Commonwealth Edison Company
1400 Opus Place
Downers Grove, IL 60515-5701

February 28, 1997

ComEd

United States Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Document Control Desk

Subject: Byron Nuclear Power Station, Units 1 & 2
Facility Operating Licenses NPF-37 & NPF-66
NRC Docket No. 50-454 and 50-455

Braidwood Nuclear Power Station, Units 1 & 2
Facility Operating Licenses NPF-72 & NPF-77
NRC Docket No. 50-456 and 50-457

"Steam Generator Tube Surveillance and Reactor Coolant System"

References:

1. J. Hosmer letter to the Nuclear Regulatory Commission dated January 31, 1997, transmitting Request to Amend Technical Specification Applicable to Reactor Coolant System

2. J. Hosmer letter to the Nuclear Regulatory Commission dated August 19, 1996, transmitting Request to Amend Technical Specification Applicable to the Renewal of the 3.0 Volt Interim Plugging Criteria

Pursuant to Title 10, Code of Federal Regulations, Part 50, Section 90 (10 CFR 50.90), Commonwealth Edison Company (ComEd) proposes to amend Appendix A, Technical Specifications, for Facility Operating Licenses NPF-37 and NPF-66 for Byron Nuclear Power Station, Units 1 & 2 and Facility Operating Licenses NPF-72 and NPF-77 for Braidwood Nuclear Power Station, Units 1 and 2.



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ComEd proposes to revise the Technical Specifications, for Byron and Braidwood to support steam generator replacement. ComEd will be replacing the original Westinghouse D4 steam generators (OSGs) at Byron and Braidwood with Babcock & Wilcox International (BWI) steam generators. Because of the design differences between the OSGs and the replacement steam generators, the analyses performed for use of the Interim Plugging Criteria (IPC), F* Alternate Repair Criteria, and sleeving methodologies is not applicable to the replacement steam generators. Therefore, this amendment request specifies which generator model the repair mechanism is applicable. Also, the amendment request will raise the Reactor Coolant System Dose Equivalent Iodine-131 level to the pre-IPC value.

This package consists of the following:

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| Attachment A | Description and Safety Analysis of Proposed Changes to Appendix A |
| Attachment B-1 | Marked Up Pages for Proposed Changes to Appendix A for Byron Station |
| Attachment B-2 | Marked Up Pages for Proposed Changes to Appendix A for Braidwood Station |
| Attachment C | Evaluation of No Significant Hazards Consideration for Proposed Changes to Appendix A |
| Attachment D | Environmental Assessment for Proposed Changes to Appendix A |

Please note, References 1 and 2 transmitted requests to amend the same Technical Specifications sections ("Steam Generators" and the "Reactor Coolant System") as discussed in the Attachments. ComEd will be contacting the Staff prior to the issuance of the enclosed amendment request to ensure the final Technical Specification pages are correct. Also, the enclosed package does not contain the Improved Technical Specification (ITS) Pages for these sections. The ITS pages will be submitted at a later date.

The proposed changes in this license amendment have been reviewed and approved by both On-Site and Off-Site review in accordance with ComEd procedures.

ComEd is notifying the State of Illinois of our application for this license amendment request by transmitting a copy of this letter and its attachment to the designated State Official.

The Byron Unit 1 Steam Generator Replacement Outage (SGRO) is scheduled during the eighth refuel outage (B1R08). The Braidwood Unit 1 SGRO is scheduled during the seventh refuel outage (A1R07). ComEd respectively requests the NRC Staff review and approve this license amendment request no later than November 3, 1997, to support the current outage schedule for the lead station, Byron Unit 1.

I affirm that this transmittal is true and correct to the best of my knowledge, information and belief.

Please address any comments or questions regarding this matter to Denise Saccomando, Senior PWR Licensing Administrator at (630) 663-7283.

