Commonwealth Edison
1400 Opus Place
Downers Grove, Illinois 60515

June 3, 1994

Mr. William T. Russell, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Document Control Desk

Subject: Application for Amendment to Facility Operating Licenses:

Byron Station Units 1 and 2
(NPF-37/66; NRC Docket Nos. 50-454/455)

Braidwood Station Units 1 and 2
(NPF-72/77; NRC Docket Nos. 50-456/457)

"Steam Generator Tube Sleeving Followup Commitments"

References: See Attached

Dear Mr. Russell:

Reference 1 transmitted Commonwealth Edison Company's (CECo) request to amend Technical Specification 3/4.4.5, "Steam Generators" which permits the sleeving of the steam generators tubes. Subsequent to that amendment request, the Nuclear Regulatory Commission (NRC) transmitted Reference 2 (See Attachment 5) requesting CECo to commit to a near term license amendment to address conditions associated with the approval of the Westinghouse and Babcock and Wilcox Nuclear Technologies steam generator sleeves. Four conditions were specified in the letter:

1. Amend the Byron and Braidwood licenses to reflect a primary-to-secondary leakage rate limit of 150 gallons per day (gpd).
2. Amend the Byron and Braidwood licenses to reflect an inservice inspection of a minimum of 20 percent of a random sample of the sleeves for axial and circumferential indication at the end of cycle.
3. Add a condition to the Byron and Braidwood licenses to conduct additional corrosion testing to establish the design life for the kinetically or laser welded sleeved tubes in the presence of a crevice.
4. Perform post weld heat treatment at 1400^{0}F minimum soak temperature with a 5 minute minimum soak time on freespan kinetically or laser welded joints until additional supporting data becomes available.

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In Reference 3 (See Attachment 6), CECo committed to these conditions and stated that the proposed license amendments to incorporated conditions 1, 2, and 3 would be issued within 90 days from the issuance of the sleeving amendment. The approved amendment and Safety Evaluation Report (SER) was subsequently issued on March 4, 1994, and transmitted in Reference 4.

To ensure completion of these commitments, pursuant to 10 CFR 50.90, CECo proposes to amend Facility Operating Licenses NPF-37, NPF-66, NPF-72, and NPF-77 and to amend the associated Appendix A, Technical Specifications. The proposed amendment request revises Technical Specification 4.4.5.2 and 3.4.6.2, "Steam Generators", and the associated bases.

The amendment request is subdivided as follows:

Attachment 1: Description and Safety Analysis of Proposed Changes

Attachment 2: Proposed Revision to the Operating Licenses and Technical Specifications for Byron and Braidwood Stations

Attachment 3: Evaluation of Significant Hazards Considerations

Attachment 4: Environmental Assessment

The proposed changes have been reviewed and approved by the On-site and Off-site Review Committees in accordance with CECo procedures. CECo has reviewed this proposed amendment in accordance with 10 CFR 50.92(c) and has determined that no significant hazards consideration exists.

CECo is notifying the State of Illinois of our application for these amendments by transmitting a copy of this letter and the associated attachments to the designated State Official.

To the best of my knowledge and belief, the statements contained in this document are true and correct. In some respects these statements are not based on my personal knowledge, but on information furnished by other CECo employees, contractor employees, and/or consultants. Such information has been reviewed in accordance with company practice, and I believe it to be reliable.

Please address any comments or questions regarding this matter to this office.

