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ComEd

July 5, 2000

United States Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

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

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Subject: Request for a License Amendment to Permit Uprated Power Operations at Byron and Braidwood Stations

In accordance with 10 CFR 50.90, Commonwealth Edison (ComEd) Company is requesting changes to Facility Operating License Nos. NPF-72, NPF-77, NPF-37 and NPF-66, and Appendix A, Technical Specifications (TS), for Braidwood Station, Units 1 and 2,and Byron Station, Units 1 and 2, respectively. The proposed changes will revise the maximum power level specified in each unit's license and the TS definition of rated thermal power. Other TS changes associated with this power uprate amendment request are summarized in Attachment A, "Description and Safety Analysis for Proposed Changes." Once approved, associated changes to the Byron Station and Braidwood Station Core Operating Limits Report (COLR) and Pressure and Temperature Limits Report (PTLR) will also be made to reflect uprated power operations.

An increase in power will be accomplished by increasing turbine steam demand accompanied by a corresponding increase in the differential temperature across the reactor vessel.

Byron Station and Braidwood Station have completed a comprehensive uprate program to increase the licensed reactor power from 3411 Megawatts-thermal (MWt) to 3586.6 MWt for Units 1 and 2 at each station. Note that the Large Break Loss of Coolant Accident (LBLOCA) analysis to specifically address 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," at uprated power conditions, will be submitted in a separate Byron Station and Braidwood Station license amendment request in December 2000. This analysis will be performed using the NRC approved Westinghouse Best Estimate LOCA model WCOBRA/TRAC.

The uprate program included a reanalysis or evaluation of all other aspects of LBLOCA, Small Break Loss of Coolant Accidents (SBLOCA), non-LOCA accidents, and Nuclear Steam Supply System (NSSS) and balance-of-plant (BOP) structures, systems, and components. Major NSSS components (e.g., reactor pressure vessel, pressurizer, reactor coolant pumps, and steam generators); BOP components (e.g., turbine, generator, and condensate and feedwater pumps); and major systems and sub-systems (e.g., safety injection, auxiliary feedwater, residual heat removal, electrical distribution, emergency diesel generators, containment cooling, and the ultimate heat sink) have been assessed with respect to the bounding conditions expected for operation at the uprated power level. Control systems (e.g., rod control, pressurizer pressure and level, turbine overspeed, steam generator level, and steam dump) have been evaluated for operation at uprated power conditions. Reactor trip and Engineered Safety Feature (ESF) actuation setpoints have been assessed and no needed changes were identified as a result of uprated power operations. The results of all of the above analyses and evaluations have yielded acceptable results and demonstrate that all design basis acceptance criteria will continue to be met during uprated power operations.

The majority of the uprate analyses and evaluations were performed in accordance with the current Byron Station and Braidwood Station licensing bases methodologies. However, a number of specific analyses, e.g., the iodine spike factor, LOCA mass and energy release, and feedwater line break calculations were performed using new or improved methods. The specific analytical techniques used for the uprated power conditions are referenced or discussed in Attachment E, "Power Uprate Licensing Report for Byron Station and Braidwood Station." The analyses demonstrate that operation at the uprated power level can be achieved without changing the criteria that have been previously used as the bases for acceptable operation. All acceptance criteria including those for LBLOCA, SBLOCA, non-LOCA accidents, containment pressure and temperature, and radiological dose limits, continue to be met.

The power uprate analyses for Byron Station and Braidwood Station were performed consistent with the guidelines set forth in Westinghouse Energy Systems Report, WCAP-10263, "A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Power Plant," dated 1983. The WCAP was submitted to the NRC for review in a letter from E. P. Rahe (Westinghouse Electric Corporation) to C. O. Thomas (NRC), "WCAP-10263 Power Uprating Topical Report Review," dated February 11, 1983. This methodology, although not formally reviewed and approved by the NRC, was followed by North Anna, Salem, Indian Point 2, Callaway, Vogtle, and Turkey Point Nuclear Plants, for their respective power uprate initiatives. The NRC found these power uprate submittals acceptable and has most recently issued License Amendment No. 137 and No. 129 to Joseph M. Farley Nuclear Plant, Units 1 and 2, respectively on April 29, 1998, for uprated power operations. The Farley units also followed the WCAP-10263 methodology.

This amendment request is subdivided as follows.

1. Attachment A provides a description and safety analysis of the proposed changes.

2. Attachments B-1 and B-2 provide the marked up TS pages with the proposed changes indicated for Byron Station and Braidwood Station. Attachments B-3 and B-4 provide the typed TS pages with the proposed changes incorporated. The associated Bases pages are also included for informational purposes.

3. Attachment C describes our evaluation performed using the criteria in 10 CFR 50.91(a)(1), "Notice for public comment," which provides information supporting a finding of no significant hazards consideration using the standards in 10 CFR 50.92(c), "Issuance of amendment."

4. Attachment D provides information supporting an Environmental Assessment. We have determined that the proposed changes will not significantly increase the amount of any effluent which may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure.

5. Attachment E contains the "Power Uprate Licensing Report for Byron Station and Braidwood Station." This report details the evaluations and analyses to demonstrate that Byron Station and Braidwood Station can safety operate at uprated power conditions.

We request that the NRC review and, if found acceptable, approve the proposed license and TS changes by May 7, 2001, to support a mid-cycle power uprate of Byron Station, Unit 1 and Unit 2, and Braidwood Station, Unit 2, prior to the summer months of 2001. Braidwood Station Unit 1 will not implement the power uprate changes until after the fall 2001 outage as noted below. Modifications to the high pressure (HP) turbine and other BOP components will be necessary to fully support uprated power conditions. Operations at the current power level with these modifications installed, prior to implementing the power uprate changes, was evaluated and found to be acceptable. Note that the power increase for Unit 2 at both Byron Station and Braidwood Station will be less than the power increase for Unit 1 at both stations. This is because Unit 2 at each station has the original Westinghouse Model D5 Steam Generators (SGs) whereas Unit 1 at each station has new BWI SGs which are capable of a larger power increase.

Byron Station Unit 1 will perform the necessary HP turbine and BOP modifications to support uprated power conditions during the fall 2000 refueling outage. Power uprate will be implemented during mid-cycle operations upon receipt of the license amendment. It is anticipated that operating Byron Station Unit 1 at uprated power conditions will allow the additional generation of approximately 70 Megawatts-electric (MWe) of power for the summer of 2001.

Byron Station Unit 2 will perform the necessary HP turbine and BOP modifications, to support uprated power conditions, during the spring 2001 refueling outage. Power uprate will be implemented during mid-cycle operations upon receipt of the license amendment. It is anticipated that operating Byron Station Unit 2 at uprated power conditions will allow the additional generation of approximately 40 MWe of power.

Braidwood Station Unit 1 will perform the necessary HP turbine and BOP modifications to support uprated power conditions during the fall 2001 refueling outage and, assuming receipt of the license amendment, will operate the unit at the uprated power conditions upon returning Unit 1 to service following the refueling outage. It is anticipated that operating Braidwood Station Unit 1 at uprated power conditions will allow the additional generation of approximately 70 MWe of power.

Braidwood Station Unit 2 will implement the power uprate changes during mid-cycle operations upon receipt of the license amendment. Although the HP turbine and BOP modifications to support full uprated power conditions will not be performed until the spring 2002 refueling outage, we anticipate that Unit 2 will be able to immediately increase electrical power output by approximately 10 MWe. Braidwood Station will operate Unit 2 at full uprated power conditions, assuming receipt of the license amendment upon returning Unit 2 to service following the spring 2002 refueling outage. It is anticipated that operating Braidwood Station Unit 2 at the full uprated power conditions will allow the generation of approximately 40 MWe of power above the current electrical output.

These proposed changes have been reviewed and approved by the Byron and Braidwood Stations' Plant Operations Review Committees and the Nuclear Safety Review Boards in accordance with the requirements of the ComEd Quality Assurance Program.

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

Should you have any questions relative to this submittal, please contact Mr. J. A. Bauer at (630) 663-7287.

Respectfully,

R. M. Krich
Vice President – Regulatory Services

Attachments: Attachment A, Description and Safety Analysis for Proposed Changes
Attachment B-1, Marked-up Pages For Proposed Changes, Byron Station
Attachment B-2, Marked-up Pages For Proposed Changes, Braidwood Station
Attachment B-3, Incorporated Proposed Changes, Typed Pages, Byron Station
Attachment B-4, Incorporated Proposed Changes, Typed Pages, Braidwood Station
Attachment C, Information Supporting a Finding of No Significant Hazards Consideration
Attachment D, Information Supporting an Environmental Assessment
Attachment E, "Power Uprate Licensing Report for Byron Station and Braidwood Station"

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

STATE OF ILLINOIS	**)**	
COUNTY OF DUPAGE	**)**	
IN THE MATTER OF	**)**	
COMMONWEALTH EDISON (COMED) COMPANY	**)**	**Docket Numbers**
BYRON STATION UNITS 1 AND 2	**)**	**STN 50-454 AND STN 50-455**
BRAIDWOOD STATION UNITS 1 AND 2	**)**	**STN 50-456 AND STN 50-457**

SUBJECT: Request for a License Amendment to Permit Uprated Power Operations at Byron and Braidwood Stations

AFFIDAVIT

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

R. M. Krich
Vice President - Regulatory Services

Subscribed and sworn to before me, a Notary Public in and

for the State above named, this 5th day of

July, 2000.

Notary Public

" OFFICIAL SEAL "
Joseph V. Sipek
Notary Public, State of Illinois
My Commission Expires 11/24/2001

ATTACHMENT A

BYRON STATION, UNITS 1 AND 2
BRAIDWOOD STATION, UNITS 1 AND 2

DESCRIPTION AND SAFETY ANALYSIS FOR PROPOSED CHANGES

A. SUMMARY OF PROPOSED CHANGES

In accordance with 10 CFR 50.90, Commonwealth Edison (ComEd) Company is requesting changes to Facility Operating License Nos. NPF-72, NPF-77, NPF-37 and NPF-66, and Appendix A, Technical Specifications (TS), for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively. The proposed changes include:

- increasing the maximum power level specified in each unit's license;
- revising the value of Rated Thermal Power (RTP) in the TS definitions;
- revising the reference source for conversion factors in the calculation of Dose Equivalent Iodine (I) - 131 as noted in the TS definitions;
- adding a Departure from Nucleate Boiling Ratio (DNBR) limit specifically for a thimble cell;
- increasing the minimum limit for Reactor Coolant System (RCS) total flow;
- revising the steam generator laser welded sleeve plugging limit; and
- reducing the peak calculated containment internal pressure P_a, for the design basis loss of coolant accident (LOCA).

The proposed changes are based on performance of new analyses and detailed evaluations to support an increase in RTP from 3411 Megwatts-thermal (MWt) to 3586.6 MWt. The change in RTP will be incorporated into Section 2.C(1) of Facility Operating License Nos. NPF-72, NPF-77, NPF-37 and NPF-66, for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively, and to TS Section 1.1, "Definitions," for RTP. The minimum value for total RCS flow specified in TS Section 3.4, "Reactor Coolant System (RCS)," Limiting Condition for Operation (LCO) 3.4.1.c, and Surveillance Requirements (SR) 3.4.1.3 and 3.4.1.4 will be increased from ≥ 371,400 gpm to ≥ 380,900 gpm. TS Section 5.5.16, "Containment leakage Rate Testing Program," states that the peak calculated containment internal pressure for the design basis LOCA, P_a, is 47.8 psig for Unit 1 and 44.4 psig Unit 2. The Unit 1 P_a value will be changed to 42.8 psig and the Unit 2 P_a value will be changed to 38.4 psig. The associated TS Bases pages are included with this submittal for informational purposes. Upon approval of the license amendments, changes to the Byron Station and Braidwood Station Core Operating Limits Report (COLR) and Pressure and Temperature Limits Report (PTLR) will also be made to reflect uprated power operations.

