LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable steam generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube* Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. When applying the expectations of 4.4.5.2.a through 4.4.5.2.c, previous defects or imperfections in the area repaired by the sleeve are not considered an area requiring reinspection. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

*When referring to a steam generator tube, the sleeve shall be considered a part of the tube if the tube has been repaired per Specification 4.4.5.4.a.10.

1) All tubes that previously had detectable tube wall penetrations greater than 20% that have not been plugged or sleeved in the affected area, and all tubes that previously had detectable sleeve wall penetrations that have not been plugged,

2) Tubes in those areas where experience has indicated potential problems,

3) At least 3% of the total number of sleeved tubes in all four steam generators or all of the sleeved tubes in the generator chosen for the inspection program, whichever is less. These inspections will include both the tube and the sleeve, and

4) A tube inspection (pursuant to Specification 4.4.5.4a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

2) The inspections include those portions of the tubes where imperfections were previously found.

INSERT A

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes or sleeves must exhibit significant (greater than 10% of wall thickness) further wall penetrations to be included in the above percentage calculations.

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d. 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 Specification 4.4.5.4 for tube structural integrity is not confirmed, the tubes containing the sleeves in question shall be removed from service.

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

600 gallons per day

c. ~~1 gpm~~ total reactor-to-secondary leakage through all steam generators not isolated from the Reactor Coolant System and 150 ~~500~~ gallons per day through any one steam generator,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. 40 gpm CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig, and

f. 1 gpm leakage at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.*

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, reduce the leakage rate to within limits within 4 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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*Test pressures less than 2235 psig but greater than 350 psig are allowed. Observed leakage shall be adjusted for the actual test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.

3/4.4.5 STEAM GENERATORS

The 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.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = ~~500~~ gallons per day per steam generator). 150 Cracks having a reactor-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. Operating plants have demonstrated that reactor-to-secondary leakage of ~~500~~ gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. 150 Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged or repaired by sleeving. The technical bases for sleeving are described in Westinghouse report WCAP-13698 Rev. 1 and Babcock & Wilcox Topical ~~Report BAW-2045PA Rev.~~ 1.

, mainsteam lines, or the steam jet air ejectors.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging or sleeving will be required for all tubes with imperfections exceeding the plugging or repair limit of 40% of the tube nominal wall thickness. If a sleeved tube is found to contain a through wall penetration in the sleeve of equal to or greater than 40% of the nominal wall thickness, the tube must be plugged. The 40% plugging limit for the sleeve is derived from Reg. Guide 1.121 analysis and utilizes a 20% allowance for eddy current uncertainty and additional degradation growth. Inservice inspection of sleeves is required to ensure RCS integrity. Sleeve inspection techniques are described in Westinghouse Report WCAP-13698 Rev. 1 and Babcock & Wilcox Topical Report BAW-2045PA Rev. 1. Steam Generator tube and sleeve inspections have demonstrated the capability to reliably detect degradation that has penetrated 20% of the pressure retaining portions of the tube or sleeve wall thickness. Commonwealth Edison will validate the adequacy of any system that is used for periodic inservice inspection of the sleeves and, as deemed appropriate, will upgrade testing methods as better methods are developed and validated for commercial use.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission pursuant to Specification 6.9.2 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These Detection Systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The total steam generator tube leakage limit of ~~1 gpm~~ 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 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. INSERT B ~~The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents.~~ The ~~500~~ 150 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the Leakage Detection Systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 40 gpm with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the Safety Injection flow will not be less than assumed in the safety analyses.

The 1 gpm leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series check valve failure. It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA that bypasses containment, those valves should be tested periodically to ensure low-probability of gross failure.

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INSERT B

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.

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AFFECTED PAGES

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 is not in excess of 3411 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein and other items identified in Attachment 1 to this license. The items in Attachment 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, as revised through **Amendment 48, dated March 11, 1994**, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3) Emergency Planning

In the event that the NRC finds that the lack of progress of completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of emergency preparedness, the provisions of 10 CFR Section 50.54(s)(2) will apply.

4) Initial Startup Test Program

Any changes to the Initial Test Program described in Section 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.

5) Regulatory Guide 1.97, Revision 2 Compliance

The licensee shall submit the final report and a schedule for implementation within six months of NRC approval of the DCRDR.

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(page 3 of NPF-72)

6) 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 7. 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.