Sincerely,

Denise Saccomando

Denise Saccomando
Nuclear Licensing Administrator

Attachments

cc: G. F. Dick, Byron Project Manager - NRR
R. R. Assa, Braidwood Project Manager - NRR
H. Peterson, SRI - Byron
S. G. Dupont, SRI - Braidwood
B. Clayton, Branch Chief - Region III
Office of Nuclear Facility Safety - IDNS

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OFFICIAL SEAL
MARYELLEN D LONG
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES:04/15/98

REFERENCES

1. J. A. Bauer letter to T. E. Murley dated August 13, 1993, transmitting Application for Amendment to Facility Operating Licenses for Byron and Braidwood Stations pertaining to Steam Generator Tube Sleeving Methodology

2. R. R. Assa letter to D. L. Farrar dated February 22, 1994, requesting that CECo commit to a near term license amendment to address conditions associated with the approval of Westinghouse and BWNT Steam Generator Sleeves

3. D. M. Saccomando letter to T. E. Murley dated February 24, 1994, committing to the conditions specified in Reference 2 above

4. R. R. Assa letter to D. L. Farrar dated March 4, 1994, transmitting Amendment 58 (Byron) and Amendment 46 (Braidwood) and associated SER approving the use of Westinghouse and BWNT sleeves

ATTACHMENT 1

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

Description of the Proposed Changes

Commonwealth Edison Company (CECo) proposes to amend Technical Specification (TS) 3.4.5 and 3.4.6.2, and facility operating licenses NPF-37, NPF-66, NPF-72 and NPF-77 to include the conditions contained in the Nuclear Regulatory Commission's (NRC) Safety Evaluation Report (SER) for TS Amendment 58 (Byron) and TS Amendment 46 (Braidwood), "Steam Generator Sleeving", issued March 4,1994, (TAC NOS. M87227, M87228, M87229, and M87230).

In a J. A. Bauer letter to T. E. Murley dated August 13, 1993, CECo submitted an amendment request for Byron and Braidwood Stations proposing to perform steam generator tube sleeving in accordance with the Westinghouse and B&W processes. The NRC Technical Staff found the proposed amendment request acceptable subject to certain conditions. These four conditions were discussed in a teleconference on February 18, 1994 and subsequently documented in a R. R. Assa letter to D. L. Farrar dated February 22, 1994 (Attachment 5). Byron and Braidwood found these four contingencies acceptable and formally committed to address these conditions as documented in a D. M. Saccomando letter to T. E. Murley dated February 24, 1994 (Attachment 6). Upon CECo's acceptance of the specified conditions, the SG sleeving processes addressed in the amendment were approved for Byron and Braidwood.

The conditions specified in R. R. Assa's letter dated February 22, 1994, are restated below:

1. Amend the license to reflect a primary-to-secondary leakage rate limit of 150 gpd.

2. Amend the license to reflect an inservice inspection of a minimum of 20% of a random sample of the sleeves for axial and circumferential indications at the end of each cycle. In the event that an imperfection of 40% or greater depth is detected, an additional 20% (minimum) of the unsampled sleeves should be inspected, and if an imperfection of 40% or greater depth is detected in the second sample, all remaining sleeves should be inspected. The inservice

inspection is required until the licensee demonstrates the corrosion resistance for laser-welded or kinetically welded joints in tubes that bound the material properties of the tubes installed in the Byron and Braidwood Steam Generator. If conformance with the requirements of the plant TS for tube structural integrity is not confirmed, the tubes containing the sleeves in question shall be removed from service.

3. Add a condition to the license to conduct additional corrosion testing to establish the design life for the kinetically or laser welded sleeved tubes in the presence of a crevice. The testing should determine the effects that material microstructure, chemistry, and joint crevices will have on primary water stress corrosion cracking (PWSCC) initiation and growth, bound the material condition that exist in Byron and Braidwood steam generators, and include the associated stress intensity values.

4. Perform post weld heat treatment at 1400 ^{0}F minimum soak temperature with a 5-minute minimum soak time on freespan kinetically or laser welded joints until additional supporting data becomes available.

Conditions 1 and 2 are being addressed by amending Technical Specifications 3.4.5, and 3.4.6.2. Condition 3 will be addressed by adding a license condition to Operating Licenses NPF-37, NPF-66, NPF-72, and NPF-77. Condition 4 will be addressed by making appropriate sleeving installation procedure changes to address the PWHT requirements and therefore is not addressed in this submittal.