An increase in power will be accomplished by increasing turbine steam demand accompanied by a corresponding increase in the differential temperature across the reactor vessel.

Byron Station and Braidwood Station have completed a comprehensive uprate program to increase the licensed reactor power from 3411 Megawatts-thermal (MWt) to 3586.6 MWt for Units 1 and 2 at each station. Note that the Large Break Loss of Coolant Accident (LBLOCA) analysis to specifically address 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," at uprated power conditions, will be submitted in a separate Byron Station and Braidwood Station license amendment request in December 2000. This analysis will be performed using the NRC approved Westinghouse Best Estimate LOCA model WCOBRA/TRAC.

The uprate program included a reanalysis or evaluation of all other aspects of LBLOCA, Small Break Loss of Coolant Accidents (SBLOCA), non-LOCA accidents, and Nuclear Steam Supply System (NSSS) and balance-of-plant (BOP) structures, systems, and components. Major NSSS components (e.g., reactor pressure vessel, pressurizer, reactor coolant pumps, and steam generators); BOP components (e.g., turbine, generator, and condensate and feedwater pumps); and major systems and sub-systems (e.g., safety injection, auxiliary feedwater, residual heat removal, electrical distribution, emergency diesel generators, containment cooling, and the ultimate heat sink) have been assessed with respect to the bounding conditions expected for operation at the uprated power level. Control systems (e.g., rod control, pressurizer pressure and level, turbine overspeed, steam generator level, and steam dump) have been evaluated for operation at uprated power conditions. Reactor trip and Engineered Safety Feature (ESF) actuation setpoints have been assessed and no needed changes were identified as a result of uprated power operations. The results of all of the above analyses and evaluations have yielded acceptable results and demonstrate that all design basis acceptance criteria will continue to be met during uprated power operations.

The majority of the uprate analyses and evaluations were performed in accordance with the current Byron Station and Braidwood Station licensing bases methodologies. However, a number of specific analyses, e.g., the iodine spike factor, LOCA mass and energy release, and feedwater line break calculations were performed using new or improved methods. The specific analytical techniques used for the uprated power conditions are referenced or discussed in Attachment E, "Power Uprate Licensing Report for Byron Station and Braidwood Station." The analyses demonstrate that operation at the uprated power level can be achieved without changing the criteria that have been previously used as the bases for acceptable operation. All acceptance criteria including those for LBLOCA, SBLOCA, non-LOCA accidents, containment pressure and temperature, and radiological dose limits, continue to be met.

The power uprate analyses for Byron Station and Braidwood Station were performed consistent with the guidelines set forth in Westinghouse Energy Systems Report, WCAP-10263, "A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Power Plant," dated 1983. The WCAP was submitted to the NRC for review in a letter from E. P. Rahe (Westinghouse Electric Corporation) to C. O. Thomas (NRC), "WCAP-10263 Power Uprating Topical Report Review," dated February 11, 1983. This methodology, although not formally reviewed and approved by the NRC, was followed by North Anna, Salem, Indian Point 2, Callaway, Vogtle, and Turkey Point Nuclear Plants, for their respective power uprate initiatives. The NRC found these power uprate submittals acceptable and has most recently issued License Amendment No. 137 and No. 129 to Joseph M. Farley Nuclear Plant, Units 1 and 2, respectively on April 29, 1998, for uprated power operations. The Farley units also followed the WCAP-10263 methodology.

The methodology described in WCAP-15315, "Reactor Vessel Closure Flange Requirements Evaluation for Operating PWR and BWR Plants," dated October 1999, is being used in the Byron Station and Braidwood Station power uprate analysis to eliminate the flange temperature requirement of 10 CFR 50 Appendix G, "Fracture Toughness Requirements." This WCAP was submitted to the NRC in a letter from H. A. Sepp (Westinghouse Electric Company) to the

Secretary of the NRC dated November 4, 1999. The NRC is currently reviewing this WCAP. Should this WCAP not be approved prior to the uprate implementation, a reanalysis will be performed and the necessary adjustments made to the PTLR.

A complete description of the proposed changes is given in Section E, "Description of the Proposed Changes," of this Attachment A. The marked-up TS pages are provided in Attachments B-1 and B-2 for Byron Station and Braidwood Station, respectively.

B. DESCRIPTION OF THE CURRENT REQUIREMENTS

Operating License Maximum Power Level

Item 2.C(1) of the current operating license for each unit at the Byron Station and the Braidwood Station states that "The licensee is authorized to operate the facility at reactor core power levels not in excess of 3411 megawatts thermal (100% power) in accordance with the conditions specified herein."

TS Definition of Rated Thermal Power

TS Section 1.1, "Definitions," defines "Rated Thermal Power (RTP)" as follows: "RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3411 MWt."

TS Definition of Dose Equivalent I-131

TS Section 1.1, "Definitions," defines Dose Equivalent I-131 as follows: "DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites."

DNBR Limit for a Thimble Cell and Typical Cell

Reactor Core Safety Limit, TS 2.1.1.1, currently states, "In Mode 1, the Departure from Nucleate Boiling Ratio (DNBR) shall be maintained $\geq$ 1.25 for the WRB-2 DNB correlation." This current requirement does not specify a difference in DNBR limits between a thimble cell and a typical cell since the limit values are the same.

RCS Total Flow Rate

TS LCO 3.4.1.c states, "RCS total flow rate $\geq$ 371,400 gpm and within the limit specified in the COLR." This LCO is applicable in Mode 1.

SR 3.4.1.3 states, "Verify RCS total flow rate is ≥ 371,400 gpm and within the limit specified in the COLR." This SR has a frequency of 12 hours.

SR 3.4.1.4 states, "Verify by precision heat balance that RCS total flow rate is ≥ 371,400 gpm and within the limit specified in the COLR." This SR has a frequency of 18 months.

Steam Generator Laser Welded Sleeve Plugging Limit

TS 5.5.9, "Steam Generator (SG) Tube Surveillance Program," Item 6, states that, "Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving in the affected area. The plugging or repair limit imperfection depth for the tubing and laser welded sleeves is equal to 40% of the nominal wall thickness. The plugging limit imperfection depth for TIG welded sleeves is equal to 32% of the nominal wall thickness."

Peak Calculated Containment Internal Pressure

TS 5.5.16, "Containment Leakage Rate Testing Program," states, "The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 47.8 psig for Unit 1 and 44.4 psig Unit 2."

C. BASES FOR THE CURRENT REQUIREMENTS

Rated Thermal Power

The current operating license and the affected TS sections are currently based on an RTP of 3411 MWt. The supporting transient and accident analyses justifying operation are also based on this RTP with appropriate margins added, in accordance with regulatory guidance. Limits placed on RTP, RCS pressure, RCS temperature and flow, ensure that DNB limits will be met for each of the transients analyzed.

All plant conditions have been placed into four categories in accordance with anticipated frequency of occurrence and potential radiological consequences to the public. The four categories are as follows.

Condition I: Normal Operation and Operational Transients
Condition II: Faults of Moderate Frequency
Condition III: Infrequent Faults
Condition IV: Limiting Faults

The basic principle applied in relating design requirements to each of the conditions is that the most probable occurrences should yield the least radiological risk to the public and those extreme situations having the potential for the greatest risk to the public shall be those least likely to occur. Where applicable, reactor trip system and engineered safeguards functioning is assumed to the extent allowed by considerations such as the single failure criterion, in fulfilling this principle.

All the applicable Condition II transients analyzed met the applicable Condition II acceptance criteria. For all Condition II transients analyzed, the calculated minimum DNBR was greater than the limit value. For each of these transients, the peak RCS pressure was less than the safety limit of 110% of design pressure (i.e., 2750 psia) and there was no failed fuel as a result of the transients. Since DNB does not occur for any Condition II transients, peak cladding temperature does not increase significantly above nominal values for these events.

All the applicable Condition III transients analyzed met the applicable Condition III acceptance criteria. For all of the applicable Condition III transients, the minimum DNBR was greater than the limit value and there was no failed fuel except for the single Rod Cluster Control Assembly (RCCA) withdrawal at full power transient. For this transient, the upper bound of the number of fuel rods experiencing DNBR was less than the limit value of 5% of the total rods in the core. All of the applicable Condition III transients experienced a peak RCS pressure less than 2750 psia. The only Condition III transient for which cladding temperature was calculated was the SBLOCA and yielded a peak value less than 2200°F.

All the applicable Condition IV transients analyzed met the applicable Condition IV acceptance criteria. For the locked rotor event, DNB was assumed to occur at the initiation of the transient and the peak cladding temperature was calculated to be less than 2700°F. The LOCA analysis demonstrated that the amount of failed fuel calculated could be ≤ 100%; and for rod ejection it was ≤ 10%. The amount of failed fuel for a major break of a steamline was ≤ 1%. No failed fuel resulted for the feedwater line break or for the steam generator tube rupture. All of the applicable Condition IV transients experienced a peak RCS pressure less than 2750 psia. The peak cladding temperature calculated for LOCA was less than 2200°F and the peak cladding temperature for rod ejection was less than 2700°F.

TS Definition of Dose Equivalent I-131

The current TS definition of I-131 uses thyroid dose conversion factors from Table III of Atomic Energy Commission (AEC) Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," dated 1962. These conversion factors are based on information contained in International Commission on Radiological Protection (ICRP) Publication 2, "Permissible Dose for Internal Radiation," which was an acceptable revision of the ICRP publication at the time that this TS Definition was written.

DNBR Limit for a Thimble Cell and Typical Cell

The overall objective of the thermal and hydraulic design of the reactor core is to provide adequate heat transfer which is compatible with the heat generation distribution in the core such that heat removal by the RCS or the emergency core cooling system (ECCS), when applicable, assures that the following performance and safety criteria requirements are met.

a. Fuel damage, defined as penetration of the fission product barrier, (i.e., the fuel rod cladding), is not expected during normal operation and operational transients (i.e., Condition I events) or any transient conditions arising from faults of moderate frequency (i.e., Condition II events). It is not possible, however, to preclude a very small number of rod failures. These rod failures will be within the capability of the plant cleanup system and are consistent with the plant design bases.

b. The reactor can be brought to a safe condition following a Condition III event with only a small fraction of fuel rods damaged although sufficient fuel damage might occur to preclude immediate resumption of operation.

c. The reactor can be brought to a safe condition and the core can be kept subcritical with acceptable heat transfer geometry following transients arising from Condition IV events.

In order to satisfy the above requirements, the following design bases have been established for the thermal and hydraulic design of the reactor core. There will be at least a 95% probability that DNB will not occur on the limiting fuel rods during normal operation and operational transients, and any transient conditions arising from faults of moderate frequency (i.e., Condition I and II events), at the 95% confidence level.