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 is not in excess of 3411 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein in other items identified in Attachment 1 to this license. 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, as revised through **Amendment 48, dated March 11, 1994**, and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-72, dated July 2, 1987, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3) Emergency Planning

In the event that the NRC finds that the lack of progress of completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of emergency preparedness, the provisions of 10 CFR Section 50.54(s)(2) will apply.

4) Initial Startup Test Program

Any changes to the Initial Test Program described in Section 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.

INSERT D →

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(page 3 of NPF-77)

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 7. 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.

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable steam generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube* Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. When applying the expectations of 4.4.5.2.a through 4.4.5.2.c, previous defects or imperfections in the area repaired by the sleeve are not considered an area requiring reinspection. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

*When referring to a steam generator tube, the sleeve shall be considered a part of the tube if the tube has been repaired per Specification 4.4.5.4.a.10.

SURVEILLANCE REQUIREMENTS (Continued)

1) All tubes that previously had detectable tube wall penetrations greater than 20% that have not been plugged or sleeved in the affected area, and all tubes that previously had detectable sleeve wall penetrations that have not been plugged,

2) Tubes in those areas where experience has indicated potential problems,

3) At least 3% of the total number of sleeved tubes in all four steam generators or all of the sleeved tubes in the generator chosen for the inspection program, whichever is less. These inspections will include both the tube and the sleeve, and

4) A tube inspection (pursuant to Specification 4.4.5.4a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

2) The inspections include those portions of the tubes where imperfections were previously found.

d. For Unit 1 Cycle 5, implementation of the tube support plate interim plugging criteria limit requires a 100% bobbin coil probe inspection for all hot leg tube support plate intersections and all cold leg intersections down to the lowest cold leg tube support plate with outer diameter stress corrosion cracking (ODSCC) indications. An inspection using a rotating pancake coil (RPC) probe is required in order to show OPERABILITY of tubes with flaw-like bobbin coil signal amplitudes greater than 1.0 volt but less than or equal to 2.7 volts. For tubes that will be administratively plugged or repaired, no RPC inspection is required. The RPC results are to be evaluated to establish that the principal indications can be characterized as ODSCC.

Insert A

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.

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e. 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 Specification 4.4.5.4 for tube structural integrity is not confirmed, the tubes containing the sleeves in question shall be removed from service.

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 600 gallons per day total reactor-to-secondary leakage through all steam generators not isolated from the Reactor Coolant System and 150 gallons per day through any one steam generator,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. 40 gpm CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig, and

f. 1 gpm leakage at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.*

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, reduce the leakage rate to within limits within 4 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*Test pressures less than 2235 psig but greater than 350 psig are allowed. Observed leakage shall be adjusted for the actual test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.

3/4.4.5 STEAM GENERATORS

The 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.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = 150 gallons per day per steam generator). Cracks having a reactor-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. Operating plants have demonstrated that reactor-to-secondary leakage of 150 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged or repaired by sleeving. The technical bases for sleeving are described in Westinghouse report WCAP-13698 Rev. 1 and Babcock & Wilcox Topical Report BAW-2045PA Rev. 1.

, Mainsteam lines, or the steam jet Air ejectors.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging or sleeving will be required for all tubes with imperfections exceeding the plugging or repair limit of 40% of the tube nominal wall thickness. If a sleeved tube is found to contain a through wall penetration in the sleeve of equal to or greater than 40% of the nominal wall thickness, the tube must be plugged. The 40% plugging limit for the sleeve is derived from Reg. Guide 1.121 analysis and utilizes a 20% allowance for eddy current uncertainty and additional degradation growth. Inservice inspection of sleeves is required to ensure RCS integrity. Sleeve inspection techniques are described in Westinghouse Report WCAP-13698 Rev. 1 and Babcock & Wilcox Topical Report BAW-2045PA Rev. 1. Steam Generator tube and sleeve inspections have demonstrated the capability to reliably detect degradation that has penetrated 20% of the pressure retaining portions of the tube or sleeve wall thickness. Commonwealth Edison will validate the adequacy of any system that is used for periodic inservice inspection of the sleeves and, as deemed appropriate, will upgrade testing methods as better methods are developed and validated for commercial use.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These Detection Systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

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 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. INSERT B ~~The 600 gpd limit is consistent with the assumptions used in the analysis of these accidents.~~ The 150 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the Leakage Detection Systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 40 gpm with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the Safety Injection flow will not be less than assumed in the safety analyses.