The applicable Technical Specification and License pages, with the changes indicated, are included in Attachment 2 of this submittal.

1. Amend the License to Reflect a Primary-to-Secondary Leakage Rate Limit of 150 gpd

<u>Description and Bases of the Current Requirements</u>

For Byron Units 1 and 2, and Braidwood Unit 2, Technical Specification 3/4.4.6.2 c "Operational Leakage", limits the total reactor-to-secondary leakage to 1 gallon per minute (gpm) through all steam generators not isolated from the Reactor Coolant System (RCS) and 500 gallons per day (gpd) through any one steam generator (SG).

Please note that for Braidwood Station Unit 2, TS 3.4.6.2. c "Operational Leakage", limits the total reactor-to-secondary leakage to 1 gallon per minute through all steam generators not isolated from the Reactor Coolant System (RCS) and 500 gpd through any one steam generator. This specification was revised from 1 gallon per minute (gpm) and 500 gpd respectively by the Braidwood Unit 1 Interim Plugging Criteria (IPC) Amendment (Amendment 50) issued May 7, 1994. The proposed change to operational leakage, noted below, is not applicable to Braidwood Unit 2 since this license change has already been approved.

The primary-to-secondary leakage is maintained by limiting the leakage to 500 gpd through any one steam generator and a total of 1 gpm through all unisolated steam generators. This limit ensures that the dosage contribution from SG tube leakage will be limited to a small fraction of the 10 CFR 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line break or under loss of coolant accident (LOCA) conditions. Operating plants have shown that the 500 gpd leakage limit per steam generator can be readily detected by the steam generator blowdown radiation monitors.

Description and Bases of the Requested Revision

Specification 3.4.6.2, Reactor Coolant System Operational Leakage

NOTE: This proposed change is applicable to Byron Units 1 and 2, and Braidwood Unit 2.

The RCS leakage limit of 500 gpd for primary-to-secondary leakage through any one steam generator has been reduced to 150 gpd. This reduction requires the total leakage limit of 1 gpm through all steam generators not isolated from the RCS to be limited to 600 gpd.

Bases for 3/4.4.5, Steam Generators

References to the reactor-to-secondary leakage of "500 gpd" per steam generator, in the second paragraph of this bases section, are being changed to "150 gpd". In addition, the radiation monitors, specified in the second paragraph, that have proven effective in identifying SG tube leakage, have also been changed.

These changes read as follows:

> "The extent of cracking during plant operation will be limited by the limitation of steam generator tube leakage between the RCS and the Secondary Coolant System (reactor-to-secondary leakage = 150 gpd per steam generator)." (**NOTE:** This change has already been approved for Braidwood Unit 1 and therefore is not applicable to the Braidwood Unit 1 license).
>
> "Operating plants have demonstrated that reactor-to-secondary leakage of 150 gpd per steam generator can readily be detected by the radiation monitors of steam generator blowdown, main steam lines, or the steam jet air ejectors." (**NOTE:** The addition of the radiation monitors is also applicable to the Braidwood Unit 2 license).

Bases for 3/4.4.6.2, Operational Leakage

The third paragraph of this bases section discusses steam generator tube leakage. The total steam generator tube leakage limit of "1 gpm" for all steam generators is changed to "600 gpd". The "500 gpd" leakage limit per steam generator is changed to "150 gpd". The sentence that formerly read: "The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents", has been enhanced for clarification purposes.

Please note that in the Braidwood Unit 1 Interim Plugging Criteria Amendment (Amendment 50) issued May 7, 1994, the sentence addressing the accident analysis assumptions had been changed to read, "The 600 gpd limit is consistent with the assumptions used in the analysis of these accidents." This sentence will also be changed as noted below for consistency and clarification purposes.

