Company, LLC) to US NRC, "Request for Technical Specifications Change Braidwood Station, Unit 1, Steam Generator Inspection Frequency Revision for the Fall 2001 Refueling Outage," dated February 9, 2001

2. Letter from R. M. Krich (Exelon Generation Company, LLC) to US NRC, "Response to Request for Additional Information for Technical Specifications Change to Revise Steam Generator Inspection Frequency for the Fall 2001 Refueling Outage for Braidwood Station, Unit 1," dated May 18, 2001

Enclosure 1

Technical Justification for the Braidwood Station, Unit 1 and the Byron Station, Unit 1 Replacement Steam Generators Plus-Point Eddy Current Inspection Scope

Memorandum

ComEd

December 16, 1999
ED-BRW-99-0205

To: File

From: Mike Sears / Jay Smith

Subject: Braidwood Unit 1 and Byron Unit 1 BWI Replacement Steam Generator Plus-Point Eddy Current Inspection Scope

The steam generator inspection scope requirements for replacement steam generators are specified in Revision 5 of the EPRI Steam Generator Examination Guidelines and are also reiterated in Section 4.5.1.1 of CWPI-NSP-ER-20-1, "Conduct of Steam Generator Management Program Activities". In part, these documents require a 100% full length bobbin coil inspection, a 20% plus-point inspection at the top of the tubesheet and a 20% plus-point inspection of the low row U-Bends at the end of the first operating period following steam generator replacement. Inspections totaling 100% of these regions are also required on a 60 Effective Full Power Month (E.F.P.M.) frequency thereafter.

Both the EPRI Guidelines and CWPI-NSP-ER-20-1 procedure have provisions that allow requirement exceptions, providing that a technical justification is performed. Attachment 1 provides the technical justification to eliminate the plus-point inspection of the top of tubesheet expansion and low row U-bends for the first 60 E.F.P.M. of operation following steam generator replacement. The full-length bobbin inspection in each steam generator as required by the EPRI Guidelines and CWPI-NSP-ER-20-1 will be performed in accordance with the requirements of these documents. This technical justification is required to be reviewed prior to each outage as part of the degradation assessment. Review shall include assessment of Alloy 690 and Alloy 600TT tubing performance based on performance within ComEd units and the industry to determine if conditions warrant inspection.

Should there be any questions regarding this matter, please contact Mike Sears on Braidwood extension 2251, Jay Smith on Byron extension 2604, or Roman Gesior on Downers Grove extension 7671.

Prepared by ____________________
Braidwood SG Engineer

Prepared by ____________________
Byron SG Engineer

Approved by ____________________
Braidwood SEM

Approved by ____________________
Byron SEM

Approved by ____________________
Engineering Programs Chief

Attachment

Cc:	M. Sears	
	M. Smith	
	H. Smith	(DG NES)
	R. Gesior	(DG NES)
	J. Smith	(Byron SEC)
	G. Contrady	(Byron SEC)

Attachment 1

Technical Justification for Braidwood Unit 1 and Byron Unit 1
BWI Replacement Steam Generator Inspection Scope

Technical justification is provided to eliminate the Braidwood Unit 1 and Byron Unit 1 replacement steam generators top-of-the-tube-sheet (TTS) and low row U-bends Plus Point inspection for the following periods:

- After the first cycle of operation.
- During the first 60 E.F.P.M. of operation. This equates to operation through A1R11 for Braidwood Unit 1 and B1R12 for Byron Unit 1.

Inspection Requirements:

NEI 97-06, Steam Generator Program Guidelines require performing a degradation assessment to identify active and potential degradation mechanisms prior to each outage. The degradation assessment is performed to identify active and potential damage mechanisms that may be encountered during the upcoming inspection. The degradation assessment is also designed to choose techniques to test for these degradation mechanisms based upon technique qualification, probability of detection, sizing capability and to establish the number of tubes required to be inspected during the upcoming outage. The degradation assessment encompasses the inspection requirements contained in the EPRI PWR Steam Generator Examination Guidelines and ComEd SG Program CWPI-NSP-ER-20-1. Any exception to the requirements requires a technical justification with appropriate approval.

The following is a listing of the EPRI guidelines, ComEd SG Program CWPI-NSP-ER-20-1 inspection requirements and exceptions being taken to those requirements for the first 60 E.F.P.M. following steam generator replacement.

<u>EPRI PWR Steam Generator Examination Guidelines Revision 5 Volume 1, Section 3.3.1</u>:

1. After the first cycle of operation for either new or replacement steam generators, a 100% full length examination using general purpose eddy current probes shall be performed on all steam generators.
2. During subsequent ISIs, if active damage mechanisms are identified, all steam generators shall be examined at the end of each fuel cycle or 24 E.F.P.M., whichever is less, or as necessary to satisfy published regulatory requirements.
3. During subsequent ISIs, if active damage mechanisms are not identified, the number of steam generators to be examined and/or the frequency of examination , shall be performed as required by Section 3.3.2

4. 100% of tubing and 100% of each type of repair shall be inspected within a rolling 60 E.F.P.M. time frame. If 60 E.F.P.M. occurs during an operating cycle completion of that cycle is acceptable and is within the stated requirement.
5. No steam generator shall operate more than two fuel cycles between inspections.