Sincerely,

R. Blomgren for

John B. Hosmer
Engineering Vice President

Signed before me on this 28th day of February, 1997 by

Jacqueline T. Evans
Notary Public

OFFICIAL SEAL
JACQUELINE T EVANS
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES: 12/15/97

Attachments

cc: A. B. Beach, Regional Administrator - RIII
G. F. Dick, Jr., Byron/Braidwood Project Manager - NRR
S. D. Burgess, Senior Resident Inspector - Byron
C. J. Phillips, Senior Resident Inspector - Braidwood
Office of Nuclear Safety - IDNS

# ATTACHMENT A

**DESCRIPTION AND SAFETY ANALYSIS
FOR PROPOSED CHANGES
TO APPENDIX A, TECHNICAL SPECIFICATIONS,
OF FACILITY OPERATING LICENSES
NPF-37, NPF-66, NPF-72, NPF-77**

## A. DESCRIPTION OF PROPOSED CHANGE

Commonwealth Edison Company (ComEd) proposes to amend Technical Specifications (TS) 3/4.4.5, "Steam Generators," and TS 3/4.4.8, "Reactor Coolant System Specific Activity," for Byron Nuclear Power Station Units 1 and 2 (Byron) and Braidwood Nuclear Power Station Units 1 and 2 (Braidwood) to support steam generator (SG) replacement. ComEd will be replacing the original Unit 1 Westinghouse D4 steam generators (OSGs) at Byron and Braidwood with Babcock and Wilcox International (BWI) steam generators. The replacements are scheduled to occur at the end of Cycle 8 for Byron Unit 1 and the end of Cycle 7 for Braidwood Unit 1. The replacement steam generators (RSGs) will use a different tubing and support plate design than the OSGs. The design differences invalidate the analyses performed for use of Interim Plugging Criteria (IPC), F* Alternate Repair Criteria, and sleeving methodologies (Westinghouse laser welded or Combustion Engineering Tungsten Inert Gas (TIG) welded sleeves) for repair of Unit 1 tubes. Since there will be no accident leakage beyond the primary-to-secondary leakage limits assumed in the Main Steam Line Break (MSLB) analyses, the Reactor Coolant System (RCS) Dose Equivalent (DE) Iodine-131 (I-131) limit for Unit 1 will be restored consistent with the industry standard for plants without degraded SG tubes.

The changes proposed to TS 3/4.4.5 and the associated Bases will: 1) permit application of IPC to Unit 1 OSGs only through Cycle 8 for Byron and Cycle 6 for Braidwood, 2) permit application of F* to Westinghouse Model D4 SGs only, and 3) limit tube repair methods by sleeving to Westinghouse Model D4 and D5 steam generators. Note that on August 19, 1996, a request was submitted to extend application of IPC through Cycle 9 for Byron and through Cycle 7 for Braidwood. As a result of steam generator replacement, Byron would only require IPC through Cycle 8.

The changes proposed to TS 3/4.4.8 and the associated Bases will raise the Unit 1 RCS DE I-131 limit from 0.35 microCuries per gram (μCi/gm) to 1.0 μCi/gm after completion of Cycle 8 for Byron and Cycle 7 for Braidwood.

The proposed changes are described in detail in Section E of this Attachment. Affected current TS pages showing the proposed change are included in Attachments B-1 and B-2 for Byron and Braidwood, respectively. Affected Improved Technical Specifications (ITS) pages will be prepared and submitted at a later date showing the proposed changes for Byron and Braidwood.

## B. DESCRIPTION OF THE CURRENT REQUIREMENTS

TS 3/4.4.5, Steam Generators

The existing TS 4.4.5 provides the surveillance requirements fo. sample selection, inspection frequency, acceptance criteria, repair methods, and required reports to the NRC. Affected provisions of this TS are as follows:

- TS Surveillance Requirement (TSSR) 4.4.5.2.b.4 requires that for all indications left in service as a result of application of the tube support plate (TSP) voltage-based repair criteria, the indications shall be inspected by bobbin coil probe during all future refueling outages.

- TSSR 4.4.5.2.b.5 requires that tubes which remain in service due to the application of the F* criteria will be inspected, in the tubesheet region, during all future outages.

- TSSR 4.4.5.2.e requires a random sample inspection of at least 20% of the total number of laser welded and TIG welded sleeves installed for axial and circumferential indications at the end of each cycle. This paragraph also describes the required expansion program for sleeve inspection should an imperfection exceeding the repair limit be found.

- TSSR 4.4.5.3.a requires the first inservice inspection to be performed in a specified interval of time following initial criticality. Frequencies of subsequent inservice inspections are also addressed.

- TSSR 4.4.5.4.a.6) defines the plugging or repair limit for tubing as the imperfection depth equal to 40% of the nominal tube wall thickness. The plugging limit imperfection depth for laser welded sleeves is equal to 40% of the nominal sleeve wall thickness. The plugging limit imperfection depth for TIG welded sleeves is equal to 32% of the nominal sleeve wall thickness. For Unit 1, this definition does not apply to defects in the tubesheet that meet the criteria for an F* tube.