The third paragraph of the basis will be changed to read as follows:

> "The total steam generator tube leakage limit of 600 gpd for all steam generators not isolated from the RCS ensures that the dosage contribution

from the tube leakage will be limited to a small fraction of the 10 CFR Part 100 dose guideline values in the event of either of steam generator tube rupture or steam line break. A 1 gpm total steam generator tube leakage limit for all steam generators and a 500 gpd limit per steam generator were the assumptions used in the analysis of these accidents in Chapter 15 of the UFSAR. The assumptions in Chapter 15 remain valid since the leakage limitations implemented for total and individual steam generator leakages are more conservative that those used for the analysis. The 150 gpd leakage limit per steam generator ensures that steam generator tube integrity shall be maintained in the event of a main steam line break or under LOCA conditions."

Historically, SG sleeve-to-tube joints were basically mechanical seals and were not considered leak-tight. The NRC has not required that sleeve repairs be leak-tight but only leak-limiting. The staff evaluated leakage based on plant-specific TS requirements for primary-to-secondary leakage limits under normal and accident conditions. CECo has analyzed the leakage effects of the HEJ, the HEJ with laser-welded joint, the freespan laser-welded joint and kinetically welded joint. The analyses show that even under extreme postulated conditions, the three joints will maintain satisfactory integrity against leakage. The continuous circumferential laser welded joints are inherently leak-tight.

Degraded tubes that were restored to operation as a result of sleeving are susceptible to additional degradation in the same SG environment. The sleeve is designed to extend past the welded joint and into the tubing. In the event that a sleeved tube fails near or at the weld, the sleeve extension will restrict tube movement and leakage. Leakage monitoring devices are intended to alert plant personnel to implement the appropriate procedures. However, based on experience with various causes of leakage through tubes including experience related to tubes repaired by sleeving, the staff has concluded that the current primary-to-secondary leakage limits in the Byron and Braidwood TS are not sufficient to detect early stages of sleeve degradation. To reach a satisfactory conclusion regarding the acceptability of the sleeving application, particularly in view of corrosion issues, the staff requested that the license be amended to reflect a primary-to-secondary leakage limit of 150 gallons per day per steam generator.

Impact of the Proposed Change

The original amendment requested tubesheet sleeves and tube support plate sleeves be approved as an alternate tube repair method for Byron and

Braidwood Units 1 and 2. The steam generator sleeves approved for installation use the Westinghouse process (laser welded joints) or the Babcock and Wilcox (B&W) process of kinetically welded joints. The sleeve configuration was designed and analyzed in accordance with the criteria of Regulatory Guide (RG) 1.121 and the design requirements of Section III of the American Society of Mechanical Engineers (ASME) Code. Fatigue and stress analyses of the sleeved tube assemblies for both processes produced acceptable results as documented in the Westinghouse and B&W topical reports submitted in the original sleeving package. Mechanical testing has shown that the structural strength of the sleeves under normal, faulted, and upset conditions is within acceptable limits. Leakage rate testing for the tube sleeves has demonstrated that primary-to-secondary leakage is not expected during all plant conditions. Any leakage through the sleeved region of the tube is fully bounded by the leak-before-break considerations and ultimately the existing steam generator tube rupture analysis included in the Byron and Braidwood Updated Final Safety Analysis Report (UFSAR).

Reduction of the leakage rate requirement from 500 to 150 gpd per SG will continue to ensure SG tube integrity is maintained in the event of a Main Steam Line Break (MSLB) or under LOCA conditions. The reduction to 150 gpd also limits the allowable primary-to-secondary leakage from 1 gpm to 600 gpd for all SGs not isolated from the RCS. This previous leakage limit, used in the UFSAR accident analysis, ensured the dosage contribution from tube leakage would be limited to a small fraction of the 10 CFR 100 dose guideline values in the event of either a steam generator tube rupture or MSLB. Reducing these limits does not result in a reduction of any safety margins.