ComEd will be taking exception to the requirements in items 3 and 4 as discussed below for the Plus Point low row U-bend and TTS inspections.

EPRI PWR Steam Generator Examination Guidelines Revision 5 Volume 1, Section 3.3.2 states:

If the steam generators are free from active damage mechanisms, some latitude is provided in terms of the number of steam generators to be inspected and/or frequency of inspection. For these steam generators, any of the following three inspection options may be performed:

1. Inspect >= 20% of the tubes and >= 20% of each type of repair in each steam generator at each refueling outage (RFO), or
2. Inspect >= 40% of the tubes and >= 40% of each type of repair in half the number of steam generators at each RFO, or
3. Inspect >= 40% of the tubes and >= 40% of each type of repair in each steam generator at every other RFO.

Regardless of the inspection option chosen, no steam generator shall operate more than two refuel cycles between inspections. The scope of the inspection is defined as using a qualified ECT technique for the entire hot leg length of the tube and the U-bend.

ComEd will be taking exception to this sampling requirement by not inspecting the TTS or low row U-bends with the Plus Point probe at the given frequency or sampling discussed above.

EPRI PWR Steam Generator Examination Guidelines Revision 5 Volume 1, Section 3.7 States:

Accommodation to the expected improved service experience of second generation steam generators should be exercised with due attention paid to differences in tube material and corrosive effects of the chemical environments in the secondary system. Among the second generation steam generators, some were tubed with thermally treated Alloy 600 and others with Alloy 690 or Alloy 800. Though all of these exhibit greater resistance to PWSCC and OD corrosion mechanisms, laboratory data suggest that Alloy 690 and Alloy 800 are distinctly more resistant than Alloy 600.

While the tube sampling program of this document does not differentiate between first and second generation steam generators, it is recognized that utilities with second generation steam generators may be able to provide technical justification for a period longer than 60 E.F.P.M. to achieve 100% inspection. In this case, the requirement that no steam generator shall operate for more than two cycles without inspection and the requirement for a minimum 20% periodic sample shall still apply.

ComEd will be taking exception to the 20% periodic sample using the Plus Point probe in the low row U-bend and TTS regions.

PWR Steam Generator Examination Guidelines, Revision 5 Volume 2, Basis, Section 3.7 States:

Some licensees may choose to justify inspection programs which depart from the recommendations of Volume 1 Section 3. The experience of the advanced steam generator designs and the corrosion resistance of improved materials provide the basis for less stringent inspection criteria. In such circumstances, the discovery of active tube degradation mechanisms during an inspection outage signals the need for sampling strategies consistent with first generation steam generators.

ComEd SG Program CWPI-NSP-ER-20-1 Section 4.5.1.1 contains the following tube inspection scope requirements:

1. 100% full length bobbin inspection (after the first inservice inspection 25% sample is required)
2. 25% HL TTS inspection with Plus Point probe
3. 25% tight radius U-bend inspection with Plus Point
4. 25% HL dents and dings > 5.0 Volts with Plus Point
5. 25% Plus Point of installed plugs (if design allows)
6. 25% Visual inspection of installed plugs (if not design will not permit Plus Point inspection)
7. 100% Visual inspection of newly installed plugs

ComEd will be taking exception to the requirements in items 2 and 3 for the first 60 E.F.P.M. of operation after steam generator replacement.

Assessment:

The purpose of this technical justification is to evaluate the elimination of the TTS and low row U-bends Plus Point inspection for Braidwood Unit 1 and Byron Unit 1 for the first 60 E.F.P.M. of operation following steam generator replacement.

The HL TTS and low row U-bend sampling inspection is performed to detect IGA/ODSCC/PWSCC degradation. The Plus Point probe is necessary to detect this form of degradation if it is active or potentially active because of the poor probability of detection using the bobbin probe, particularly in the circumferential direction. An assessment was performed to identify the susceptibility of the replacement steam generators to this type of degradation and establish whether this form of degradation is active or potentially active in similarly designed steam generators.

Improved Tube Material:

The Braidwood 1 and Byron 1 replacement steam generators contain tubing fabricated from thermally treated Alloy 690. The development of Alloy 690 was driven by the failure of Alloy 600 in primary and secondary side water environments and the need for SCC resistant tubing materials. Alloy 690 tubing corrosion behavior was extensively studied and tested by tubing manufacturers, steam generator manufacturers, utilities and Industry groups, such as EPRI. After nearly 10 years of research, Alloy 690 was accepted as the best steam generator tubing available.