- TSSR 4.4.5.4.a.10) defines acceptable tube repairs using approved sleeving processes for Units 1 and 2.

- TSSR 4.4.5.4.a.12) defines the F* distance as the distance into the tubesheet from the secondary face of the tubesheet or the top of the last hardroll, whichever is further into the tubesheet, that has been determined to be 1.7 inches.

- TSSR 4.4.5.4.a.13) defines the F* Tube as a Unit 1 steam generator tube with degradation below the F* distance which has no indications of degradation (i.e., no indication of cracking) within the F* distance. Defects contained in an F* tube are not dependent on flaw geometry.

- TSSR 4.4.5.5.e identifies reporting requirements associated with the application of IPC and F* to Unit 1.

TS 3/4.4.8, Reactor Coolant System Specific Activity

The existing TS 3.4.8 requires that the specific activity of the RCS be limited to less than or equal to 0.35 μCi/gm DE I-131 for Unit 1 and 1.0 μCi/gm DE I-131 for Unit 2. When in Modes 1, 2, or 3 (greater than or equal to 500°F), action is required to place the unit in at least Hot Standby with $T_{avg}$ less than 500°F within 6 hours if the DE I-131 limit has been exceeded for more than 48 hours or if the limits of TS Figure 3.4-1 have been exceeded. When in Modes 1, 2, 3, 4, and 5, sampling and analysis in accordance with Table 4.4-4 is required when the DE I-131 limits are exceeded until the specific activity of the RCS is restored to within its limits.

## C. BASES FOR THE CURRENT REQUIREMENT

TS 3/4.4.5, Steam Generators

The surveillance requirements for inspection of the SG tubes ensure that the structural integrity of this portion of the RCS will be maintained. This integrity is necessary to ensure that the structural and leakage integrity of the steam generator (tubes) are acceptable for continued service and will withstand the peak pressures that could develop following a postulated MSLB accident.

Prior to development of the specialized plugging criteria (IPC and F*), plugging criteria based on tube integrity minimum wall thickness requirements were established for the steam generator as required by ASME Section XI, IWB-3521 and in accordance with the design requirements of ASME Section III. The specialized plugging criteria for SG tubes and sleeves is based on analysis performed to the requirements of Regulatory Guide (RG) 1.121 and Generic Letter 95-05. For tubes left in service due to application of IPC or F* Criteria, analyses have been performed to demonstrate acceptable probability of tube burst, and that MSLB leakage is within site specific limits.

The current repair method of sleeving permits a process to reestablish the primary to secondary pressure retaining boundary of a tube containing degradation exceeding the repair limit. Installation of the sleeve provides a leak tight boundary that spans the defective area and restores the structural integrity of the tubing to satisfy the RG 1.121 requirements.

The current program for inservice inspection of SG tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of SG tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of SG tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

Reporting criteria associated with application of IPC and F* ensure that the Staff is aware of any abnormalities detected during an inspection and provide notification as to the extent of application of the criteria.

TS 3/4.4.8, Reactor Coolant System Specific Activity

The provisions of TS 3.4.8 ensure that the resulting 2-hour doses at the Site Boundary will not exceed an appropriately small fraction of Title 10 Code of Federal Regulations Part 100 (10CFR100) dose guidelines. The current site allowable accident primary-to-secondary leak rates using the 0.35 μCi/gm dose equivalent from I-131, are 26.8 gpm for Braidwood Unit 1 and 36.5 gpm for Byron Unit 1. These leakage values include postulated accident leakage from the faulted SG and 0.1 gpm (150 gpd) primary-to-secondary leakage from each of the three unfaulted SGs, as allowed per TS 3.4.6.2c. The Unit 1 RCS DE I-131 limit was previously lowered from 1.0 μCi/gm to 0.35 μCi/gm as a compensatory measure in order to account for event-induced leakage resulting from the peak pressures that could develop following a postulated MSLB accident. This was adopted as a defense-in-depth measure when the 3-Volt IPC was implemented. On January 31, 1997, ComEd submitted a request to reduce the Byron Unit 1 DE I-131 limit to 0.2 μCi/gm to permit a cycle length of 540 days. The corresponding site allowable accident primary-to-secondary leak limit for Byron Unit 1 is 64.0 gpm.