The design method employed to meet the DNB design basis for the VANTAGE 5/VANTAGE+ fuel assemblies, which are currently utilized at Byron Station and Braidwood Station, is the revised thermal design procedure (RTDP). The RTDP methodology is described in WCAP 11397-P-A dated April 1989. This WCAP was submitted to the NRC for review in a letter from W. J. Johnson (Westinghouse Electric Corporation) to J. Lyons (NRC), "Submittal of Westinghouse Topical, WCAP-11397, 'Revised Thermal Design Procedure,' for Review and Approval," dated March 16, 1987. The WCAP was subsequently approved by the NRC in a letter from A. C. Thadani (NRC) to W. J. Johnson (Westinghouse Electric Corporation), "Acceptance for Referencing of Licensing Topical Report WCAP-11397, 'Revised Thermal Design Procedure,'" dated January 17, 1989. With the RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and DNB correlation predications are considered statistically to obtain DNB uncertainty factors. Based on the DNB uncertainty factors, RTDP design limit DNBR values are determined such that there is at least a 95% probability at a 95% confidence level that DNB will not occur on the most limiting fuel rod during normal operation and operational transients and during transient conditions arising from faults of moderate frequency (i.e., Condition I and II events). Since the parameter uncertainties are considered in determining the RTDP design limit DNBR values, the plant safety analyses are performed using input parameters at their nominal values.

The RTDP design limit DNBR value is 1.25 for the typical and thimble cells in VANTAGE 5/VANTAGE+ fuel. The design limit DNBR values are used as a basis for the TS.

To maintain DNBR margin to offset DNB penalties, such as those due to fuel rod bow, safety analyses were performed for DNBR limits higher than the design limit DNBR values. The difference between the design limit DNBRs and the safety analysis limit DNBRs results in available DNBR margin. The net DNBR margin, after consideration of all penalties, is available for operating and design flexibility.

By preventing DNB, adequate heat transfer is ensured between the fuel cladding and the reactor coolant, thereby preventing cladding damage as a result of inadequate cooling. Limits provided by the nuclear control and protection systems are such that this design basis will be met for transients associated with Condition II events including overpower transients. There is an additional large DNBR margin at rated power operation and during normal operating transients.

RCS Total Flow Rate

RCS total flow rate, along with RCS pressure and temperature, comprise the DNB limits assumed in the safety analyses. The safety analyses for normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on RCS pressure, temperature, and flow rate ensure that the DNB limits will be met for each of the transients analyzed.

The RCS flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit corresponds to that assumed for DNB analyses. Flow rate indications are averaged to come up with a value for comparison to the limit. A lower RCS flow will cause the core to approach DNB limits.

Operation for significant periods of time outside these DNB limits increases the likelihood of a fuel cladding failure in a DNB limited event.

The requirements of this TS LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses. The safety analyses have shown that transients initiated from the limits of this TS LCO will result in meeting the DNBR acceptance criterion. This is the acceptance limit for the RCS DNB parameters. Changes to the unit that could impact these parameters must be assessed for their impact on the DNBR acceptance criteria. The transients analyzed include loss of coolant flow events and dropped or stuck control rod events.

Current safety analyses assumed a total RCS flow rate of 358,800 gpm. This value is bounded by the TS LCO value of 371,400 gpm assuming a flow measurement uncertainty of 3.5%.

TS LCO 3.4.1 specifies limits on the monitored process variables , i.e., pressurizer pressure, RCS average temperature (T_{avg}), and RCS total flow rate , to ensure the core operates within the limits assumed in the safety analyses. These variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. However, the minimum RCS flow, based on maximum analyzed steam generator tube plugging, is retained in the TS LCO to assure that a lower flow rate than reviewed by the NRC will not be used. Operating within these limits will result in meeting the DNBR design criterion in the event of a DNB limited transient.

The DNBR limit is provided in TS Safety Limit (SL) 2.1.1, "Reactor Core SLs." TS LCO 3.4.1 represents the initial conditions of the safety analysis which are far more restrictive than the conditions which define the DNBR limit. Should a violation of this LCO occur, the operator must determine whether or not an SL may have been exceeded.

In Mode 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS total flow rate must be maintained during steady state operation in order to ensure DNB design criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other Modes, the power level is low enough that DNB is not a concern.

<u>SR 3.4.1.3</u>

The 12 hour Surveillance Frequency for RCS total flow rate is performed using the installed control room flow instrumentation. The required minimum RCS flow rate is determined by performing a precision calorimetric for each unit at the beginning of the fuel cycle. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation within safety analysis assumptions.

<u>SR 3.4.1.4</u>

Measurement of RCS total flow rate by performance of a precision calorimetric heat balance once every 18 months allows the installed RCS flow instrumentation to be calibrated and verifies the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate.

The frequency of 18 months reflects the importance of verifying flow after a refueling outage when the core has been altered, which may have caused an alteration of flow resistance.

Steam Generator Laser Welded Sleeve Plugging Limit

Analysis has shown that it is necessary to plug, (i.e., remove from service), steam generator (SG) tubes with laser welded sleeves when an imperfection depth of 40% of the nominal wall thickness is indicated. The removal of SG tubes with laser welded sleeves from service upon discovering an imperfection of 40% wall thickness, ensures the structural integrity of SG tubes which have been sleeved and precludes the occurrence of a SG tube rupture from these tubes under all operating conditions.

Peak Calculated Containment Internal Pressure

The containment consists of the concrete containment building, its steel liner, and the penetrations through this structure. The structure is designed to contain radioactive material that may be released from the reactor core following a design basis LOCA. Additionally, this structure provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment is a reinforced concrete structure with a cylindrical wall, a flat foundation mat, and a shallow dome roof. The inside surface of the containment is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions. The cylinder wall is prestressed with a post tensioning system in the vertical and horizontal directions, and the dome roof is prestressed utilizing a three way post tensioning system. The concrete containment building is required for structural integrity of the containment under Design Basis Accident (DBA) conditions. The steel liner and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment operable, limits the leakage of radioactive material from the containment to the environment.

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBAs that result in a challenge to containment operability from high pressures and temperatures are a LOCA and a steam line break. In addition, release of significant radioactive material within containment can occur from a LOCA, secondary system pipe break, or fuel handling accident. In the DBA analyses, it is assumed that the containment is operable such that, for the DBAs involving release of radioactive material, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.10% of containment air weight per day. This leakage rate, used to evaluate offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," Option B, as L_a, the maximum allowable containment leakage rate at the calculated peak containment internal pressure, P_a, resulting from the limiting design basis LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.10% per day in the safety analysis at P_a = 47.8 psig for Unit 1 and P_a = 44.4 psig for Unit 2.

In Modes 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In Modes 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these Modes. Therefore, containment is not required to be operable in Mode 5 or Mode 6 to prevent leakage of radioactive material from containment.

D. NEED FOR REVISION OF THE REQUIREMENTS

The proposed changes would allow an increase in licensed core thermal power from 3411 MWt to 3586.6 MWt and allow ComEd to increase the electrical output of the Byron Station Unit 1 and the Braidwood Station Unit 1 by approximately 70 MWe each, and to increase the electrical output of Byron Station Unit 2 and Braidwood Station Unit 2 by approximately 40 MWe each. Note that the power increase for Unit 2 at both Byron Station and Braidwood Station will be less than the power increase for Unit 1 at both stations. This is because Unit 2 at each station has the original Westinghouse Model D5 SGs whereas Unit 1 at each station has new BWI SGs which are capable of a larger power increase.

The beneficial aspects of this power uprate is that a total increase of approximately 220 MWe in generation, from all four units, is expected and will provide additional electric power to service commercial and domestic loads on the ComEd electrical grid. This power increase is needed to help meet the growth demand on the ComEd electrical distribution system while avoiding major capital expenditures associated with building new generating capacity. The power uprate program will also result in direct displacement of higher cost fossil fuel generation with lower cost nuclear fuel generation.

E. DESCRIPTION OF THE PROPOSED CHANGES

Unless otherwise stated, the affected operating license pages and TS sections are the same for Unit 1 and Unit 2 of the Byron Station and the Braidwood Station.

Operating License Maximum Power Level

Item 2.C(1) of the current operating license for each unit at Byron Station and Braidwood Station states that, "The licensee is authorized to operate the facility at reactor core power levels not in excess of 3411 megawatts thermal (100% power) in accordance with the conditions specified herein." The core power level will be revised to 3586.6 MWt.

TS Definition of Rated Thermal Power

TS Section 1.1, "Definitions," defines "Rated Thermal Power (RTP)" as follows: "RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3411 MWt." The value for RTP will be revised to 3586.6 MWt.

TS Definition of Dose Equivalent I-131

TS Section 1.1, "Definitions," defines Dose Equivalent I-131 as follows: "DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites." Two updated reference sources for the thyroid dose conversion factors for this calculation will be added to this definition. These reference sources are Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision. 1, 1977, and ICRP 30, "Limits for Intakes of Radionuclides by Workers," Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity." The TS statement will be revised to state, "...The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, 1977, or ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."

DNBR Limit for a Thimble Cell

TS SL 2.1.1.1 states, "In Mode 1, the Departure from Nucleate Boiling Ratio (DNBR) shall be maintained ≥ 1.25 for the WRB-2 DNB correlation." This requirement is applicable for both a thimble cell and a typical cell. This SL will be changed to state, "In Mode 1, the Departure from Nucleate Boiling Ratio (DNBR) shall be maintained ≥ 1.24 for the WRB-2 DNB correlation for a thimble cell and ≥ 1.25 for the WRB-2 DNB correlation for a typical cell."

RCS Total Flow Rate

TS LCO 3.4.1.c states, "RCS total flow rate ≥ 371,400 gpm and within the limit specified in the COLR." The value for RCS total flow rate will be revised to ≥ 380,900 gpm.

SR 3.4.1.3 states, "Verify RCS total flow rate is ≥ 371,400 gpm and within the limit specified in the COLR." The value for RCS total flow rate will be revised to ≥ 380,900 gpm.

SR 3.4.1.4 states, "Verify by precision heat balance that RCS total flow rate is ≥ 371,400 gpm and within the limit specified in the COLR." The value for RCS total flow rate will be revised to ≥ 380,900 gpm.

Steam Generator Laser Welded Sleeve Plugging Limit

TS Section 5.5.9, "Steam Generator (SG) Tube Surveillance Program," Item 6, states that, "Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving in the affected area. The plugging or repair limit imperfection depth for the tubing and laser welded sleeves is equal to 40% of the nominal wall thickness. The plugging limit imperfection depth for TIG welded sleeves is equal to 32% of the nominal wall thickness."

This statement will be revised to read, "...The plugging or repair limit imperfection depth for the tubing is equal to 40% of the nominal wall thickness. The plugging limit imperfection depth for laser welded sleeves is equal to 38.7% of the nominal wall thickness...."

Peak Calculated Containment Internal Pressure

TS Section 5.5.16 states "The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 47.8 psig for Unit 1 and 44.4 psig Unit 2." P_a for Unit 1 will be revised to 42.8 psig and P_a for Unit 2 will be revised to 38.4 psig.

F. SAFETY ANALYSIS OF THE PROPOSED CHANGES

Overview

The proposed change to RTP and all associated parameters is supported by the overall power uprate program results. The Byron Station and Braidwood Station Updated Final Safety Analysis Report (UFSAR) was used as the baseline document for analysis purposes. The analyses demonstrate that operation at the uprated power level can be achieved without changing the acceptance criteria on safety limits that have previously been used as the design basis for acceptable operation in the UFSAR. All acceptance criteria including those for LBLOCA, SBLOCA, non-LOCA accidents, containment pressure and temperature, and radiological dose limits, continue to be met. As noted earlier, analysis of the LBLOCA under power uprate conditions for compliance with 10 CFR 50.46 criteria will be submitted in December 2000.

The TS RCS DNB parameters of pressure, temperature and flow, were calculated based on the most limiting analytical values plus indication uncertainties. All values reflect Byron Station and Braidwood Station specific analytical values used to verify the adequacy of safe operation at the proposed uprated power level. Instrumentation uncertainties associated with Byron Station and Braidwood Station calibration practices and equipment have been included in the calculation of the final indicated values.