Resistance to SCC in Alloy 690 material is accomplished by a higher chromium content than Alloy 600. Alloy 690 contains 27-31% chromium, while Alloy 600 contains only 14-17%. The higher chromium content reduces the degree of sensitization (i.e., the amount of chromium depleted in areas adjacent to the grain boundaries), thus increasing resistance to corrosion attack at the grain boundaries. Grain boundary corrosion is also reduced with the precipitation of carbides in the grain boundaries. Proper mill annealing at temperatures greater than 1940 °F followed by thermal treatment maximizes the precipitation of carbides into the grain boundaries.

The superiority of Alloy 690 over Alloy 600 to SCC is demonstrated by extensive corrosion tests, as documented by BWC Report "Replacement Steam Generators Tube Survivability Report", (Reference 13). The tubing was subject to accelerated corrosion tests that used high temperature, high contaminant environments and high stressed tubing in the attempt to crack the tubing. Alloy 600 proved to be susceptible to cracking in nearly every type of environment tested. Cracking occurred in Alloy 690 only in a high caustic-high lead environment at temperatures of 620-630 °F. This environment is outside the limits of normal steam generator operation. Therefore, thermally treated Alloy 690 tubing is demonstrated to provide a significantly more resistance to SCC over mill annealed Alloy 600.

Improved Recirculation Ratio and Flow Velocities:

Recirculation flow is defined as the ratio of the riser mass flow rate versus the steam outlet flow rate. By maximizing the recirculation ratio, secondary side concerns regarding deposit loading, corrosion product transfer, tube dry out and sludge management can be alleviated. Maintaining a high recirculation ratio encourages the secondary bulk water contaminants to remain in suspension, thus benefiting the effectiveness of blowdown cleanup and reducing sludge pile height on the tubesheet. A high recirculation ratio also minimizes the potential of low flow areas, where impurity hideout may occur to produce harmful micro-environments. By reducing deposit/sludge loadings and minimizes low flow areas, the chemical environment that can promote SCC initiation and growth is also minimized.

The RSGs are designed with a recirculation ratio of 5.76 (Reference 13). This is significantly higher than the Model D-5 recirculation ratio of 3.2 (Reference 14, Table 2.1). Therefore, the RSGs are expected to perform better than the D-5s in regards to benefiting flow velocities and circulation.

Top-of-the-tube-sheet Region:

The basis of this assessment is that the Westinghouse Model D5 Steam Generators at Byron and Braidwood Unit 2 have operated greater than 8 E.F.P.Y. without detecting IGA/ODSCC/PWSCC degradation at the TTS. This includes a 100% Plus Point inspection at both Byron and Braidwood Unit 2 and removal of 3 tubes from Byron Unit 2. The tube pull results indicated that there was no IGA or SCC at the TTS location of the 3 tubes analyzed. Byron 2 has operated over 10 E.F.P.Y. without detecting IGA or SCC. Braidwood 2 has operated over 8 E.F.P.Y. without detecting IGA or SCC in the TTS region.

Comparison of designs between the replacement steam generators (RSG) and the Westinghouse model D5 steam generators was performed to assess the susceptibility of the RSG at the TTS region compared to the Model D5.

The primary improvement of the RSG is tube material (i.e. Alloy 690 TT vs. Alloy 600 TT). The benefits of using Alloy 690 TT and its improved resistance to IGA and SCC over Alloy 600 TT in BWC Report "Replacement Steam Generators Tube Survivability Report", (Reference 13), and proceedings from the 1989 EPRI Alloy 690 Workshop (Reference 1).

Both steam generator models (RSG's and D5) use a hydraulic expansion. Precautions were taken in the design of the RSG's (Reference 3) to minimize the residual stress in the hydraulic expansion to levels below 20 ksi on the ID and OD (Reference 4). Qualification testing of the Tube to Tube-sheet joint found the residual stress to be on the order of 13 ksi in the hoop direction (Reference 9, Table 6) and the total operating stress at the TTS to be 23.5 ksi in the hoop direction and 20.3 ksi in the axial direction. All tubes identified during the pre-service ECT inspection to have been expanded above acceptance criteria were repaired. The TTS total operating stress in the axial direction for the Unit 1 RSG's (20.3 ksi) is comparable to that estimated for the Unit 2 model D5 hydraulic expansions of 15 ksi on the OD (Reference 8).

The RSG's have a feedring compared to the D5 that has a pre-heater, which has been reported by some to contribute to the increased degradation of steam generators (Reference 2) at the TTS. The chemistry program and operating temperatures of the two units are similar.