The impact of event-induced SG tube leakage on Unit 2 has not been an issue since the Unit 2 Model D5 SGs satisfy applicable structural integrity requirements without adopting IPC or F*. Unit 2 employs the TS plugging limit criteria of 40% of the nominal tube wall thickness.

## D. NEED FOR REVISION OF THE REQUIREMENT

ComEd plans to replace the Unit 1 steam generators beginning in November 1997 for Byron Station and in the Fall of 1998 for Braidwood Station. For Unit 1, the current surveillance requirements for sample selection, inspection frequency, acceptance criteria, repair methods and required reports were specifically developed for the Westinghouse Model D4 SGs, in part to permit tubes to remain in service that were experiencing various tube degradation mechanisms. After replacement, the current IPC, F* Criteria, and sleeving repair methods will no longer be applicable, and the reduced Unit 1 RCS activity limit will no longer be necessary.

The RSGs have several design differences that improve the resistance to or eliminate the tube degradation mechanisms experienced by the OSGs. These models differ significantly in tube and TSP materials and design. The OSGs have 0.75" thick carbon steel TSPs with drilled hole tube supports. The RSGs have 3" deep lattice grid tube supports manufactured from stainless steel. The OSG tubes are mill annealed Inconel 600 which were hard rolled into the tubesheet during initial assembly. The OSG tubes were shot peened in the tubesheet area after initial operation for Byron and prior to initial operation for Braidwood. The U-bend area of rows 1 and 2 were thermally stress relieved at both stations after initial operation. The RSGs have thermally treated Alloy 690 tubes which are hydraulically expanded into the tube sheet during initial assembly. Alloy 690 is more resistant to stress corrosion cracking (SCC) than Alloy 600.

The Unit 2 Westinghouse Model D5 SGs have not experienced the same degradation mechanisms as the D4 SGs, primarily due to the different design features and fabrication practices that were used on the Model D5 SGs. The D5s have 1.125" stainless steel support plates with quatrefoil tube supports. The D5 tubes are made of thermally treated Inconel 600 and were hydraulically expanded into the tubesheet during initial assembly.

TS 3/4.4.5, Steam Generators

After replacement of the OSGs, at the end of Cycle 8 for Byron Unit 1 and Cycle 7 for Braidwood Unit 1, the IPC and F* Criteria requirements will no longer apply. The analyses performed to support application of the IPC and F* Criteria were specifically based on the Westinghouse Model D4 design steam generators. These analyses do not apply to the RSGs.

After Cycle 8 for Byron Unit 1 and Cycle 7 for Braidwood Unit 1, the currently approved Westinghouse laser welded and Combustion Engineering TIG welded sleeving processes will only be applicable to Unit 2. These sleeving processes were developed specifically for Westinghouse Model D4 and D5 steam generators.

TS 3/4.4.8, Reactor Coolant System Specific Activity

After the Model D4 SGs are replaced, the Unit 1 activity limit will be returned to the 1.0 μCi/gm value associated with undegraded SG tubes. The calculation of event-induced SG tube leakage limits is not required for undegraded SG installations which employ tube plugging criteria that preclude event-induced leakage. As such, lowering of the DE I-131 limit to permit calculation of higher event-induced SG tube leakage limits will be removed from the Unit 1 design basis.

## E. DESCRIPTION OF THE REVISED REQUIREMENT

ComEd proposes to revise current TS 3/4.4.5 and 3/4.4.8 as follows:

TSSR 4.4.5.2, Steam Generator Tube Sample Selection and Inspection

- TSSRs 4.4.5.2.b.4) and 4.4.5.2.b.5) are revised to be specific to Westinghouse Model D-4 steam generators and not Unit 1 specific.

- TSSR 4.4.5.2.e is revised to indicate that the inspection sampling program for sleeves is applicable to Westinghouse Model D4 and D5 steam generators only.

TSSR 4.4.5.3, Inspection Frequencies

- TSSR 4.4.5.3.a is revised to apply the same interval for inservice inspection to replacement steam generators as was done for the original steam generators.

TSSR 4.4.5.4, Acceptance Criteria

- TSSR 4.4.5.4.a.6) is revised for the plugging or repair limit definitions. Tubes will be required to be plugged or repaired for imperfections exceeding the plugging or repair limit of 40% of nominal tube wall thickness. This limit is applicable to both units and not designated unit or steam generator type specific. The criteria for plugging sleeved tubes is applicable to Westinghouse Model D4 or D5 steam generators only. Presently there are no approved sleeves for use on the RSGs. Also, reference to F* tubes is made specific to Westinghouse Model D4 steam generators.