The 1 gpm leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series check valve failure. It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA that bypasses containment, those valves should be tested periodically to ensure low-probability of gross failure.

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INSERT B

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.

ATTACHMENT 3

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS FOR PROPOSED CHANGES TO NPF-37, NPF-66, NPF-72, AND NPF-77 AND APPENDIX A TECHNICAL SPECIFICATIONS OF FACILITY OPERATING LICENSE NPF-37, NPF-66, NPF-72, AND NPF-77

Commonwealth Edison (CECo) has evaluated this proposed amendment and determined that it involves no significant hazards considerations. According to Title 10 Code of Federal Regulations Part 50 Section 92 Sub-Section c (10 CFR 50.92 (c)), a proposed amendment to an operating license involves no significant hazards considerations if operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3. Involve a significant reduction in a margin of safety.

A. INTRODUCTION

At Byron and Braidwood, two Steam Generator (SG) sleeving processes were requested in a previous amendment submitted to the Nuclear Regulatory Commission (NRC) in August of 1993. The Commission issued a Safety Evaluation Report authorizing the use of both sleeving processes, subject to the conditions in Amendment Number 58 for Facility Operating License NPF-37 and NPF-66 (Byron Station) and Amendment Number 46 for Facility Operating License NPF-72 and NPF-77 (Braidwood Station). The NRC requested that four additional conditions be addressed for approval of the two sleeving methodologies. Those four conditions are:

1. Amend the license to reflect a primary-to-secondary leakage rate limit of 150 gallons per day (gpd).

2. Amend the license to reflect an inservice inspection of a minimum of 20 percent 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 percent or greater depth is detected, an additional 20 percent (minimum) of the unsampled sleeves should be inspected, and if an imperfection of 40 percent 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 the laser-welded or kinetically welded joints in tubes that bound the material parameters of the tubes installed in the Byron and Braidwood Steam Generators. 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

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 growt' ·ound the material conditions that exist in Byron and Braidwood steam generators, and include the associated stress intensity values.

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

The amendment request addresses conditions 1, 2, and 3. Condition 4 will be implemented with appropriate sleeving installation procedures changes addressing the PWHT requirements.

B. 10 CFR 50.92 ANALYSIS

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The original amendment requested approved of tubesheet sleeves and tube support plate sleeves 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) and 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 the 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).

The reduction in TS leakage rate requirements from 500 gpd allowable per SG to 150 gpd further ensures that SG tube integrity is maintained in the event of a main steam line break (MSLB) or under Loss Of Coolant Accident (LOCA) conditions. The RG 1.121 criteria for establishing operational leakage rate limits require a plant shutdown based upon a leak-before-break consideration to detect a free span crack before a potential tube rupture. The 150 gpd limit will continue to allow for early leakage detection and require a plant shutdown in the event of the occurrence of an unexpected crack resulting in leakage that exceeds the revised Technical Specification limit.

The sleeve sample size has been increased to a minimum of twenty (20) percent of the inservice sleeves. Increasing the sample size of the sleeves to be inspected will increase the monitoring of tubes using sleeves for any further degradation while they remain 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. Increasing the monitoring of the sleeved tubes will decrease the probability of occurrence of an accident previously evaluated in the UFSAR.

Implementation of a corrosion testing program should determine the effects that material microstructure, chemistry, and joint crevices will have on primary water stress corrosion cracking initiation and growth. This program will not cause an increase in the probability or consequence of an accident previously evaluated because the testing program is conducted in laboratory conditions. If the results of the testing program do not confirm the structural integrity of the tubes, the tubes containing the sleeves in question shall be removed from service. These changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.**

The implementation of the proposed amendment will not introduce significant or adverse changes to the plant design basis. The proposed changes do not involve plant modification or changes to equipment, and consist of. reduction in allowable steam generator leakage limits, increase in the sample size of the steam generator tube sleeved and the addition of a commitment to perform a corrosion testing program on the sleeved tubes.

The reduction in TS leakage rate requirements from 500 gpd allowable per SG to 150 gpd further ensures that SG tube integrity is maintained in the event of a MSLB or under LOCA conditions. The 150 gpd limit is designed to provide for leakage detection and a plant shutdown in the event of the occurrence of an unexpected single crack resulting in excessive tube leakage. The limit provides for early detection and a plant shutdown prior to a postulated crack reaching critical crack lengths for Main Steam Line Break conditions.