The proposed revisions to the Byron Station and Braidwood Station operating licenses and TS will allow an increase in RTP from 3411 MWt to 3586.6 MWt with no significant increase in risk or environmental impact.

Upon issuance of the approved amendment, certain TS referenced limits contained in the COLR and PTLR will be made to reflect uprated power operations.

After implementation of uprated power operations, the routine comprehensive operator logs will be used to verify proper equipment operation. These logs are taken in the control room and in the physical plant.

Increase in RTP from 3411 MWt to 3586.6 MWt

The detailed analyses, documented in Attachment E, "Power Uprate Licensing Report for Byron Station and Braidwood Station," demonstrate that Units 1 and 2 of the Byron Station and Units 1 and 2 of the Braidwood Station, can operate safely with the proposed five percent increase in maximum core thermal power.

The following discussion summarizes the information provided in Attachment E.

Nuclear Steam Supply System (NSSS) Parameters

The power uprate project included NSSS performance analyses to develop bounding NSSS Performance Capability Working Group (PCWG) Parameters for use in the analyses and evaluations of the NSSS, including NSSS design transients, systems, components, accidents, and nuclear fuel.

The NSSS PCWG parameters are the fundamental parameters which are used as input in all the NSSS analyses. They provide the RCS and secondary system conditions (i.e., temperatures, pressures and flows) that are used as the basis for the design transients, systems, components, accidents, and fuel analyses and evaluations.

The PCWG parameters are established using conservative assumptions in order to provide bounding conditions to be used in the NSSS analyses. For example, the RCS flow assumed in generating the primary and secondary side conditions is the Thermal Design Flow (TDF), which is a conservatively low flow that accounts for flow measurement uncertainty and assumes the maximum steam generator tube plugging (SGTP) level. The PCWG parameters were determined such that the Byron Station and the Braidwood Station would have operating flexibility; therefore, a range of conditions was set on the RCS average temperature (T_{avg}) and the SGTP level. The T_{avg} range was specified between 575°F and 588°F, while the SGTP level can vary from 0% to 5% for the new BWI SGs for Byron Unit 1 and Braidwood Unit 1 and 0% to 10% for the original Westinghouse D5 SGs for Byron Unit 2 and Braidwood Unit 2. An uprated NSSS power level of 3600.6 MWt and the new TDF value of 92,000 gpm/loop were also used to generate the PCWG parameters. The NSSS power level of 3600.6 MWt is the sum of 3586.6 MWt core thermal power plus 14 MWt from reactor coolant pump heat.

The primary acceptance criteria for the determination of the PCWG parameters were that they would pose as few potential feasibility issues as possible for the uprate project from an analysis perspective, and that they provide Byron Station and Braidwood Station with adequate flexibility and margin in the operation of the plants. The PCWG parameters are given in Attachment E, Table 2.1-1 for Unit 1 and Table 2.1-2 for Unit 2. The PCWG parameters are evaluated throughout Attachment E.

NSSS Design Transients

The current NSSS design transients were analyzed for their continued applicability at uprated power and the resulting transient curves were provided to all system and component designers for use in their specific analyses. Auxiliary equipment design transients were also evaluated and were determined to remain applicable for use in the uprating analysis of all auxiliary equipment in the NSSS.

The PCWG parameters for the original and proposed uprate parameters were compared. This evaluation confirmed that all current design transients, with the exception of the "loss of load" transient, (i.e., a Condition II event) remain applicable. The "loss of load" transient was further analyzed and found to be acceptable. These Byron Station and Braidwood Station specific design transients have been used in the NSSS component and fatigue analyses. These evaluations are presented in Attachment E, Chapter 5, "NSSS Components." In summary, all NSSS components were shown to be capable of performing their design functions at uprated power conditions without modification.

A review of the current auxiliary equipment transients determined that the only transients that could be potentially impacted by the uprating are those temperature transients that are impacted by full load NSSS operating temperatures, namely T_{hot} and T_{cold}. These transients are currently based on an assumed full load NSSS worst case T_{hot} of 630°F and worst case T_{cold} of 560°F. These NSSS temperatures were originally selected to ensure that the resulting design transients would be conservative for a wide range of NSSS operating temperatures.

The PCWG parameter ranges for T_{hot} (i.e., 608.0°F to 620.3°F) and T_{cold} (i.e., 542.0°F to 555.7°F), are less limiting than the temperature ranges which established the current auxiliary equipment design transients. Since the actual temperature transients are less severe than the current design temperature transients, it is concluded that these design transients remain applicable at uprated power conditions.

<u>NSSS Systems</u>

Evaluations and analyses were performed to confirm that the NSSS systems continue to perform their intended design basis functions under the uprated power conditions. The systems addressed fall into three categories: Fluid Systems, NSSS/BOP Interface Systems, and NSSS Control Systems.

The Fluid Systems reviewed include the:

- Reactor Coolant System;
- Chemical and Volume Control System;
- Residual Heat Removal System;
- Emergency Core Cooling System;
- Boron Thermal Regeneration System;
- Component Cooling Water System;
- Boron Recycle System;
- Sampling System; and
- Waste Processing System.

The NSSS/BOP Interface Systems reviewed include the:

- Main Steam System;
- Steam Dump System;
- Condensate Feedwater System;
- Auxiliary Feedwater System; and
- Steam Generator Blowdown System.

The NSSS Control Systems analysis included:

- Pressure Relief Component Sizing (i.e., pressurizer power-operated relief valves (PORVs), pressurizer spray valves, pressurizer heaters, and steam dump relieving capacity);
- Control Systems Setpoints Analysis (i.e., load rejection steam dump controller, plant trip controller setpoints, pressurizer pressure control system, pressurizer level control system, rod control system, and turbine load control); and
- Low Temperature Overpressure Protection System Setpoint Analysis.

The results of the NSSS analysis demonstrated that all systems were capable of performing their current design basis functions with either no changes or with appropriate changes to programs, setpoints and alarms. Attachment E, Chapter 4.0, "NSSS Systems," presents the details of this analysis and associated results.

<u>NSSS Components</u>

Evaluations were performed to determine the effects of the power uprate parameters on the NSSS components. In general, the uprate-related inputs used for these evaluations were the PCWG parameters and the NSSS design transient changes. Additional input parameters specific to particular components (e.g., NSSS auxiliary equipment design transients for the auxiliary equipment evaluations) were also considered in the evaluation. The purpose of the evaluations performed for the NSSS components was to confirm that they continue to satisfy the applicable codes, standards, and regulatory guides under uprated power conditions.

Evaluations were performed in the following areas:

- Reactor Vessel Structural Evaluations and Integrity;
- Reactor Pressure Vessel System;
- Fuel Assemblies;
- Control Rod Drive Mechanisms;
- Reactor Coolant Loop Piping and Supports;
- Reactor Coolant Pumps and Motors;
- Steam Generators;
- Pressurizer;
- NSSS Auxiliary Equipment; and
- Loop Stop Isolation Valves.

The results of these analyses indicated that uprated power operations will not have a significant effect on any of the NSSS components. All NSSS components are capable of performing their current design basis functions. Attachment E, Chapter 5.0, "NSSS Components," presents the details of these analyses and associated results.

NSSS Accident Analyses

NSSS accident analyses were performed in support of the power uprate program. The accident analysis areas addressed included:

- Small Break Loss of Coolant Accident, Hot Leg Switchover, and Post-LOCA Long Term Cooling;
- Non-LOCA Events;
  - Excessive Heat Removal Due to Feedwater System Malfunctions,
  - Excessive Increase in Secondary Steam Flow,
  - Inadvertent Opening of a Steam Generator Relief or Safety Valve,
  - Steam System Piping Failure at Zero Power,
  - Steam System Piping Failure at Full Power,
  - Loss of External Electrical Load and/or Turbine Trip,
  - Loss of Non-emergency AC Power to the Plant Auxiliaries,
  - Loss of Normal Feedwater,
  - Feedwater System Pipe Break,
  - Partial Loss of Forced Reactor Coolant Flow,
  - Complete Loss of Forced Reactor Coolant Flow,
  - Single Reactor Coolant Pump Locked Rotor/Shaft Break,
  - Uncontrolled RCCA Withdrawal From a Subcritical or Low Power Startup Condition,
  - Uncontrolled RCCA Bank Withdrawal at Power,
  - Rod Cluster Control Assembly Misoperation,
  - Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature,
  - Chemical and Volume Control System Malfunction That Results in a Decrease in Boron Concentration in the Reactor Coolant,
  - Inadvertent Loading of a Fuel Assembly into an Improper Position,
  - Rod Cluster Control Assembly Ejection,
  - Inadvertent Operation of the ECCS During Power Operation,
  - Inadvertent Opening of a Pressurizer Safety or Relief Valve,
- Steam Generator Tube Rupture Transient;
- LOCA Containment Integrity;
- Main Steamline Break Consequences;
- LOCA Hydraulic Forces; and
- Radiological Consequences of the following accidents:
  - Main Steamline Break,
  - Locked Reactor Coolant Pump (RCP) Rotor,
  - Locked RCP Rotor with PORV Failure,
  - Rod Ejection,
  - Small Line Break Outside Containment,
  - Steam Generator Tube Rupture,
  - Large-Break Loss-of-Coolant Accident,
  - Small-Break Loss-of-Coolant Accident,
  - Waste Gas Decay Tank Rupture,
  - Liquid Waste Tank Failure, and

- Fuel Handling Accident.

The analyses and evaluations indicate that, for all accident scenarios listed above, the requirements of 10 CFR 50.46 continue to be met (i.e., except for LBLOCA as noted below), DNBR limits are maintained; therefore, no fuel or cladding damage is predicted, and no significant changes occur to the resultant radiological consequences of any accident due to uprated power conditions. 10 CFR 100 limits continue to be met. Attachment E, Chapter 6.0, "NSSS Accident Analyses," presents the details of these analyses and associated results.

As previously noted, the LBLOCA analysis to specifically address the criteria of 10 CFR 50.46 under uprated power conditions will be submitted in a separate Byron Station and Braidwood Station license amendment request in December 2000. This analysis will be performed using the NRC approved Westinghouse Best Estimate LOCA model WCOBRA/TRAC.

All the applicable Condition IV transients analyzed met the applicable Condition IV acceptance criteria. For the rod ejection event, DNB was assumed to occur at the initiation of the transient. The analysis demonstrated that up to 100% failed fuel was projected for the LOCA results, and ≤ 10% for the rod ejection accident. No failed fuel resulted for the feedwater line break, steam line break, or for the steam generator tube rupture event. All of the applicable Condition IV transients experienced a peak RCS pressure less than 2750 psia. The peak cladding temperature calculated for SBLOCA was less than 2200°F. For the locked rotor event, the peak clad temperature was calculated to be less than 2700°F.

Attachment E, Section 6.2.17, "Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant," addresses a boron dilution event in Mode 1 "Power Operation," Mode 2 "Startup," and Mode 6 "Refueling." The boron dilution analysis for Mode 3 "Hot Standby," Mode 4 "Hot Shutdown," and Mode 5 "Cold Shutdown," was not addressed, as this analysis was submitted to the NRC in a separate submittal via a letter from R. M. Krich to the NRC, "Request for Technical Specification Change, Revise the Applicability of Technical Specification 3.3.9, 'Boron Dilution Protection System (BDPS),'" dated June 19, 2000. This amendment request proposes to revise the applicability of TS 3.3.9 and remove the existing automatic valve actuation feature of the BDPS on detection of neutron flux doubling. To ensure adequate boron dilution protection, two redundant Volume Control Tank high level alarms will be added along with revising procedures and controls. TS Section 3.3.9, "Boron Dilution Protection System (BDPS)," currently requires that two trains of BDPS be operable in Modes 3, 4, and 5. Due to uncertainties associated with the flux doubling circuit, if TS LCO 3.3.9 cannot be met, both Byron Station and Braidwood Station follow TS Required Actions D1, close the unborated water isolation values within one hour, D2, verify that the shutdown margin is within the limits specified in the COLR within one hour and once per 12 hours thereafter, and D3, verify that the unborated water isolation valves are closed once per 12 hours. The approval of the power uprate license amendment request is not dependent on the NRC review and approval of the subject boron dilution analysis for Modes 3, 4, and 5, as Byron Station and Braidwood Station will continue to follow the above specified TS Required Actions should TS LCO 3.3.9 not be met, as is currently done.