U-Bend Region:

The basis of this assessment is that the Westinghouse Model D5 Steam Generators at Byron and Braidwood Unit 2 have operated 5.84 EFPY before detecting a U-bend indication with the characteristics of IGA/ODSCC/PWSCC. One U-bend indication with the characteristics of cracking has been detected to date at Braidwood Unit 2, a tube was not pulled to confirm the degradation. A 100% Row 1 and 2 U-bend Plus Point inspection has been performed at both Byron and Braidwood Unit 2. Byron 2 has operated over 10 E.F.P.Y. without detecting IGA or SCC in the Row 1&2 tight radius U-bends.

Comparison of designs between the replacement steam generators (RSG) and the Westinghouse model D5 steam generators was performed to assess the susceptibility of the RSG at the U-bend region compared to the Model D5.

The RSG's tightest radius U-bend has a minimum radius of 3.632" (Reference 6) compared to the D5 radius of 2.25" (Reference 7) and both steam generator models received a thermal stress relief (Reference 10 & 11) after the tubes were bent. Residual stresses in the RSG's U-bend after stress relief is negligible (Reference 12).

The chemistry program and operating temperatures of the two units are similar.

Industry Experience:

A review was performed on the industry experience with Alloy 690 TT tubing. A summary of plants experience is provided in Attachment 2. Although the operating experience is not significant there are 9 plants with experience beyond 5 E.F.P.Y., with the oldest at 7.8 E.F.P.Y. No plants have experienced IGA or SCC in Alloy 690 TT tubing.

Reference 2 contains degradation predictions for D5 steam generators with Alloy 600 TT tubing. In this report, it is predicted that stress corrosion cracking is expected to occur at 8 – 10 E.F.P.Y. when operated at a Thot of 618^{0}F. The design and material improvements of the RSGs with Alloy 690 TT tubing is expected to be more resistant than the D5's to stress corrosion cracking and therefore, its initiation is expected to be well beyond 8 – 10 E.F.P.Y..

Conclusion:

With the design improvements of the RSG's, TTS and low row U-bend degradation for which the Bobbin probe has a poor probability of detection (IGA/ODSCC/PWSCC) is bound by the degradation in the Westinghouse Model D5 steam generators with Alloy 600 TT tubing. Degradation (IGA/ODSCC/PWSCC) in the RSG's is not expected within the first 60 E.F.P.M. of operation based on the experience of the Model D5 steam generators and therefore Plus Point inspection of the TTS and low row U-bends is not required. However, because of the uncertainties in SG tubing degradation it is necessary to stay abreast of the degradation experience in other plants with Alloy 600 TT and Alloy 690 TT tubing and maintain a proactive inspection plan. Additionally, inspections for other types of potential and active degradation mechanisms (i.e. loose parts and fan bar wear) must meet the requirements of the EPRI guidelines stated above. This technical justification is required to be reviewed prior to each outage as part of the degradation assessment. Review shall include assessment of Alloy 690 and Alloy 600TT tubing performance based on performance within ComEd units and the industry to determine if conditions warrant inspection.

References:

1. Proceedings: 1989 EPRI Alloy 690 Workshop, EPRI NP-6750-M and NP-6750-SD, April 1990
2. Predicted Tube Degradation for Westinghouse Models D5 and F Type Steam Generators, EPRI TR-108501, September 1997
3. Design Specification for Replacement Steam Generator, 18-1229648
4. Technical Specification for Replacement Steam Generator, December 9, 1991, Revision 5, Section 304.9.9.
5. Steam Generator Progress Report, EPRI TR-106365-R12, October 1997
6. BWI Drawing 7720D112 Rev. 00
7. Westinghouse Technical Manual, 1440-C312, Rev. 2
8. Causes and Occurrences of Circumferential Cracking in PWR Steam Generator Alloy 600 Tubing, Draft EPRI Report, Dominion Engineering, February 1996
9. Tube-to-Tube-sheet Joint Qualification: Program Summary, BWI-TR-95-05, Rev. 02, September 1996
10. Letter from J.L. Tain to J.D. Deress, Byron and Braidwood Stations Unit 1 and 2 Steam Generator Tubing Material Certification, CAE/CCE-120, June 13, 1985
11. Technical Specification for Nickel-Chromium-Iron (Alloy 690) Nuclear Steam Generator Quality Tubing, TS-8905, Rev. 04, June 13, 1996
12. BWI Nuclear SG Tube Business, ComEd- S/R of Tight Radius Tubes, October 20, 1995
13. BWI Report, "Replacement Steam Generators Tube Survivability Report", 222-7720-PR-04, Revision 0, April 1997
14. Westinghouse Report, "Steam Generator Information Report", SG-90-02-026, Revision 6, February 1990.