- TSSR 4.4.5.4.a.10) is revised to indicate that tube repair by the currently approved laser welded and TIG welded sleeving methodologies are only applicable to Westinghouse Model D4 or D5 steam generators.

- TSSR 4.4.5.4.a.12) and TSSR 4.4.5.4.a.13) are revised to indicate that the definition of the F* distance and F* tube are only applicable to Westinghouse Model D4 SGs.

<u>TSSR 4.4.5.5, Reports</u>

- TSSR 4.4.5.5.e is revised to be applicable for Westinghouse Model D4 SGs.

<u>BASES TS 3/4.4.5, Steam Generators</u>

- Bases section 3/4.4.5 is revised to indicate that the acceptable plugging criteria for laser welded and TIG welded sleeved tubes are specific to Westinghouse Model D4 or D5 steam generators. It is also indicated that the F* criteria is specific to Westinghouse Model D4 steam generators.

<u>TS 3/4.4.8, RCS Specific Activity</u>

- Footnotes applicable to Limiting Condition for Operation 3.4.8a and the associated Action Statements are revised to indicate that the Unit 1 reactor coolant DE I-131 is limited to 0.35 μCi/gm through Cycle 8 for Byron and through Cycle 7 for Braidwood.

- Figure 3.4-1 is revised to indicate the Unit 1 (after Cycle 8 for Byron and after Cycle 7 for Braidwood) and Unit 2 reactor coolant DE I-131 specific activity limit versus percentage of rated thermal power for conditions where the DE I-131 >1.0 μCi/gm. Figure 3.4-2 is added to indicate the specific activity limits for Unit 1 (through Cycle 8 for Byron and through Cycle 7 for Braidwood) with DE I-131 >0.35 μCi/gm.

- Footnotes for Table 4.4-4 are revised to indicate that the Unit 1 reactor coolant DE I-131 is limited to 0.35 μCi/gm through Cycle 8 for Byron and through Cycle 7 for Braidwood.

<u>BASES TS 3/4.4.8</u>

- For Braidwood Station, the Bases are updated to clarify that the limiting accident for Unit 1 through Cycle 7 is the MSLB with event-induced leakage on the faulted SG and 150 gpd from each unfaulted SG. Byron Station submitted this clarification in a January 31, 1997 submittal that also reduced the Byron Unit 1 DE I-131 to 0.20 μCi/gm.

## F. BASES FOR THE REVISED REQUIREMENT

TS 3/4.4.5, Steam Generators

For Unit 1, after Cycle 8 for Byron and Cycle 7 for Braidwood, the IPC and F* requirements will no longer be applicable to the RSGs and the TS acceptance limits will solely be based on the through-wall criteria which requires that tubes be plugged with imperfections exceeding the plugging limit of 40% of the nominal tube wall thickness. The proposed program for periodic inservice inspection of the replacement steam generators monitors the integrity of the SG tubing to ensure that there is sufficient time to take proper and timely corrective action if any tube degradation is present. The proposed program is consistent with the Standard Technical Specifications.

The purpose of the TS repair limit in conjunction with surveillance and maintenance programs is to ensure that tubes accepted for continued service will retain adequate structural and leakage integrity during normal, transient, and postulated accident conditions, consistent with General Design Criteria (GDC) 14, 15, 30, 31, and 32 of Title 10 Code of Federal Regulations Part 50 (10 CFR 50), Appendix A.

The RSG design, surveillance, and maintenance meet the requirements of GDC 14. The RSG portions of the reactor coolant pressure boundary are designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, rapidly propagating failure or gross rupture. The RSG design meets ASME Section III requirements and complies with 10CFR50.55a.

The RSG design, surveillance, and maintenance meet the requirements of GDC 15. The RSG portions of the reactor coolant system are designed with sufficient margin to assure that ASME Section III stress limits are not exceeded during any condition of normal operation or anticipated operational occurrences. The RSG tubing has been structurally evaluated under the requirements of ASME Section III, 1986 Edition for Service Levels A, B, C, and D (normal, upset, emergency and faulted conditions, respectively).