Byron Station and Braidwood Station will perform a re-evaluation of the Individual Plant Examination (IPE). This evaluation will verify that current IPE assumptions, compensatory actions, and success paths remain valid under uprated power conditions. This evaluation will be completed by December 1, 2000.

Nuclear Fuel

Analyses were performed in support of the power uprate project in the nuclear fuel and fuel-related areas. The specific areas addressed are:

- Core Thermal-Hydraulic Design;
- Reactor Core Design;
- Fuel Rod Design and Performance;
- Reactor Internals Heat Generation Rates;
- Neutron Fluence; and
- Radiation Source Terms.

The RCS at Byron Station, Units 1 and 2 and Braidwood Station, Units 1 and 2 are similar. The analyses performed accounted for known differences related to the installed SGs at both the Byron Station and the Braidwood Station. Unit 1 at each station has BWI replacement SGs, and Unit 2 at each station has the original Westinghouse D5 SGs.

The analysis indicated that all aspects of fuel performance remain acceptable under uprated power operations. Attachment E, Chapter 7.0, "Nuclear Fuel," presents the details of this analysis and associated results.

Turbine Generator (TG)

The main TGs have been evaluated for their ability to operate at the uprated inlet steam conditions that will be supplied by the NSSS after the power uprate changes are implemented. These design conditions correspond to a revised maximum NSSS power level of 3600.6 MWt. Two sets of inlet conditions were identified as being required. This was necessary because Byron Station Unit 1 and Braidwood Station Unit 1 are operating with replacement BWI SGs, while Byron Station Unit 2 and Braidwood Station Unit 2 are operating with the original Westinghouse D5 SGs. Differences in the BWI and Westinghouse D5 SG designs resulted in two sets of design steam inlet conditions for the steam turbines.

The following TG components and systems at the Byron Station and the Braidwood Station were reviewed:

- HP Turbines;
- LP Turbines;
- Generators;
- Exciters;

- Moisture Separator Reheaters;
- Heat Exchangers (Generator Hydrogen Coolers, Exciter Air Coolers, Air and Hydrogen Side Seal Oil Coolers, Generator Stator Water Coolers, Lube Oil Coolers);
- Gland Sealing Steam Systems;
- Lube Oil Systems;
- Turbine Control Systems;
- Steam Generator Feed Pump Turbines;
- Steam Admission Valves; and
- Turbine Steam Piping Systems (Main Inlet, Extraction, Crossover, Crossunder).

The basis for this evaluation was a review of the above TG components and systems at the expected design steam conditions. Steam is to be supplied to the steam turbines by the NSSS at the uprated power level. These conditions were compared to the applicable design criteria to determine the acceptability of operation at the higher power level. Unit history records maintained by Siemens-Westinghouse (i.e., the TG supplier) were reviewed to ensure that the latest TG conditions and configurations were evaluated.

The study results identified that certain TG components required either replacement or modification to operate acceptably at the designed NSSS 3600.6 MWt uprated power level. Those components will be upgraded over the course of implementing the Byron Station and Braidwood Station power uprate. All other related TG components and systems have been determined to be acceptable for the plants to operate safely at the uprated power level. Attachment E, Chapter 8.0, "Turbine Generator," presents the details of this analysis and associated results.

<u>BOP Systems, Structures and Components</u>

BOP systems, structures, and components (SSCs) were assessed to verify that they are structurally and functionally capable of safe, reliable operation at the uprated power conditions. The study included a review of major components and systems typically impacted by a power uprate. The following systems and components were evaluated:

- Main Steam System and the Steam Dump System;
- Heater Drains System;
- Condensate and Feedwater System;
- Steam Generator Blowdown System;
- Extraction Steam System;
- Circulating Water System;
- Essential Service Water System;
- Non-Essential Service Water System;
- Component Cooling Water System;
- Spent Fuel Pool Fuel Cooling;
- Ultimate Heat Sink;
- Reactor Containment Cooling System;

- Auxiliary Feedwater System;
- Combustible Gas Control;
- NSSS/ECCS Support Systems;
- Instrumentation and Controls;
- Electrical Systems;
- Heating, Ventilation, and Air Conditioning System;
- Miscellaneous Systems;
- Piping and Supports; and
- Equipment Qualification.

In addition, a radiological evaluation was conducted that addressed:

- normal operation dose rates and shielding;
- normal operations annual radwaste effluent releases;
- post-accident access to vital areas; and
- radiological equipment qualification (EQ) for equipment in the Byron Station and Braidwood Station EQ Program.

The containment, steam pipe tunnels and valve rooms, and spent fuel pool structures were also evaluated to verify that they remain structurally sound under the effects of uprated power operations.

The analysis indicated that all BOP systems and components, some with applicable modifications, are capable of performing their design basis functions under uprated power conditions.

The radiological evaluation demonstrated that the power uprate did not have a significant effect on normal operational dose rates, radwaste effluent releases, post-accident access to vital areas, or equipment qualification.

The analysis also confirmed that the containment, steam pipe tunnels and valve rooms, and spent fuel pool structures are not significantly affected by uprated power operations. Attachment E, Chapter 9.0, "BOP Systems, Structures and Components," presents the details of these analyses and associated results.

Program Reviews

The power uprate effort has the potential to affect programs which demonstrate that various topical areas comply with design and licensing requirements. The following plant programs were reviewed. It is noted that not all of the listed programs are regulatory requirements.

- Plant Simulator;
- Fire Protection;
- Check Valve Program;

- Motor Operated Valve Program;
- Air Operated Valve Program;
- Heat Exchanger Program;
- Inservice Inspection Program;
- Inservice Testing Program;
- Containment Integrity (i.e., 10 CFR 50, Appendix J);
- High Energy Line Break (HELB);
- Special HELB in Turbine Building;
- Human Factors;
- Internal Containment Flooding;
- Station Blackout;
- Internal Missiles;
- Anticipated Transient Without Scram;
- Flow Accelerated Corrosion; and
- Programs listed in TS Section 5.5, "Programs and Manuals."

This review did not identify any program that would be compromised due to the power uprate effort. Attachment E, Chapter 10.0, "Program Reviews," presents the details of this analysis and associated results.

Environmental Impacts Review

This assessment determined whether operation at uprated power conditions will cause the plants to exceed the effluent discharge limitations and other environmental conditions associated with operation of the plants. This review is based upon information contained in the Environmental Report and the latest National Pollutant Discharge Elimination System (NPDES) Permit for the Byron Station and the Braidwood Station. The Byron Station and the Braidwood Station NPDES Permits cover discharge limitations, monitoring, and reporting requirements. The permits include restrictions on various normal plant operations and effluent limitations including cooling tower blowdown (i.e., for Byron Station only), wastewater treatment, and operation of the radwaste treatment system.

This review concluded that the non-radiological environmental impacts related to the proposed power uprate operations will not have a significant adverse affect on the current NPDES Permits or other plant administrative limits. Attachment E, Chapter 11.0, "Environmental Impacts Review," presents the details of this analysis and associated results.

Station Procedures Impact

The power uprate effort has the potential to affect plant procedures used to operate and maintain the facility in accordance with design basis and licensing requirements. Procedures that are affected must be identified, revised, reviewed, approved, and where required, training conducted, prior to implementation of uprated power operations.

Procedures that are identified as being affected by the power uprate will be revised prior to the uprate implementation. These procedures will be reviewed and approved in accordance with the station procedure change program. Priority will be given to those operational procedures that require operator training. Attachment E, Chapter 12.0, "Station Procedures Impact," presents the details of this analysis and associated results.

Conclusions

The analyses have demonstrated that all systems and components can adequately perform their design basis function under uprated power conditions. All acceptance criteria including those for LBLOCA, SBLOCA, non-LOCA accidents, containment pressure and temperature, and radiological dose limits, continue to be met.

TS Definition of Dose Equivalent I-131

NUREG 1431, "Standard Technical Specifications, Westinghouse Plants," Revision 1, dated April 1995, allows Dose Equivalent I-131 to be calculated using any one of three dose conversion factors; Table III of TID-14844, AEC, 1962, Table E-7 of NRC Regulatory Guide 1.109, Rev. 1, 1977, or ICRP 30, Supplement to Part 1. Using thyroid dose conversion factors other than those given in TID-14844 results in lower doses and higher allowable activity but is justified by the discussion given in the Federal Register (i.e., Federal Register (FR) page 23360 Vol. 56, May 21, 1991). This discussion accompanied the final rulemaking on 10 CFR 20, "Standards for Protection Against Radiation," by the NRC. In that discussion, the NRC stated that it was incorporating modifications to existing concepts and recommendations of the ICRP into NRC regulations. Incorporation of the methodology of ICRP-30 into the 10 CFR 20 revision was specifically mentioned with the changes being made resulting from changes and updates in the scientific techniques and parameters used in calculating dose. This FR reference clearly shows that the NRC was updating 10 CFR 20 to incorporate ICRP-30 recommendations and data. Regulatory Guide 1.109 thyroid dose conversion factors are higher than the ICRP-30 thyroid dose conversion factors for all five iodine isotopes of concern. Therefore, using Regulatory Guide 1.109 thyroid dose conversion factors to calculate Dose Equivalent I-131 is more conservative than ICRP-30 and is therefore acceptable. Having all three references available provides flexibility such that a license amendment will not be necessary should one of the additional reference sources be utilized.

ICRP-30 is the updated reference source used in the power uprate accident analysis radiological evaluation and must be reflected in the Dose Equivalent I-131 definition. As previously noted, the current version of 10 CFR 20, utilizes ICRP-30 data.

DNBR Limit for a Thimble Cell and Typical Cell

The thermal-hydraulic design criteria and methods for the power uprate remain the same as those presented in the Byron Station and the Braidwood Station UFSAR.

The design method employed to meet the DNB design basis for the VANTAGE 5/VANTAGE+ fuel is the RTDP. With the RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and DNB correlation predictions are considered statistically to obtain DNB uncertainty factors. Based on the DNB uncertainty factors, RTDP design limit DNBR values were determined such that there is at least a 95% probability at a 95% confidence level that DNB will not occur on the most limiting fuel rod during normal operation, operational transients, or transient conditions arising from faults of moderate frequency (i.e., Conditions I and II events).

Uncertainties in plant operating parameters (i.e., pressurizer pressure, primary coolant temperature, reactor power, and RCS flow) are shown in Attachment E, Table 7.1-2. The current plant operating parameter uncertainties, and the uncertainties used in the uprate RTDP analyses, are also presented in Attachment E, Table 7.1-2, for comparison. Only the random portion of each plant operating parameter uncertainty is included in the statistical combination for RTDP. Any adverse instrumentation bias is treated either as a direct DNBR penalty or direct analysis input.

The RTDP design limit DNBR values for Byron Station and Braidwood Station were revised for uprated power operations from 1.25 to 1.24 for a thimble cell. The DNBR value for a typical cell remained unchanged at 1.25.