Attachment 2
Industry Experience w/ Alloy 690 Tube Material

Plant	Tube	Manuf	Model	EFPY*	Temp*	Degradation
Byron-1	I-690TT	BWI	7720	1.0	610	None
Braidwood-1	I-690TT	BWI	7720	1	610	1st ISI Spring '00
Beznau-1	I-690TT	Fram	33/19	5.2	594	None
**Catawba-1	I-690TT	BWI	CFR-80	1.1	613	Preventive (19)
Chooz B1	I-690TT	Fram	7319		625	Other Mech (2)
Chooz B2	I-690TT	Fram	7319		625	Other Mech (2)
Cook-2	I-690TT	West.	51F	5.8	606	Wear TSP (1) Other Mech (8)
Dampierre-1	I-690TT	Fram	51B	7.8	613	Wear AVB (2)
Dampierre-3	I-690TT	Fram	4722	1	613	Other Mech (2)
Doel 4	I-690TT	Fram	7919	0.5	621	Other Mech (4)
Genkai-1	I-690TT	MHI	52F	3.9	613	None
Genkai-3	I-690TT	MHI	52FA	4.3	617	None
Genkai-4	I-690TT	MHI	52FA	1.2	617	None
**Ginna	I-690TT	BWI	RSG	1.5	589	Other Mech (2)
Golffech-2	I-690TT	Fram	6819	4.4	616	Preventive (2) Other Mech (5)
Gravelines-1	I-690TT	Fram	4722	2.2	613	Other Mech (5)
Gravelines-2	I-690TT	Fram	4722	1	613	Other Mech (2)
Ikata-1	I-690TT	MHI	51	0.5	605	None
Ikata-3	I-690TT	MHI	52F	3.3	613	
Indian Point-3	I-690TT	West.	44F	4.4	597	None
Kori-1	I-690TT	West	D60	0.4	607	None
**McGuire-1	I-690TT	BWI	RSG	1.5	618	Preventive (10) Fan Bar Wear (2 Ind)
**McGuire-2	I-690TT	BWI	RSG	1.5	618	Other Mech (2)
Mihama-1	I-690TT	West	35F	2.3	603	None
Mihama-2	I-690TT	MHI	46F	3.2	607	None
Mihama-3	I-690TT	MHI	54F	1.8	608	None
Millstone-2	I-690TT	BWI	RSG	1.8	596	Other Mech (2)
North Anna-1	I-690TT	West.	54F	5.0	613	Preventive (1)
North Anna-2	I-690TT	West.	54F	2.6	613	None
Ohi-1	I-690TT	MHI	52FA	2.1	617	None
Ohi-2	I-690TT	MHI	54FA	0.8	613	
Ohi-3	I-690TT	MHI	52FA	6.4	617	None
Ohi-4	I-690TT	MHI	52FA	5.2	617	None
Penly-2	I-690TT	Fram	6819	5.2	616	Preventive (1) Other Mech (4) AVB Wear (1)
Point Beach-2	I-690TT	West	D47F	0.9	597	None
Ringhals-2	I-690TT	KWU	RSG	6.1	610	

Ringhals-3	I-690TT	KWU	RSG	2.6		None
Sizewell B	I-690TT	West	F	2.6	617	None
St.Laurent Des Eaux B1	I-690TT	Fram	4722	5.5	613	Other Mech (6)
**St. Lucie-1	I-690TT	BWI	RSG	1.2	599	Wear at TSP (16)
Summer	I-690TT	West.	Delta-75	2.7	619	Other Mech (3)
Takahama-1	I-690TT	MHI	54F	1.9	613	None
Takahama-2	I-690TT	MHI	52F	3.3	613	None
Tihange-1	I-690TT	MHI	RSG	1.3	609	Other Mech (2)
Tihange-3	I-690TT	Fram	7919	0.4	623	
Tricastin-1	I-690TT	Fram	4722		613	
Tricastin-2	I-690TT	Fram	4722	1	613	Other Mech (1)

* EFPY as report in the October 1997 EPRI Steam Generator Progress Report, Revision 14

** Unit contains BWI Replacement Steam Generators of similar design of the Byron Unit 1 steam generators.

Enclosure 2

Technical Justification for using Dual Automated Eddy Current Systems for the Braidwood Station, Unit 1 and the Byron Station, Unit 1 Replacement Steam Generators

MEMORANDUM

ComEd

March 2, 2000
Letter # DG00-000228

To: File

Subject: Technical Justification for using Dual Automated Eddy Current Systems in Byron and Braidwood Unit 1 Replacement Steam Generators

This technical justification is applicable for all Bobbin Coil eddy current examinations in Thermally Treated Inconel 690 tubing in replacement steam generators at Byron and Braidwood Unit 1. This technical justification is written to support automated analysis of steam generator tube eddy current data by both the primary and secondary analysis teams. Automated analysis has been used in the past by one of the primary or secondary analysis teams but not both.