Increasing the sample size of tubes sleeved during each scheduled inservice inspection will increase the monitoring of these tubes for any further degradation. The improved monitoring and evaluation of the tube and the sleeves assures tube structural integrity is maintained or the tube is removed for service.

Additionally, corrosion testing to establish sleeve design life and corrosion resistance to confirm tube structural integrity will be performed. If the tube structural integrity is not confirmed, the tubes containing the sleeves in question shall be removed from service.

With these actions the possibility of a new or different type of accident from any accident previously evaluated is not created.

3. The proposed change does not involve a significant reduction in a margin of safety.

Implementation of the proposed changes will not reduce the margin of safety. This amendment involves the reduction of steam generator leakage limit, and increase in the amount of sleeved tubes inspected and the incorporation of a corrosion testing program for sleeved tubes. All of these actions will help ensure steam generator tube integrity.

Reduction of the leakage rate requirement from 500 to 150 gallons per day (gpd) per steam generator will continue to ensure steam generator tube integrity is maintained in the event of main steam line break or under LOCA conditions. The reduction to 150 gpd also limits the allowable primary-to-secondary leakage from1 gallon per minute to 600 gpd for all steam generators not isolated from the Reactor Coolant System (RCS). This previous leakage limit, used in UFSAR accident analysis, ensured the dosage contribution from tube leakage would be limited to a small fraction of the 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. Reducing these limits will not result a reduction in the margin of safety.

The portions of the installed sleeve assembly which represent the reactor coolant pressure boundary can be monitored for the initiation and progression of sleeve/tube wall degradation, thus satisfying the requirement of Regulatory Guide 1.83. The portion of the tube bridged by the sleeve joints is effectively removed from the pressure boundary, and the sleeve then forms the new pressure boundary. The sleeve enhances the safety of the plant by increasing the protective boundaries of the steam generator. Keeping the tube in service with the use of a sleeve instead of plugging the tube and removing it from service increases the heat transfer efficiency of the steam generator. Monitoring for any increased degradation of a repaired steam generator tube shall be implemented at Byron and Braidwood by increasing the sampling size of inservice sleeves to include an additional twenty (20) percent of the sleeves inservice. During each scheduled in service inspection, each sampled sleeve evaluated and found to have unacceptable degradation shall be removed from service.

Implementation of a corrosion testing program should determine the effects that material microstructure, chemistry, and joint crevices will have on primary water stress corrosion cracking initiation and growth. This program is conducted in laboratory setting; therefore, will not involve a significant reduction in a margin of safety. In addition, the corrosion testing program will be performed to establish sleeve design life and corrosion resistance to confirm tube structural integrity. If the tube structural integrity is not confirmed, the tubes containing the sleeves in question shall be removed from service. These actions does not involve a significant reduction in a margin of safety.

Based on the preceding analysis it is concluded that operation of Byron and Braidwood Units 1 and 2, in accordance with the proposed amendment does not increase the probability of an accident previously evaluated, does not create the possibility of a new or different kind of accident from any accident previously evaluated, nor reduce any margins to plant safety. Therefore, the license amendment does not involve a Significant Hazards Consideration as defined in 10 CFR 50.92.

ATTACHMENT 4

ENVIRONMENTAL ASSESSMENT

Commonwealth Edison has evaluated the proposed amendment against the criteria for and identification of licensing and regulatory actions requiring environmental assessment in accordance with 10CFR51.21. It has been determined that the proposed change meets the criteria for a categorical exclusion as provided for under 10CFR51.22(c)(9).

The proposed change does not involve a significant hazards consideration as discussed in Attachment 3 to this letter. Also, this proposed amendment will not involve significant changes in the types or amounts of any radioactive effluents nor does it affect any of the permitted release paths. The amendment will limit the allowable leakage through all steam generator not isolated from the RCS or through any one steam generator reducing the amount of primary-to-secondary leakage permitted during normal plant operations. In addition, this change does not involve a significant increase in individual or cumulative occupational exposure. Therefore, this change meets the categorical exclusion permitted by 10CFR51.22(c)(9).