2. Inservice Inspection of Sleeves

<u>Description and Bases of the Current Requirements</u>

The current surveillance requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator 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 steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

Description and Bases of the Requested Revision

Specification 4.4.5.2, Steam Generator Tube Sample Selection and Inspection

NOTE: This proposed change is applicable to Byron Units 1 and 2, and Braidwood Units 1 and 2.

Section of 4.4.5.2 d has been added to the specification and reads as follows:

"A random sample of at least 20 percent of the total number of sleeves shall be inspected for axial and circumferential indications at the end of each cycle. In the event that an imperfection of 40 percent or greater depth is detected, an additional 20 percent of the unsampled sleeves shall be inspected, and if an imperfection of 40 percent or greater depth is detected in the second sample, all remaining sleeves shall be inspected. These inservice inspections will include the entire sleeve and the tube at the heat treated area. The inservice inspection for the sleeves is required until the corrosion resistance for the laser welded or kinetically welded joints in tubes that bound the material parameters of the tubes installed in the steam generators has been demonstrated acceptable. If conformance with the acceptance criteria of section 4.4.5.4 for tube structural integrity is not confirmed, the tubes containing the sleeves in question shall be removed from service."

The surveillance requirements for inspection of the SG tubes ensure that the structural integrity of this portion of the RCS is maintained. The program for inservice inspection of SG tubes is essential in maintaining a surveillance on the condition 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 the steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken. The growth rate of previously identified tube degradation will be monitored and new imperfections identified, documented, and monitored for growth rates over the upcoming cycles.

Impact of the Proposed Change

The sleeve sample size has been increased from three (3) to twenty (20) percent during each scheduled inservice inspection. Increasing the random sample size of the sleeves to be inspected will increase the monitoring of those tubes for any further degradation while remaining inservice. If the sample identifies a sleeve with an imperfection of greater that 40 percent depth, an additional 20 percent of the sleeves shall be inspected. The sleeves that have identified imperfections of greater than 40 percent shall be evaluated and removed from service. The inservice inspections and additional corrosion testing for the sleeves and welded joints will continue until the corrosion resistance is demonstrated acceptable to the NRC. If conformance with the acceptance criteria of section 4.4.5.4 for tube structural integrity is not confirmed, the tubes containing the sleeves in question shall be removed from service.

3. License Condition to Address Corrosion Testing

Description and Bases of the Current Requirements

There are currently no license conditions to conduct additional corrosion testing for steam generator sleeves. Corrosion testing of the Westinghouse and B&W sleeves was however performed as described in the technical reports submitted in support of the original sleeving license amendment.

Description and Bases of the Requested Revision

Facility Operating License No. NPF-37 part 2.C (Byron Unit 1)

A license condition (item 16) will be added to the Unit 1 license to specify that a corrosion testing program, to establish sleeve design life and corrosion resistance to confirm tube structural integrity, will be performed for Byron. If the tube structural integrity is not confirmed or the corrosion testing program is not accepted by the NRC within the next two fuel cycles, the tubes containing the sleeves in question shall be removed from service.

The NRC staff and CECo concur that additional corrosion testing is needed to establish the design life for the sleeved tubes in the presence of the crevice created at the joint between the sleeve and the tube. The test data should determine the effects that material microstructure, chemistry, and joint crevices will have on primary water stress corrosion cracking initiation and growth. This testing program should be conducted for the sleeving processes installed for Byron Unit 1.

Facility Operating License No. NPF-66 part 2.C (Byron Unit 2)

A license condition (item 5) will be added to the Unit 2 license to specify that a corrosion testing program, to establish sleeve design life and corrosion resistance to confirm tube structural integrity, will be performed for Byron. If the tube structural integrity is not confirmed or the corrosion testing program is not accepted by the NRC within the next two fuel cycles, the tubes containing the sleeves in question shall be removed from service.