Deviation Description

The EPRI PWR Steam Generator Examination Guidelines (Reference 1) Section 6.3.3.3 requires a manual analysis of 100% of the eddy current data in addition to the dual automated analysis to identify potential degradation mechanisms that the automated systems were not programmed to detect. An example might be absolute drift for IGA for which the possibility of the occurrence plays a major role in older non-replaced I-600 steam generators but not for replaced steam generators with I-690 TT tubing which is not as susceptible to the degradation mechanism. The automated analysis systems will be qualified and tested to all the potential damage mechanisms in the Byron and Braidwood Unit 1 steam generators. The deviation from the EPRI Guidelines Section 6.3.3.3, for which this technical justification is being written, is that a 100% manual analysis of all the data in addition to the dual automated analysis will not be performed.

Based upon the degradation assessment for A1R08 and B1R10 (Reference 2 and 4) the potential degradation mechanisms, based on industry experience, for the Byron and Braidwood Unit 1 steam generator design are:

- Fan bar and lattice grid wear
- Foreign object wear
- Tube to tube contact wear
- Dents and dings
- Manufacturing Burnish Marks (MBM)

Automated analysis systems have been used for many years on many different steam generator designs for the detection and analysis of steam generator tube degradation. The Braidwood A1R08 outage will be the first instance under Revision 5 of the PWR Steam Generator Examination Guidelines that automated analysis systems will be used for both primary and secondary analysis.

Presently automated analysis is not being proposed for rotating pancake coil (RPC) data.

<u>Approach for Dual Automated Systems</u>

Automated analysis systems must have equivalent or better reliability than manual methods. They must complete the EPRI guidelines (Reference 1) Appendix G, QDA examination of analysts for the type of data being evaluated (i.e. bobbin). The systems must be capable of passing a site specific performance demonstration test which facilitate sizing and/or detection of wear, loose parts, dents at a low level (2.5V and greater), and manufacturing burnish marks without user interaction once site testing has begun. The automated analysis will be implemented in the interactive mode in which the analyst reviews the calls identified by the computer and compares them with his own analysis of the call before the computer results are accepted (i.e. auto edited). As an overview of the automated system 20% of the strip chart data (10% primary and 10% secondary) will be reviewed by experienced analysts to ensure that degradation not detected by the automated analysis systems is detected. Additionally, calls made by the auto analysis primary and secondary analysis, which are discarded by manual analysis, require independent QDA sampling of approximately 20%. Final degradation sizing will result from the resolution process and not be based solely upon the automated analysis systems for condition monitoring and operational assessment applications.

Two different automated analysis systems will be applied for primary and secondary analysis to provide two independent detection/analysis schemes.

ComEd will use the Westinghouse ANSER Auto-Analysis Software Module (ADS) for primary analysis. ADS software is a rule based system for screening eddy current data. The rule based system is established by using information from the ComEd analysis guidelines, general sorts, and eddy current data from the steam generator tubes. Extraction parameters are set for the signal extraction algorithm. If a signal is detected using the established parameters it will be evaluated by the rule base for reporting. For signals that met the extraction parameters each line of the rule base will have a measurement classification (e.g. volts peak to peak, maximum rate, vertical maximum) that will be applied to the signal and if the first line criteria is met than the next line criteria will be applied. This continues until the signal is reported or disregarded.

ComEd will also use CoreStar AutoVISION software for secondary analysis. The major difference between AutoVISION and ADS is the signal extraction method employed. AutoVISION employs pattern recognition algorithms to extract signals of interest. Many algorithms are used to extract signals of interest in critical areas defined to be analyzed.

Pattern recognition based algorithms can distinguish relevant signals from extraneous background noise effects and "interpret" the data as a human analyst would. AutoVISION is similar to ADS in the signal dispositioning after it has been extracted using rule based applications in accordance with ComEd analysis guidelines.

Technical Justification Basis for Using Dual Auto Analysis Systems

Two different automated analysis systems will be used to provide independent primary and secondary analysis ensuring a high level of detection during the inspection. Both automated analysis systems have passed the appropriate EPRI Guidelines Appendix G, QDA examination (see Attachment 1 for results). Additionally, both systems have passed the site specific performance demonstration (SSPD) test that includes all potential degradation from the degradation assessment. The SSPD scores for the automated analysis systems demonstrate that the systems reliability is equivalent to or better than the manual analysis. The SSPD demonstrates that all potential degradation mechanisms are detectable with a qualified bobbin coil technique and are included in the sort routines for the auto analysis systems.

The SSPD includes fan bar/lattice grid wear flaws and localized degradation (pit-like) from similar steam generators that are small in size (smallest is 0.17 volts for fan bar wear and 0.25 volts for lattice grid wear). Appendix H qualified techniques will be used until indications are detected and site qualification in accordance with the EPRI guidelines (Reference 1) can be performed. Detection of these small flaws during the SSPD will demonstrate that the detection level of the automated analysis systems will ensure that no flaws will challenge the operational assessment limit of the tubes.