ATTACHMENT 5

NRC's STEAM GENERATOR SLEEVING REVIEW LETTER

dated

FEBRUARY 22, 1994

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 22, 1994

Docket Nos. STN 50-454, STN 50-455,
and STN 50-456, STN 50-457

RECEIVED

Byron Central File

Mr. D. L. Farrar, Manager
Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III, Suite 500
1400 OPUS Place
Downers Grove, Illinois 60515

Dear Mr. Farrar:

SUBJECT: STEAM GENERATOR SLEEVING REVIEW (TAC NOS. M87227, M87228, M87229, AND M87230)

By letter dated August 13, 1993, and additional supplements Commonwealth Edison Company (CECo or the licensee), submitted an amendment request to the licenses of Braidwood Station (Units 1 and 2) and Byron Station (Units 1 and 2) proposing steam generator (SG) tube sleeving in accordance with the (1) Westinghouse and (2) B&W processes. The NRC technical staff has completed its review of your proposed amendment request and finds it acceptable subject to certain conditions. These conditions, which were discussed with your staff during a telephone call on February 18, 1994, are as follows:

1. Amend the license to reflect a primary-to-secondary leakage limit of 150 gallons per day.

2. Amend the license to reflect an inservice inspection of a minimum of 20 percent 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 percent or greater depth is detected, an additional 20 percent (minimum) of the unsampled sleeves should be inspected, and if an imperfection of 40 percent 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 parameters of the tubes installed at Byron and Braidwood SG. If conformance with the requirements of the plant Technical Specifications (TS) for tube structural integrity is not confirmed, the tubes containing the sleeves in question should 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 presences 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.

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

As discussed with your staff, we are expecting a letter stating CECo's intent to incorporate these four conditions. We request that your letter be immediately forthcoming so that we may expeditiously continue to process the license amendment regarding approval of the steam generator tube sleeving.

Sincerely,

Ramin R. Assa, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

cc: See next page

ATTACHMENT 6

COMMONWEALTH EDISON'S LETTER CONFIRMING

ACCEPTANCE OF CONDITIONS IDENTIFIED IN

STEAM GENERATOR SLEEVING REVIEW

LETTER FROM THE NRC

dated

FEBRUARY 24, 1994

Commonwealth Edison
1400 Opus Place
Downers Grove, Illinois 60515

February 24, 1994

Dr. Thomas E. Murley, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Document Control Desk

Subject: Additional Information Pertaining to the Byron/Braidwood Proposed License Amendment Regarding Steam Generator Tube Sleeving

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

References: (a) J. Bauer letter to T. 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

(b) Teleconference on February 18, 1994, between Commonwealth Edison Company and the Nuclear Regulatory Commission concerning the Proposed Sleeving Amendment

(c) R. Assa letter to D. Farrar dated February 22, 1994, regarding the NRC's Steam Generator Sleeving Review

Dear Dr. Murley:

Commonwealth Edison Company (CECo) and the Nuclear Regulatory Commission (NRC) participated in a teleconference on February 18, 1994 to further discuss issues concerning the Byron and Braidwood proposed license amendments transmitted by Reference (a). Following this discussion, the NRC transmitted a letter (Reference (c)) stating that the NRC Technical Staff found the subject license amendments to be acceptable provided four contingency actions, which will be stated in the forthcoming Safety Evaluation Report, are met. CECo was requested to make a formal commitment to address these four contingencies.

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CECo agrees to the contingencies as stated in Reference (c). These contingencies are summarized below.

1. Amend the Byron and Braidwood licenses to reflect a primary-to-secondary leakage limit of 150 gallons per day through any one steam generator.

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 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 (minimum) of the unsampled sleeves should be inspected, and if an imperfection of 40 percent or greater depth is detected in the second sample, all remaining sleeves should be inspected.

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°F minimum soak temperature with a 5-minute minimum soak time on freespan kinetically or laser welded joints until additional supporting data becomes available.

CECo will submit a proposed license amendment for Byron and Braidwood within 90 days from the issuance date of the pending sleeving amendment. This amendment will address items 1, 2 and 3 above. CECo does not intend to ask for an extension of this 90 day period. Additionally, CECo will ensure that the sleeve post weld heat treatment is performed as specified in item 4.

CECo appreciates the Staff's efforts in expediting the review and issuance of the Byron and Braidwood steam generator sleeving license amendment. If you have any questions or comments concerning this matter, please contact this office.

Sincerely,

Joseph A Bauer for

Denise M. Saccomando
Nuclear Licensing Administrator

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

In each of these base and sensitivity cases, a significant fraction of core material was retained in the vessel. As a result, Case 11a was run with FMAXCP = 0.6 to ensure 100% of the core was forced out of the vessel at vessel failure. With all of the corium on the floor of the CRD room and the coolability of the debris limited to 100 kw/m^2, Case 11a gives an upper limit to the amount of core concrete attack and hydrogen generation expected in the presence of an overlying pool of water.