The NRC staff and CECo concur that additional corrosion testing is needed to establish the design life for the sleeved tubes in the presence of the crevice created at the joint between the sleeve and the tube. The test data should determine the effects that material microstructure, chemistry, and joint crevices will have on primary water stress corrosion cracking initiation and growth. This testing program should be conducted for the sleeving processes installed for Byron Unit 2.

Facility Operating License No. NPF-72 part 2.C (Braidwood Unit 1)

A license condition (item 6) will be added to the Unit 1 license to specify that a corrosion testing program, to establish sleeve design life and corrosion resistance to confirm tube structural integrity, will be performed for Braidwood. If the tube structural integrity is not confirmed or the corrosion testing program is not accepted by the NRC within the next two fuel cycles, the tubes containing the sleeves in question shall be removed from service.

The NRC staff and CECo concur that additional corrosion testing is needed to establish the design life for the sleeved tubes in the presence of the crevice created at the joint between the sleeve and the tube. The test data should determine the effects that material microstructure, chemistry, and joint crevices will have on primary water stress corrosion cracking initiation and growth. This testing program should be conducted for the sleeving processes installed for Braidwood Unit 1.

Facility Operating License No. NPF-77 part 2.C (Braidwood Unit 2)

A license condition (item 5) will be added to the Unit 2 license to specify that a corrosion testing program, to establish sleeve design life and corrosion resistance to confirm tube structural integrity, will be performed for Braidwood. If the tube structural integrity is not confirmed or the corrosion testing program is not accepted by the NRC within the next two fuel cycles, the tubes containing the sleeves in question shall be removed from service.

The NRC staff and CECo concur that additional corrosion testing is needed to establish the design life for the sleeved tubes in the presence of the crevice created at the joint between the sleeve and the tube. The test data should determine the effects that material microstructure, chemistry, and joint crevices will have on primary water stress corrosion cracking initiation and growth. This testing program should be conducted for the sleeving processes installed for Braidwood Unit 2.

Impact of the Proposed Change

The addition of a license condition to conduct additional corrosion testing to establish the design life for the kinetically or laser welded sleeved tubes will enhance the confidence that sleeves will maintain integrity for extended operations. The corrosion testing program will be conducted in a laboratory environment and will not in any way adversely affect plant operations or safety.

Schedule Requirements

This license amendment is being submitted as requested in the Safety Evaluation Report for Amendment 58 (Byron) and Amendment 46 (Braidwood) dated March 4, 1994. Byron and Braidwood Stations formally committed to address the conditions in the SER in a D. M Saccomando letter to T. E. Murley dated February 24, 1994.

Administrative controls are already in effect at Byron and Braidwood which address the proposed steam generator leak rate limits. Since no steam generator sleeves are currently installed at Byron or Braidwood, inservice inspection of the sleeves will not become an issue until completion of Byron Unit 1 Cycle 6 (spring 1996), assuming sleeves are installed in Byron Unit 1 during the fall 1994 outage. The corrosion testing program is currently under development. Based on this information, CECo respectfully requests that this amendment be approved prior to February, 1996.

ATTACHMENT 2

PROPOSED CHANGES TO FACILITY OPERATING LICENSES NPF-37, NPF-66, NPF-72 AND NPF-77

Revised Pages:	NPF-37 pg 5
	NPF-66 pg 2
	NPF-72 pg 3
	NPF-77 pg 3

PROPOSED CHANGES TO APPENDIX A TECHNICAL SPECIFICATIONS FOR FACILITY OPERATING LICENSES NPF-37, NPF-66, NPF-72 AND NPF-77

Revised Pages:		3/4 4-13*
		3/4 4-14
		3/4 4-21
	B	3/4 4-3
	B	3/4 4-4

Pages indicated with and asterisk(*) are not being revised and are only included for convenience.