Any abnormal condition would be picked up by the 20% sample of the strip chart data by an experienced analyst performed on a random basis. This sampling scheme provides assurance that abnormal conditions will not go undetected provided a statistically significant number of conditions exist. It is recognized that these may be of a benign condition and of no consequence but awareness and documentation is the key element.

The ETSS's used for the automated analysis are the same as those for manual analysis and therefore a level of detection equivalent or better than manual analysis is expected. Final degradation sizing will result from the manual resolution process and not be based solely upon the automated analysis systems for condition monitoring and operational assessment applications. ComEd analysis guidelines will be implemented.

Application of automated detection/analysis will provide more consistent results and remove many human performance issues related to the inspection of steam generator tubes.

EPRI PWR Steam Generator Examination Guidelines Requirements

EPRI PWR Examination Guidelines Requirements for Automated Analysis:

Requirement	Resolution
Analysis methods should be consistent with the ETSS	Analysis will be performed using techniques, which have been site and/or industry qualified and are in accordance with the EPRI ETSS and degradation assessment
Computerized screening and data analysis of eddy current data is achieved by incorporating a detection-analysis rule base in software and allowing that rule base to interact with eddy current data	The automated analysis systems to be used will incorporate rule based and pattern recognition software that interacts with the eddy current data
Analysis logic used to establish the auto-screening processes shall be reviewed by a Qualified Data Analyst (QDA)	Analysis logic and data sorts will be reviewed by a Qualified Data Analyst (QDA) and be consistent with ComEd eddy current guidelines
Performance shall be demonstrated by the site-specific examination	A site specific performance demonstration was completed (see Attachment 2 for results)
Both teams must successfully complete a plant specific analysis performance demonstration	Both teams will be required to pass the SSPD examination.
It is recommended that the computer and individual analyst be measured separately	Analysts and the automated analysis systems will independently complete the SSPD examination
Computer detection and/or analysis may be used when demonstrated, by the plant specific performance demonstration, to be of equivalent or better reliability than manual methods	Prior to implementation the automated analysis system SSPD results must be demonstrated to be equivalent or better reliability than manual methods for potential degradation mechanisms as identified by the degradation assessment.
Both teams may use some form of computer-assisted screening. However, one of the systems shall use simple threshold detection with manual analysis for characterization. The second system shall use simple threshold for detection and either rule based or manual analysis for characterization	Two systems will be used that have independent methods for extraction/detection (pattern recognition and simple threshold). Ruled based methods will be used by both systems to disposition signals in accordance with ComEd guidelines. These systems represent the current state of technology for automated analysis.
Both teams may use automated analysis. However, the automated analysis results must be verified manually by at least one of the two teams	deviation from this requirement is being taken, evaluation of a 20% sample of strip chart data by a qualified data analyst will be performed to ensure degradation does not go undetected

Implementation Actions Required

1. Both automated analysis systems and all analysts must successfully complete the site specific performance demonstration with a minimum score of 80% that includes all potential degradation mechanisms (e.g. fan bar wear, lattice grid wear, loose parts, tube to tube contact and dents).
2. The SSPD shall be used to demonstrate that the automated analysis has equivalent or better reliability than manual methods for the potential degradation mechanisms identified in the degradation assessment. Analyst results on the SSPD will be evaluated against the automated analysis system to ensure that the automated analysis system has equivalent or better reliability than manual methods. Changes to the automated analysis system affecting detection and/or analysis capabilities will require SSPD examination completion.
3. The automated analysis logic and sorts used to establish the auto-screening processes shall be reviewed by a QDA.
4. Both automated analysis processes shall be accepted by the ComEd NDE Program Manager and Eddy Current Level III.
5. ComEd Analysis Guidelines shall be revised to reflect the automated analysis process and data flow.
6. The automated analysis systems shall be implemented in the interactive mode as defined in the EPRI guidelines which requires an analyst to review all automated analysis calls.
7. Evaluation of a 20% sample of strip chart data (approximately 10% primary and 10% secondary) must be reviewed by a qualified data analyst to ensure degradation not included in the data sorts does not go undetected.
8. Calls made by both the primary and secondary automated analysis systems which are discarded by manual analysis require independent QDA sampling of approximately 20%.
9. Analyst feedback must be implemented to ensure appropriate actions are taken to resolve missed indications in the automated analysis sorts and edit criteria.

Conclusion

With the actions outlined in this technical justification the consequences of a missed indication are minimized. The risk has been reduced through qualification, demonstration, and manual random review of data. The intent of the PWR SG Examination guidelines is also met and exceeded in some instances. This technical justification demonstrates that the objective and intent of the EPRI guidelines for inspection of steam generator tubes for potential degradation mechanisms is satisfied and the deviation from the EPRI PWR Steam Generator Examination Guidelines discussed above is acceptable. A review of license basis documents has been performed and no conflicts have been identified. The basis for this technical justification has been discussed and concurred with by Gary Henry of the EPRI NDE Center. This deviation is acceptable for use during future Byron and Braidwood Unit 1 inspections as determined in the degradation assessment given the same actions as outlined above are taken.