Table 13.7-7 summarizes the results of the debris coolability sensitivity runs compared with the two base cases.

When the debris to water heat transfer coefficient is set to the MAAP-BRP default value of 900 ks/m^2 (FCHF = 0.09), the debris in the CRD room valve pit sump is sufficiently cooled by the overlying water pool to prevent core concrete interaction. In the sensitivity Cases 11 and 12, where the coolability parameter is reduced to 100 kw/m^2, CCI erodes 1.5 feet of the concrete basemat, producing 63 pounds of hydrogen. In each of these four cases, approximately half of the core material remains in-vessel long term. Case 11a forced this residual debris out of the vessel and into the CRD room sump. It was found that some of the additional debris flowed out of the CRD sump and into the enclosure room through junction 9, which had failed earlier on debris contact. This allowed approximately 1.5 feet of concrete attack in both the CRD and enclosure sumps, thereby doubling the amount of hydrogen produced to 124 pounds.

In all the cases considered, the CsI release fraction was remarkably stable, varying between 6.32E-4 and 7.98E-4. It is therefore judged that the BRP containment is insensitive to the choice of FCHF.

These sensitivity studies show that the magnitude of the source term is independent of whether or not the containment sprays switch to recirculation mode. In cases where sprays switch to recirculation mode, containment pressures increase to approximately 40 psia at 36 hours after vessel failure. This value poses little threat to containment integrity and therefore the BRP containment can be judged insensitive to the use of sprays in recirculation mode. In cases where recirculation mode is not implemented, containment conditions are more benign with lower temperatures and pressures prevailing 36 hours after vessel failure. These scenarios will be examined further in Section 13.7.3.6 (Containment Flooding Sensitivities).

<u>Debris Cooling in the Sumps</u>

When vessel failure occurs, molten debris drops onto the CRD room floor and flows into the 2.25 foot deep CRD room valve pit. The sump is large enough to hold the entire core to a depth of 1.6 feet (Section 12.4.2.2). The MAAP-BRP model connects the CRD room to the enclosure room through several junctions (one of which is junction 9) originally representing a 2 inch line that runs horizontally between the nodes and lies 1.25 feet above the bottom of the CRD room sump. This line emerges 2 feet above the floor in the enclosure room, providing a convenient path for melted corium to flow between the two sumps. When implementing the MAAP-BRP code change to represent this junction, it was decided that the probability of corium freezing and plugging the passageway was sufficiently high enough for the junction to be considered a non-mechanistic pathway that would allow transport of debris between sumps in the event of a structural failure. Consequently, this line is assumed closed at the start of the transient, but will fail either on pressure differential at vessel failure or on temperature conditions after reaching the melting point of steel. Once opened, it can allow excess water in the enclosure room to drain into the CRD room, cooling the corium, or it may allow excess corium to transport to the adjacent node when the corium reaches its melting point and is at a level above the junction elevation.

Table 13.7-8 compares three low pressure cases that fail the vessel without the benefit of containment sprays. Cases 1 and 4 are typical of most MAAP-BRP cases in that approximately 50% of the core material remains in the vessel long term. In these cases, debris fills the CRD valve pit sump to a depth of 0.8 feet, somewhat below the opening to junction 9. Consequently, there can be no corium flow into the enclosure room unless more core material is forced from the vessel.

In Case 1, the low pressure failure base case, water remains on the floor of the CRD room for 8 hours after vessel failure. The containment pressurizes to 39 psia with the debris temperature peaking at 1600°F. This is below the melting point of steel and the initiation point for CCI, so junction 9 remains intact, precluding water flow into the CRD room from the enclosure.

In Case 4, the forced early containment failure reduces containment pressure to atmospheric levels. This reduces the effectiveness of the heat transfer from the debris to the CRD room atmosphere, causing the corium temperature to increase until junction 9 fails, allowing water from the enclosure to flow onto the debris. The corium temperature falls temporarily, then reheats, achieving a maximum temperature of just under 3200°F.

Case 1a was run with FMAXCP = 0.6 to ensure that 100% of the core material was forced out of the vessel at vessel failure. The volume of corium falling into the CRD room fills the sump to a level above the junction 9 elevation. When the junction fails on debris contact, approximately 33% of the core debris flows into the enclosure room. The debris remaining in the CRD room cools from 3600°F to