Conclusions

In addition to the above considerations for uncertainties, additional DNBR margin was maintained by performing the safety analyses to DNBR limits higher than the design limit DNBR values. Sufficient DNBR margin was maintained in the safety analysis DNBR limits to offset the rod bow DNBR penalty. The net remaining DNBR margin, after consideration of this penalty, is available for operating and design flexibility. Attachment E, Table 7.1-3, lists the DNBR limits and the DNBR margin summary for the Byron Station and the Braidwood Station RTDP analyses that support the power uprate.

Increase in RCS Total Flow Rate from ≥ 371,400 gpm to ≥ 380,900 gpm

The power uprate analyses assumed a total RCS flow rate value of 368,000 gpm in the evaluation of all normal and accident transients. 368,000 gpm is derived by assuming a thermal design flow value of 92,000 gpm/loop times four loops. The proposed TS LCO value of 380,900 gpm bounds the analysis value of 368,000 gpm plus a 3.5% flow measurement uncertainty. The RCS flow assumed in the power uprate analysis is a conservatively low flow that accounts for flow measurement uncertainty and assumes the maximum SGTP level. This RCS flow, together with the other PCWG parameters, detailed in Attachment E, Chapter 2.0, "Nuclear Steam Supply System (NSSS) Parameters," were determined to provide the Byron Station and the Braidwood Station with sufficient operating flexibility over a range of operating conditions.

Conclusions

The analysis has shown that the acceptance criteria, for all normal, abnormal, and accident transients, have been met assuming a total RCS flow rate value of 368,000 gpm which is bounded by the revised TS LCO value of 380,900 gpm.

Change in the SG Laser Welded Sleeve Plugging Limit

The proposed change revises the plugging limit for laser welded sleeves from 40% to 38.7% of nominal wall thickness. The analysis performed in support of the power uprate effort, indicated that it is necessary to remove SG tubes with laser welded sleeves from service upon discovering an imperfection depth of 38.7% wall thickness to ensure the structural integrity of SG tubes which have been sleeved thereby precluding the occurrence of a steam generator tube rupture of sleeved tubes under all operating conditions. The previous laser welded sleeve plugging limit was based on an analysis that used lower tolerance limit material strength values. The new analysis methodology, required for laser welded sleeves, uses minimum strength properties from the American Society of Mechanical Engineers Code. As determined by the new analysis, reducing the plugging limit from 40% to 38.7% maintains a comparable margin of safety to the previous analysis.

Reduction in Peak Calculated Containment Internal Pressure (P_a)

The current value for P_a for Unit 1 at Byron Station and Braidwood Station is 47.8 psig, while the current value of P_a for Unit 2 at Byron Station and Braidwood Station is 44.4 psig. The reanalysis of the containment LOCA mass and energy response has shown that P_a for both Unit 1 and Unit 2 at Byron Station and Braidwood Station will decrease to 42.8 psig and 38.4 psig, respectively. This reduction primarily comes from improved analytical techniques and computer codes described in Attachment E, Section 6.4, "LOCA Containment Integrity."

The most limiting accident scenario regarding containment pressure limitations is the Double Ended Hot Leg Break (DEHL) with minimum safeguards event. The analysis of this event assumes a loss of offsite power coincident with a double ended rupture of the RCS piping between the reactor pressure vessel outlet nozzle and the steam generator inlet (i.e., a break in the RCS hot leg). The associated single failure assumption is the failure of an emergency diesel generator (EDG) to start, resulting in only one train of ECCS and containment safeguards equipment being available. This combination results in a minimum set of safeguards being available. Further, loss of offsite power delays the actuation times of the safeguards equipment due to the EDG startup time after receipt of the safety injection signal.

Transient Description With BWI Steam Generator (Unit 1)

The postulated RCS break results in a rapid release of mass and energy to the containment with a resulting rapid rise in both the containment pressure and temperature. This rapid rise in containment pressure results in the generation of a containment HI-1 signal at 1.164 seconds and a containment HI-3 signal at 7.068 seconds. The containment pressure continues to rise rapidly in response to the release of mass and energy until the end of blowdown at 25.6 seconds, with the pressure reaching a value of 42.8 psig at 22.116 seconds. The end of blowdown marks a time when the initial inventory in the RCS has been exhausted and a process of filling the RCS downcomer in preparation for reflood has begun.

Transient Description With Westinghouse D5 Steam Generator (Unit 2)

The containment transient from a DEHL with minimum ECCS, using mass and energy releases developed for the Westinghouse D5 steam generator, follows a similar sequence of events as calculated with the BWI steam generator. The peak containment pressure of 38.4 psig occurs at 21.079 seconds.

Conclusions

The LOCA containment response analyses have been performed as part of the Byron Station, Units 1 and 2, and the Braidwood Station, Units 1 and 2, uprate program. The analyses included long term pressure and temperature profiles, including analyses with mass and energy releases for the BWI replacement SGs and the Westinghouse D5 SGs. The analysis resulted in a peak containment pressure that was less than the containment design pressure of 50 psig. The long term containment pressures are well below 50% of the peak value within 24 hours. Based on these results, the applicable containment accident acceptance criteria for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, have been met.

G. IMPACT ON PREVIOUS SUBMITTALS

All license amendment requests for the Byron and Braidwood Stations, currently under review by the NRC, were evaluated to determine if this submittal would impact them. No license amendment requests currently under review are impacted by the information presented in this license amendment request.

H. SCHEDULE REQUIREMENTS

We request that the NRC review and, if found acceptable, approve the proposed license and TS changes by May 7, 2001, to support a mid-cycle power uprate of Byron Station, Unit 1 and Unit 2, and Braidwood Station, Unit 2, prior to the summer months of 2001. Braidwood Station Unit 1 will not implement the power uprate changes until after the fall 2001 outage as noted below. Modifications to the high pressure (HP) turbine and other BOP components will be necessary to fully support uprated power conditions. Operations at the current power level with these modifications installed, prior to implementing the power uprate changes, was evaluated and found to be acceptable. Note that the power increase for Unit 2 at both Byron Station and Braidwood Station will be less than the power increase for Unit 1 at both stations. This is because Unit 2 at each station has the original Westinghouse Model D5 SGs whereas Unit 1 at each station has new BWI SGs which are capable of a larger power increase.

Byron Station Unit 1 will perform the necessary HP turbine and BOP modifications to support uprated power conditions during the fall 2000 refueling outage. Power uprate will be implemented during mid-cycle operations upon receipt of the license amendment. It is anticipated that operating Byron Station Unit 1 at uprated power conditions will allow the additional generation of approximately 70 Megawatts-electric (MWe) of power for the summer of 2001.

Byron Station Unit 2 will perform the necessary HP turbine and BOP modifications, to support uprated power conditions, during the spring 2001 refueling outage. Power uprate will be implemented during mid-cycle operations upon receipt of the license amendment. It is anticipated that operating Byron Station Unit 2 at uprated power conditions will allow the additional generation of approximately 40 MWe of power.

Braidwood Station Unit 1 will perform the necessary HP turbine and BOP modifications to support uprated power conditions during the fall 2001 refueling outage and, assuming receipt of the license amendment, will operate the unit at the uprated power conditions upon returning Unit 1 to service following the refueling outage. It is anticipated that operating Braidwood Station Unit 1 at uprated power conditions will allow the additional generation of approximately 70 MWe of power.

Braidwood Station Unit 2 will implement the power uprate changes during mid-cycle operations upon receipt of the license amendment. Although the HP turbine and BOP modifications to support full uprated power conditions will not be performed until the spring 2002 refueling outage, we anticipate that Unit 2 will be able to immediately increase electrical power output by approximately 10 MWe. Braidwood Station will operate Unit 2 at full uprated power conditions, assuming receipt of the license amendment upon returning Unit 2 to service following the spring 2002 refueling outage. It is anticipated that operating Braidwood Station Unit 2 at the full uprated power conditions will allow the generation of approximately 40 MWe of power above the current electrical output.

ATTACHMENT B-1

MARKED-UP PAGES FOR PROPOSED CHANGES

BYRON STATION

<u>REVISED LICENSE PAGES</u>

Unit 1 Page 3
Unit 2 Page 3

<u>REVISED TS PAGES</u>

1.1-3
1.1-6
2.0-1
3.4.1-1
3.4.1-2
5.5-12
5.5-24

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) 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

3586.6

The licensee is authorized to operate the facility at reactor core power levels not in excess of ~~3411~~ megawatts thermal (100% power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. ~~113~~ 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) Deleted.

(4) Deleted.

(5) Deleted.

(6) The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the licensee's Fire Protection Report, and as approved in the SER dated February 1987 through Supplement No. 8, subject to the following provision:

The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to posses, 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) <u>Maximum Power Level</u>

3586.6

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.

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A (NUREG-1113), as revised through Amendment No. 113 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. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Deleted.

(4) Deleted.

(5) Deleted.

1.1 Definitions

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites,"

or those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977, or ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."

$\bar{E}$ – AVERAGE DISINTEGRATION ENERGY

$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies (in MeV) per disintegration for non-iodine isotopes, with half lives > 10 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME

The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

1.1 Definitions

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits including heatup and cooldown rates, and the pressurizer Power Operated Relief Valve (PORV) lift settings for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Unit operation within these limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~3411~~ 3586.6 MWt.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the component and methodology for verification have been previously reviewed and approved by the NRC.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded.

1.24

2.1.1.1 In MODE 1, the Departure from Nucleate Boiling Ratio (DNBR) shall be maintained ≥ ~~1.25~~ for the WRB-2 DNB correlation for a thimble cell and ≥ 1.25 for the WRB-2 DNB Correlation for a typical cell.

2.1.1.2 In MODE 2, the DNBR shall be maintained ≥ 1.17 for the WRB-2 DNB correlation, and ≥ 1.30 for the W-3 DNB correlation.

2.1.1.3 In MODES 1 and 2, the peak fuel centerline temperature shall be maintained ≤ 4700°F.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature (T_{avg}), and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure within the limit specified in the COLR;

b. RCS average temperature (T_{avg}) within the limit specified in the COLR; and

c. RCS total flow rate ≥ ~~371,400~~ 380,900 gpm and within the limit specified in the COLR.

---NOTE---

Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is within the limit specified in the COLR.	12 hours
SR 3.4.1.2	Verify RCS average temperature (T_{avg}) is within the limit specified in the COLR.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is ≥ 380,900 ~~371,400~~ gpm and within the limit specified in the COLR.	12 hours
SR 3.4.1.4	-------------------NOTE------------------- Not required to be performed until 7 days after ≥ 90% RTP. -- Verify by precision heat balance that RCS total flow rate is ≥ ~~371,400~~ 380,900 gpm and within the limit specified in the COLR.	18 months

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

e. Acceptance Criteria

1. Imperfection means an exception to the dimensions, finish or contour of a tube or sleeve from that required by fabrication drawings or specifications. Eddy current testing indications < 20% of the nominal tube or sleeve wall thickness, if detectable, may be considered as imperfections;

2. Degradation means a service induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube or sleeve;

3. Degraded Tube means a tube or sleeve containing unrepaired imperfections ≥ 20% of the nominal tube or sleeve wall thickness caused by degradation;

4. % Degradation means the percentage of the tube or sleeve wall thickness affected or removed by degradation;

5. Defect means an imperfection of such severity that it exceeds the plugging or repair limit. A tube or sleeve containing an unrepaired defect is defective;

6. Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving in the affected area. The plugging or repair limit imperfection depth for the tubing ~~and laser welded sleeves~~ is equal to 40% of the nominal wall thickness. The plugging limit imperfection depth for TIG welded sleeves is equal to 32% of the nominal wall thickness;

The plugging limit imperfection depth for laser welded sleeves is equal to 38.7% of the nominal wall thickness.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an OBE, LOCA, or a steam line or feedwater line break as specified in Specification 5.5.9.d.4;

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, September 1995 and NEI 94-01, Revision 0.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is ~~47.8~~ 42.8 psig for Unit 1 and ~~44.4~~ 38.4 psig for Unit 2.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.10% of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C tests and $< 0.75\ L_a$ for Type A tests; and

ATTACHMENT B-2

MARKED-UP PAGES FOR PROPOSED CHANGES

BRAIDWOOD STATION

REVISED LICENSE PAGES

Unit 1 Page 3
Unit 2 Page 3

REVISED TS PAGES

1.1-3
1.1-6
2.0-1
3.4.1-1
3.4.1-2
5.5-12
5.5-24

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~~ 3586.6 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 identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. ~~107~~ 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 in 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.