References

1. EPRI TR-107569-V1R5. PWR Steam Generator Examination Guidelines
2. Braidwood Letter, ED-BRW-99-0204, Braidwood Unit 1 Inspection Degradation Assessment and Condition Monitoring Checklist for A1R08.
3. Technical Justification Requirements on Appropriate Documentation for Deviation from NEI 97-06 Reference Documents
4. Byron Letter, 2000-5010, Byron Unit 1 Inspection Degradation Assessment and Condition Monitoring Checklist for B1R10.

Prepared by 3/2/00

Concurred with 3/2/00
NDE Program Manager and Level III

Approved by 3.8.00
Braidwood SEM

Approved by 3/3/00
Byron SEM

Approved by
NGG Engineering Programs Chief

Cc: M. Sears
J. Smith
R. Gesior
H. Smith
Central Files

Attachment 1

Qualified Data Analyst Performance Results

Analysis Technique	**Damage Mechanism**	**>40% TW**	**<40% TW[1]**	**Overcall Rate (all)**
ADS	Wear	12/12 (100%)	50/50 (100%)	4.51%
	Pitting	20/20 (100%)	19/19 (100%)	
AutoVISION[2]	Wear	12/12 (100%)	33/36 (92%)	4.84%
	Pitting	12/12 (100%)	12/12 (100%)	
Manual (286)	Wear	98%	96%	
	Pitting	88%	90%	

Note 1: The smallest flaw in the QDA Wear group is 11% TW and the smallest flaw in the QDA Pitting Group is 18% TW.
Note 2: The flaw sizes missed by AutoVISION were all less than 20%TW due to reporting thresholds established. The reporting thresholds have been modified, detection of small flaws on the SSPD will confirm the level of detection for small flaws. Many of the flaws (sized by industry qualified techniques) are below 20%TW.

Attachment 2

Site Specific Performance Demonstration Results

Damage Mechanism	Loose Parts	Tube to Tube Contact	Fan Bar Wear	Lattice Grid Wear	MBM
SSPD Truth Flaws	5	9	12	3	9
ADS Flaw Calls	5	9	12	3	9
ADS Score (%)	100	100	100	100	100
SSPD Truth Flaws	5	10	12	3	9
AutoVISION Flaw Calls	5	10	12	3	9
AutoVISION Score (%)	100	100	100	100	100

Note 1: Loose parts indications are signals from a loose part and do not represent loose part wear
Note 2: Fan Bar Wear indications includes a sample of localized and general wear type indications

Attachment 3

Design Basis Document Review

UFSAR:

The UFSAR generally states that steam generator inspection will be performed in accordance with Regulatory Guide 1.83 which is discussed below.

ASME Section XI, IWB-2413: States that plant Technical Specifications govern Steam Generator tube examination.

Technical Specification, Programs and Manuals, Section 5.5.9, Steam Generator Tube Surveillance Program:

- No requirements specific to analysis or level of detection

Regulatory Guide 1.83, Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes:

- Equipment must be capable of locating and identifying stress corrosion cracks, tube wall thinning by chemical wastage, mechanical damage or other causes (C.2.A).
- Inspection equipment must be sensitive enough to detect imperfections 20% or more throughwall (C.2.b).
- Personnel engaged in data taking and interpreting the results of the eddy current inspection should be tested and qualified in accordance with SNT-TC-1A (C.2.h).
- The examination should be performed to written procedures (C.2.i).

NEI 97-06, Steam Generator Program Guidelines:

- Requires inspections be performed in accordance with EPRI PWR Steam Generator Examination Guidelines
- Requires qualifying the inspection program by determining the accuracy and defining the elements for enhancing system performance, including technique, analysis, field analysis feedback, human performance and process controls.

NSP-ER-3020, CWPI-NSP-ER-20-1 and CWPI-NSP-ER-20.2.1:

NDT-E-3, Evaluation of eddy current data for steam generator tubing at Braidwood and Byron nuclear stations

- All data analysts must be certified in accordance with SNT-TC-1A

- All data analysts must have successfully completed the EPRI Qualified Data Analyst course and successfully completed the associated examinations
- All personnel interpreting data must have successfully passed with a grade of 80% or higher the ComEd Site Specific Performance Demonstration (SSPD) course.
- Any automated data screening software and equipment used shall be approved by ComEd.
- Screening of eddy current data with an automated data system shall be performed and monitored by QDA analysts. All indications reported and categorized by the automated data system shall be further analyzed by the QDA analyst to determine if reporting is required per the requirements of this procedure.

Analysis Guidelines:

- The dual automated analysis is being performed in accordance with the ComEd Analysis guidelines
- A revision to the guidelines is being drafted to specifically address automated analysis requirements.