The RSG design meets the requirements of GDC 30. RSG portions of the reactor coolant pressure boundary are designed, fabricated, erected, and tested to the highest practical quality standards by meeting the ASME Code and 10CFR50, Appendix B. Detection and identification of the location of RSG leakage is through existing plant instrumentation and procedures.

The RSG design meets the requirements of GDC 31. The RSG portions of the reactor coolant pressure boundary are designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design considers service temperatures and other operating, maintenance, testing and postulated accident conditions; uncertainties in determining material properties; affects of radiation on material properties; residual, steady state and transient stresses; and size of indications.

The RSG design, surveillance, and maintenance meet the requirements of GDC 32. RSG portions of the reactor coolant pressure boundary are designed to permit periodic inspection and testing of important areas and features to assess structural and leak-tight integrity. ASME Section XI, provides the depth of an allowable outside diameter (O.D.) flaw for tubes in service. The RSG has tubing fabricated from SB-163 material (Inconel Alloy 690) which is examined by eddy current methods to the requirements of ASME Section III, NB-2550. The tubing has a radius to thickness ($r/t$) ratio less than 8.70. In accordance with ASME Section XI, for tubing having an $r/t$ ratio of less than 8.70, the depth of an allowable O.D. flaw shall not exceed 40% of the nominal tube wall thickness.

Upon replacement of the steam generators, the current sleeving methodology will no longer be applicable to the RSGs. Making the current sleeving methodology specific to Westinghouse Model D4 and D5 steam generators and not unit specific addresses this issue. The change is also required because a sleeving methodology has not been analyzed and approved by the NRC for the Unit 1 RSGs.

ComEd will continue to apply the TS maximum primary-to-secondary leakage limit of 150 gpd (0.1 gpm) through any one SG at Byron and Braidwood to help preclude the potential for excessive leakage during all plant conditions. The EPRI recommended 150 gpd limit provides for leakage detection and plant shutdown in the event of an unexpected tube leak and precludes the potential for excessive leakage or tube burst in the event of a Main Steam Line Break or under Loss of Coolant Accident conditions.

TS 3/4.4.8, Reactor Coolant System Specific Activity

The provisions of TS 3.4.8 ensure that the resulting 2-hour doses at the Site Boundary will not exceed an appropriately small fraction of 10CFR100 dose guidelines. The Unit 1 RSG installation will employ tube plugging criteria which precludes event-induced leakage. As such, the compensatory Unit 1 RCS activity limit of 0.35 μCi/gm will be removed and the accident analysis basis activity limit of 1.0 μCi/gm will be reinstated.

The design basis doses calculated for postulated accidents involving degradation of SG tubes, such as Steam Generator Tube Rupture (SGTR) and MSLB accidents, as presented in UFSAR Chapter 15 accident analysis are not affected by this change.

## G. IMPACT OF THE PROPOSED CHANGE

With the implementation of this license amendment request, the Braidwood and Byron Unit 1 SGs will continue to retain adequate structural and leakage integrity consistent with GDC 14, 15, 30, 31, and 32 of Title 10, Code of Federal Regulations Part 50 (10CFR50) Appendix A.

Raising the Unit 1 RCS dose equivalent I-131 activity limit to 1.0 μCi/gm removes the compensatory limits associated with IPC as applied to the Model D4 SGs and reestablishes the activity limit consistent with the industry standard for plants without degraded SG tubes. The 1.0 μCi/gm limit ensures that the resulting 2-hour dose rates at the Byron and Braidwood site boundaries will not exceed an appropriately small fraction of 10CFR100 dose guideline values.

Currently, there is an outstanding application for amendment to the NRC submitted August 19, 1996, that is requesting renewal of the IPC limit for ODSCC indications at locked tube support plate intersections. The proposed change requests extension of the Unit 1 IPC limit through Cycle 9 for Byron and through Cycle 7 for Braidwood. In addition, on January 31, 1997, Byron Station submitted an application for amendment to reduce the RCS DE I-131 limit to 0.20 μCi/gm. Since neither of these requests has been approved, the markups provided in Attachments B-1 and B-2 are done on current TS pages which have not incorporated these requests.

## H. SCHEDULE REQUIREMENTS

The Byron Unit 1 Steam Generator Replacement Outage (SGRO) is scheduled during the eighth refuel outage (B1R08). The Braidwood Unit 1 SGRO is scheduled during the seventh refuel outage (A1R07). Approval of this change is requested by November 3, 1997 to support the current outage schedule for the lead station, which is Byron Unit 1.