6) Deleted.

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

3586.6

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 and 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

The Technical Specifications contained in Appendix A as revised through Amendment No. ~~107~~ 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 in 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) Deleted.

6) Additional Conditions

The Additional Conditions contained in Appendix C, as revised through Amendment No. 98, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Additional Conditions.

1.1 Definitions

DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites,"
$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies (in MeV) per disintegration for non-iodine isotopes, with half lives > 10 minutes, making up at least 95% of the total non-iodine activity in the coolant.
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

or those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977, or ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."

1.1 Definitions

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits including heatup and cooldown rates, and the pressurizer Power Operated Relief Valve (PORV) lift settings for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Unit operation within these limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~3411~~ 3586.6 MWt.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded.

2.1.1.1 In MODE 1, the Departure from Nucleate Boiling Ratio (DNBR) shall be maintained ≥ ~~1.25~~ 1.24 for the WRB-2 DNB correlation for a thimble cell and ≥ 1.25 for the WRB-2 DNB Correlation for a typical cell.

2.1.1.2 In MODE 2, the DNBR shall be maintained ≥ 1.17 for the WRB-2 DNB correlation, and ≥ 1.30 for the W-3 DNB correlation.

2.1.1.3 In MODES 1 and 2, the peak fuel centerline temperature shall be maintained ≤ 4700°F.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature (T_{avg}), and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure within the limit specified in the COLR;

b. RCS average temperature (T_{avg}) within the limit specified in the COLR; and

c. RCS total flow rate ≥ ~~371,400~~ 380,900 gpm and within the limit specified in the COLR.

---NOTE---

Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is within the limit specified in the COLR.	12 hours
SR 3.4.1.2	Verify RCS average temperature (T_{avg}) is within the limit specified in the COLR.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is ≥ ~~371,400~~ 380,900 gpm and within the limit specified in the COLR.	12 hours
SR 3.4.1.4	-------------------NOTE------------------- Not required to be performed until 7 days after ≥ 90% RTP. -- Verify by precision heat balance that RCS total flow rate is ≥ ~~371,400~~ 380,900 gpm and within the limit specified in the COLR.	18 months

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

e. Acceptance Criteria

1. Imperfection means an exception to the dimensions, finish or contour of a tube or sleeve from that required by fabrication drawings or specifications. Eddy current testing indications < 20% of the nominal tube or sleeve wall thickness, if detectable, may be considered as imperfections;

2. Degradation means a service induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube or sleeve;

3. Degraded Tube means a tube or sleeve containing unrepaired imperfections ≥ 20% of the nominal tube or sleeve wall thickness caused by degradation;

4. % Degradation means the percentage of the tube or sleeve wall thickness affected or removed by degradation;

5. Defect means an imperfection of such severity that it exceeds the plugging or repair limit. A tube or sleeve containing an unrepaired defect is defective;

6. Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving in the affected area. The plugging or repair limit imperfection depth for the tubing ~~and laser welded sleeves~~ is equal to 40% of the nominal wall thickness. The plugging limit imperfection depth for TIG welded sleeves is equal to 32% of the nominal wall thickness;

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an OBE, LOCA, or a steam line or feedwater line break as specified in Specification 5.5.9.d.4;

The plugging limit imperfection depth for laser welded sleeves is equal to 38.7% of the nominal wall thickness.

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, September 1995 and NEI 94-01, Revision 0.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is ~~47.8~~ 42.8 psig for Unit 1 and ~~44.4~~ 38.4 psig for Unit 2.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.10% of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C tests and $< 0.75\ L_a$ for Type A tests; and

ATTACHMENT B-3

INCORPORATED PROPOSED CHANGES
TYPED PAGES

BYRON STATION

REVISED TS PAGES

1.1-3
1.1-6
2.0-1
3.4.1-1
3.4.1-2
5.5-12
5.5-24

REVISED BASES PAGES

B 3.3.1-22
B 3.4.1-2
B 3.4.1-3
B 3.4.1-5
B 3.4.13-2
B 3.4.13-3
B 3.4.13-4
B 3.6.1-3
B 3.6.2-2
B 3.6.4-2
B 3.6.6-5
B 3.6.6-6
B 3.6.8-1
B 3.6.8-2
B 3.7.3-1
B 3.7.5-3

1.1 Definitions

DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977, or ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."
$\bar{E}$ – AVERAGE DISINTEGRATION ENERGY	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies (in MeV) per disintegration for non-iodine isotopes, with half lives > 10 minutes, making up at least 95% of the total non-iodine activity in the coolant.
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

1.1 Definitions

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits including heatup and cooldown rates, and the pressurizer Power Operated Relief Valve (PORV) lift settings for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Unit operation within these limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3586.6 MWt.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded.

2.1.1.1 In MODE 1, the Departure from Nucleate Boiling Ratio (DNBR) shall be maintained ≥ 1.24 for the WRB-2 DNB correlation for a thimble cell and ≥ 1.25 for the WRB-2 DNB correlation for a typical cell.

2.1.1.2 In MODE 2, the DNBR shall be maintained ≥ 1.17 for the WRB-2 DNB correlation, and ≥ 1.30 for the W-3 DNB correlation.

2.1.1.3 In MODES 1 and 2, the peak fuel centerline temperature shall be maintained ≤ 4700°F.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature (T_{avg}), and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure within the limit specified in the COLR;

b. RCS average temperature (T_{avg}) within the limit specified in the COLR; and

c. RCS total flow rate ≥ 380,900 gpm and within the limit specified in the COLR.

---------------------------NOTE---------------------------

Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is within the limit specified in the COLR.	12 hours
SR 3.4.1.2	Verify RCS average temperature (T_{avg}) is within the limit specified in the COLR.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is ≥ 380,900 gpm and within the limit specified in the COLR.	12 hours
SR 3.4.1.4	-------------------NOTE------------------- Not required to be performed until 7 days after ≥ 90% RTP. -- Verify by precision heat balance that RCS total flow rate is ≥ 380,900 gpm and within the limit specified in the COLR.	18 months

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

e. Acceptance Criteria

1. Imperfection means an exception to the dimensions, finish or contour of a tube or sleeve from that required by fabrication drawings or specifications. Eddy current testing indications < 20% of the nominal tube or sleeve wall thickness, if detectable, may be considered as imperfections;

2. Degradation means a service induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube or sleeve;

3. Degraded Tube means a tube or sleeve containing unrepaired imperfections ≥ 20% of the nominal tube or sleeve wall thickness caused by degradation;

4. % Degradation means the percentage of the tube or sleeve wall thickness affected or removed by degradation;

5. Defect means an imperfection of such severity that it exceeds the plugging or repair limit. A tube or sleeve containing an unrepaired defect is defective;

6. Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving in the affected area. The plugging or repair limit imperfection depth for the tubing is equal to 40% of the nominal wall thickness. The plugging limit imperfection depth for laser welded sleeves is equal to 38.7% of the nominal wall thickness. The plugging limit imperfection depth for TIG welded sleeves is equal to 32% of the nominal wall thickness;

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an OBE, LOCA, or a steam line or feedwater line break as specified in Specification 5.5.9.d.4;

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program. the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, September 1995 and NEI 94-01, Revision 0.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 42.8 psig for Unit 1 and 38.4 psig for Unit 2.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.10% of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C tests and $< 0.75\ L_a$ for Type A tests; and

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

In MODE 1, when there is a potential for overfilling the pressurizer, the Pressurizer Water Level – High trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 interlock. On decreasing power, this trip Function is automatically blocked below P-7. Below the P-7 setpoint, transients that could raise the pressurizer water level will be slow and the operator will have sufficient time to evaluate unit conditions and take corrective actions.

10. Reactor Coolant Flow – Low

The Reactor Coolant Flow-Low Function ensures that protection is provided against violating the DNBR limit due to low flow in the RCS loops, while avoiding reactor trips due to normal variations in loop flow. Each RCS loop has three flow detectors to monitor flow. The flow signals are not used for any control system input.

The LCO requires three Reactor Coolant Flow-Low channels per loop to be OPERABLE in MODE 1 above P-7. Each loop is considered a separate Function. The channel Allowable Values are specified in percent of loop minimum measured flow. The minimum measured flow is 95,225 gpm.

The Reactor Coolant Flow-Low Function encompasses a single loop and a two loop trip logic. In MODE 1 above the P-7 setpoint and below the P-8 setpoint, a loss of flow in two or more loops will initiate a reactor trip. Above the P-8 setpoint, which is approximately 30% RTP, a loss of flow in any one RCS loop will actuate a reactor trip because of the higher power level and the reduced margin to the design limit DNBR. Below the P-7 setpoint, all reactor trips on low flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level.

APPLICABLE SAFETY ANALYSES

The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will result in meeting the DNBR criterion. This is the acceptance limit for the RCS DNB parameters. Changes to the unit that could impact these parameters must be assessed for their impact on the DNB criteria. The transients analyzed for include loss of coolant flow events and dropped or stuck rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.6, "Control Bank Insertion Limits;" LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD);" and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

Safety Analyses assumed a value of 2207 psia (2192.3 psig). This value is bounded by the limit specified in the COLR assuming a measurement accuracy of less than 26.7 psi.

Safety Analyses assumed a value of 588.0°F for the vessel average temperature.

Safety Analyses assumed a total RCS flow rate of 368,000 gpm. This value is bounded by the LCO value of 380,900 gpm and the limit specified in the COLR assuming a flow measurement uncertainty of 3.5%. This 3.5% flow measurement uncertainty assumed in the analyses included errors from known sources.

The RCS DNB parameters satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO specifies limits on the monitored process variables—pressurizer pressure, RCS average temperature (T_{avg}), and RCS total flow rate—to ensure the core operates within the limits assumed in the safety analyses. These variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. However, the minimum RCS flow, based on maximum analyzed steam generator tube plugging, is retained in the LCO to assure that a lower flow rate than reviewed by the NRC will not be used. Operating within these limits will result in meeting the DNB design criterion in the event of a DNB limited transient.

A Note has been added to indicate the limit on pressurizer is not applicable during short term operational transients such as a THERMAL POWER ramp increase > 5% RTP per minute or a THERMAL POWER step increase > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, since they typically represent transients initiated from power levels < 100% RTP, an increased Departure from Nucleate Boiling Ratio (DNBR) margin exists to offset the temporary pressure variations.

The DNBR limit is provided in SL 2.1.1, "Reactor Core SLs." LCO 3.4.1 represents the initial conditions of the safety analysis which are far more restrictive than the conditions which define the DNBR limit. Should a violation of this LCO occur, the operator must check whether or not an SL may have been exceeded.

BASES

FOR INFORMATION ONLY

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for pressurizer pressure is sufficient to ensure the pressure can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.2

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for RCS average temperature (T_{avg}) is sufficient to ensure the temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.3

The 12 hour Surveillance Frequency for RCS total flow rate is performed using the installed flow instrumentation. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation within safety analysis assumptions.