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AFFECTED PAGES

(Continued)

(15) Use of advisors who were licensed only at the RO level will be evaluated on a case-by-case basis. Advisors shall be trained on plant procedures, technical specifications and plant systems, and shall be examined on these topics at a level sufficient to assure familiarity with the plant. For each shift, the remainder of the shift crew shall be trained as to the role of the advisors. These advisors shall be retained until the experience levels identified in the first sentence above have been achieved. The NRC shall be notified at least 30 days prior to the date that the licensee proposes to release the advisors from further service.

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D. The facility requires exemptions from certain requirements of Appendices A, E and J to 10 CFR Part 50. These include (a) an exemption from the requirements of Paragraph III.D.2(b)(ii) of Appendix J, the testing of containment air locks at times when containment integrity is not required (Section 6.2.6 of the SER), (b) an exemption from GDC-2 of Appendix A, the requirement that structures, systems and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes (Section 3.10 of SSER #5), (c) an exemption from GDC-13 and GDC-17 of Appendix A, the requirement that instrumentation be provided to monitor variables and systems over their anticipated ranges, and the requirement that provisions be included to minimize the probability of losing electric power (Section 9.5.4.1 of SSER #5), (d) an exemption from GDC-19 of Appendix A, the requirement that the control room have adequate radiation protection to permit access and occupancy under accident conditions (Section 6.5.1 of SSER #6), and (e) an exemption from the requirement of Section IV.F of Appendix E that a full participation emergency planning exercise be conducted within one year before issuance of the first operating license for full power and prior to operation above 5% of rated power (Section 13.3 of SSER #6). These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, these exemptions are hereby granted pursuant to 10 CFR 50.12. With the granting of these exemptions the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. The licensee shall fully implement and maintain in effect all provisions of the Commission approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10CFR 50.90 and 10CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Byron Nuclear Power Station Security Plan," with revisions submitted through January 14, 1988; "Byron Nuclear Power Station Security Personnel Training and Qualification Plan," with revisions submitted through September 26, 1986; and "Byron Nuclear Power Station Safeguards Contingency Plan," with revisions submitted through July 30, 1985.

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(page 5 of NPF-37)

(16) Steam Generator Sleeving Corrosion Testing

The licensee shall conduct additional corrosion testing to establish the design life for the kinetically or laser welded sleeved tubes in the presence of a crevice. The corrosion testing shall demonstrate the corrosion resistance for the kinetically or laser welded joints in tubes that bound the material parameters in the steam generators. The corrosion testing results shall be reviewed and accepted by the Nuclear Regulatory Commission prior to the Beginning-of-Cycle 9. If conformance with the requirements of the plant Technical Specifications for tube structural integrity is not confirmed, the tubes containing the sleeves in question shall be removed from service.

(5) CECo, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of 3411 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A (NUREG-1113), as revised through **Amendment No. 61** and revised by Attachment 2 to NPF-60, and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-37, dated February 14, 1985, are hereby incorporated into this license. Attachment 2 contains a revision to Appendix A which is hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan. (Revised **04-18-94: Am. 61**)

(3) Initial Test Program

Any changes to the Initial Startup Test Program described in Chapter 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(4) Regulatory Guide 1.97, Revision 2 Compliance

The licensee shall submit by March 1, 1987, a preliminary report describing how the requirements of Regulatory Guide 1.97, Revision 2 have been or will be met. The licensee shall submit by September 1, 1987, the final report and a schedule for implementation (assuming the NRC approves the DCRDR by March 1, 1987).

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(page 2 of NPF-66)

(5) Steam Generator Sleeving Corrosion Testing

The licensee shall conduct additional corrosion testing to establish the design life for the kinetically or laser welded sleeved tubes in the presence of a crevice. The corrosion testing shall demonstrate the corrosion resistance for the kinetically or laser welded joints in tubes that bound the material parameters in the steam generators. The corrosion testing results shall be reviewed and accepted by the Nuclear Regulatory Commission prior to the Beginning-of-Cycle 8. If conformance with the requirements of the plant Technical Specifications for tube structural integrity is not confirmed, the tubes containing the sleeves in question shall be removed from service.