FOR INFORMATION ONLY

BASES

BACKGROUND (continued)

This LCO deals with protection of the Reactor Coolant Pressure Boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a Loss Of Coolant Accident (LOCA). However, the ability to monitor leakage provides advance warning to permit unit shutdown before a LOCA occurs. This advantage has been shown by "leak before break" studies.

APPLICABLE SAFETY ANALYSIS

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes 1 gpm primary to secondary LEAKAGE as the initial condition.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a Locked Rotor with a Concurrent Steam Generator (SG) Power Operated Relief Valve (PORV) Failure accident because such leakage contaminates the secondary fluid. Other accidents or transients involve secondary steam release to the atmosphere, such as a Steam Generator Tube Rupture (SGTR). The SGTR is more limiting than the Locked Rotor with a Concurrent SG PORV Failure for site radiation releases.

The UFSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is released for a limited time via the SG PORV. After a tube rupture occurs, reactor coolant immediately begins flowing from the primary system into the secondary side of the ruptured SG causing the RCS pressure to decrease until a reactor trip occurs on low pressurizer pressure. The analysis assumes a Loss of Offsite Power occurs coincident with the reactor trip causing the Reactor Coolant Pumps to trip and the main condenser to become unavailable when the circulating water pumps are lost.

BASES

APPLICABLE SAFETY ANALYSES (continued)

After the reactor trips, the core power quickly decreases to decay heat levels. The steam dump system cannot be used to dissipate the core decay heat due to the unavailable condenser. Therefore, the secondary pressure increases in the SGs until the SG PORVs open at which time the ruptured SG PORV is assumed to fail in the open position. The ruptured SG failed PORV is isolated when the block valve is manually closed twenty minutes after the PORV first opened. The 1 gpm primary to secondary LEAKAGE is relatively inconsequential to the results of this analysis.

The dose consequences resulting from the Locked Rotor with a Concurrent SG PORV Failure accident are well within the limits defined in 10 CFR 100.

To support the use of sleeving techniques for SG tube repair, the Unit 1 primary to secondary leakage limits are conservatively reduced from 500 gpd for any single SG and 1 gpm total to 150 gpd for any single SG and 600 gpd total (Ref. 4).

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals, valve seats, and gaskets is not pressure boundary LEAKAGE.

LCO (continued)

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump discharge flow monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled Reactor Coolant Pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through All SGS

Total primary to secondary LEAKAGE amounting to 600 gallons per day through all SGs not isolated from the RCS produces acceptable offsite doses in the Locked Rotor with a Concurrent SG PORV Failure accident analysis. Violation of this LCO could exceed the offsite dose limits for this accident. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.

APPLICABLE SAFETY ANALYSES (continued)

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a LOCA and a steam line break (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA, secondary system pipe break, or fuel handling accident (Ref. 3). In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.10% of containment air weight per day (Ref. 3). This leakage rate, used to evaluate offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting design basis LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.10% per day in the safety analysis at P_a = 42.8 psig for Unit 1 and P_a = 38.4 psig for Unit 2 (Ref. 3).

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Containment OPERABILITY is maintained by limiting leakage to ≤ 1.0 L_a, except prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test. At this time, applicable leakage limits must be met.

Compliance with this LCO will ensure a containment configuration, including the equipment hatch, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within containment are a Loss Of Coolant Accident (LOCA), secondary system pipe break, and a fuel handling accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.1% of containment air weight per day (Ref. 2). This leakage rate is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as the maximum allowable containment leakage rate at the calculated peak containment internal pressure, at P_a = 42.8 psig for Unit 1 and P_a = 38.4 psig for Unit 2 following a DBA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

The containment air locks satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Each containment air lock forms part of the containment pressure boundary. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into or exit from containment.

APPLICABLE SAFETY ANALYSES (continued)

The initial pressure condition used in the containment analysis was 1.0 psig. This resulted in a maximum peak pressure from a LOCA of 42.8 psig for Unit 1 and 38.4 psig for Unit 2. The containment analysis (Ref. 1) shows that the maximum peak calculated containment pressure, P_a, results from the limiting LOCA. The maximum containment pressure resulting from the worst case LOCA does not exceed the containment design pressure, 50 psig.

The containment was also evaluated for an external pressure load equivalent to -3.5 psig (Ref. 2). The inadvertent actuation of the Containment Spray System was analyzed to determine the resulting reduction in containment pressure. The initial pressure condition used in this analysis was 0.0 psig. This resulted in a minimum pressure inside containment of -3.48 psig, which is less than the design load.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. Therefore, for the reflood phase, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with 10 CFR 50, Appendix K (Ref. 3).

Containment pressure satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

APPLICABLE SAFETY ANALYSES (continued)

The analysis and evaluation show that under the worst case scenario, the highest peak containment pressure is 42.8 psig for Unit 1 and 38.4 psig for Unit 2 (experienced during a LOCA). The analysis shows that the peak containment temperature is 333°F for Unit 1 and 331°F for Unit 2 (experienced during an SLB). Both results meet the intent of the design basis. (See the Bases for LCO 3.6.4, "Containment Pressure," and LCO 3.6.5 for a detailed discussion.) The analyses and evaluations assume a unit specific power level of 3672.6 MWt, one containment spray train and one containment cooling train operating, and initial (pre-accident) containment conditions of 120°F and 1.0 psig. The analyses also assume a response time delayed initiation to provide conservative peak calculated containment pressure and temperature responses.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 4).

The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation results in a −3.48 psig containment pressure and is associated with the sudden cooling effect in the interior of the leak tight containment. Additional discussion is provided in the Bases for LCO 3.6.4.

The modeled Containment Spray System actuation from the containment analysis is based on a response time associated with exceeding the containment High-3 pressure setpoint to achieving full flow through the containment spray nozzles. The Containment Spray System total response time of 88.1 seconds includes Diesel Generator (DG) startup (for loss of offsite power), sequencing of equipment, containment spray pump startup, and spray line filling (Ref. 5).

APPLICABLE SAFETY ANALYSES (continued)

Containment cooling train performance for post accident conditions is given in Reference 6. The result of the analysis is that each train can provide 100% of the required peak cooling capacity during the post accident condition. The train post accident cooling capacity under varying containment ambient conditions, required to perform the accident analyses, is also shown in Reference 7.

The modeled Containment Cooling System actuation from the containment analysis is based upon a response time associated with exceeding the containment High-3 pressure setpoint to achieving full Containment Cooling System air and safety grade cooling water flow. The Containment Cooling System total response time of 65 seconds, includes signal delay, DG startup (for loss of offsite power), and service water pump startup times (Ref. 5).

The Containment Spray System and the Containment Cooling System satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

During a DBA, a minimum of one containment cooling train and one containment spray train are required to maintain the containment peak pressure and temperature below the design limits (Ref. 7). Additionally, one containment spray train is also required to remove iodine from the containment atmosphere and maintain concentrations below those assumed in the safety analysis. To ensure that these requirements are met, two containment spray trains and two containment cooling trains must be OPERABLE. The chemical aspects of iodine removal capability are addressed in LCO 3.6.7. Therefore, in the event of an accident, at least one train in each system operates, assuming the worst case single active failure occurs.

Each Containment Spray System includes a spray pump, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an ESF actuation signal and manually transferring suction to the containment sump.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.8 Hydrogen Recombiners

FOR INFORMATION ONLY

BASES

BACKGROUND

The function of the hydrogen recombiners is to eliminate the potential breach of containment due to a hydrogen oxygen reaction.

Per 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Reactors" (Ref. 1), and GDC 41, "Containment Atmosphere Cleanup" (Ref. 2), hydrogen recombiners are required to reduce the hydrogen concentration in the containment following a Loss Of Coolant Accident (LOCA) or Steam Line Break (SLB). The recombiners accomplish this by recombining hydrogen and oxygen to form water vapor. The vapor remains in containment, thus eliminating any discharge to the environment. The hydrogen recombiners are manually initiated since flammable limits would not be reached until several days after a Design Basis Accident (DBA).

Two 100% capacity independent hydrogen recombiner systems are provided and shared between the units. Each consists of controls located in the auxiliary building, a power supply and a recombiner. Recombination is accomplished by heating a hydrogen air mixture to 1325°F. The resulting water vapor and discharge gases are cooled prior to discharge from the recombiner. A single recombiner is capable of maintaining the hydrogen concentration in containment below the 4.0 volume percent (v/o) flammability limit. Two recombiners are provided to meet the requirement for redundancy and independence. Each recombiner is powered from a separate Engineered Safety Features bus, and is provided with a separate power panel and control panel.

The hydrogen recombiners are described in UFSAR, Section 6.2.5 (Ref. 3).

APPLICABLE SAFETY ANALYSES

The hydrogen recombiners provide for the capability of controlling the bulk hydrogen concentration in containment to less than the lower flammable concentration of 4.0 v/o following a DBA. This control would prevent a containment wide hydrogen burn, thus ensuring the pressure and temperature assumed in the analyses are not exceeded. The limiting DBA relative to hydrogen generation is a LOCA. Hydrogen may accumulate in containment following a LOCA as a result of:

a. A metal steam reaction between the zirconium fuel rod cladding and the reactor coolant;

b. Radiolytic decomposition of water in the Reactor Coolant System (RCS) and the containment sump;

c. Hydrogen in the RCS at the time of the LOCA (i.e., hydrogen dissolved in the reactor coolant and hydrogen gas in the pressurizer vapor space); or

d. Corrosion of metals exposed to containment spray and Emergency Core Cooling System solutions.

To evaluate the potential for hydrogen accumulation in containment following a LOCA, the hydrogen generation as a function of time following the initiation of the accident is calculated. Conservative assumptions recommended by Reference 4 are used to maximize the amount of hydrogen calculated.

Based on the conservative assumptions used to calculate the hydrogen concentration versus time after a LOCA, the hydrogen concentration in the primary containment would reach 2.6 v/o about 20 hours after the LOCA and 4.0 v/o about 4 days later if no recombiner was functioning (Ref. 3). Initiating a hydrogen recombiner when the primary containment hydrogen concentration reaches 2.6 v/o will maintain the hydrogen concentration in the primary containment below flammability limits.

B 3.7 PLANT SYSTEMS

B 3.7.3 Secondary Specific Activity

FOR INFORMATION ONLY

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.13, "RCS Operational LEAKAGE") of primary coolant at the limit of 1.0 μCi/gm (LCO 3.4.16, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives, (i.e., < 20 hours). I-131, with a half life of 8.04 days, concentrates faster than it decays, but does not reach equilibrium because of blowdown and other losses.

With the specified activity limit, the resultant 2 hour thyroid dose to a person at the exclusion area boundary (EAB) would be about 6.5 Rem if the Main Steam Safety Valves (MSSVs) open for 2 hours following a trip from full power.

Operating a unit at the allowable limits could result in a 2 hour EAB exposure of a small fraction of the 10 CFR 100 (Ref. 1) limits.

APPLICABLE SAFETY ANALYSES (continued)

In addition, the minimum available AF flow and system characteristics are serious considerations in the analysis of a small break Loss Of Coolant Accident (LOCA) and loss of offsite power (Ref. 3).

The AF System design is such that it can perform its function following an FWLB between the main feedwater isolation valves and containment, combined with a loss of offsite power following turbine trip, and a single active failure of one AF pump. The AF lines to the SGs are orificed such that at least 328 gpm is delivered to the non faulted SGs. Reactor trip is assumed to occur when the faulted SG reaches the low-low level setpoint. Sufficient flow would be delivered to the intact steam generators by the other AF pump.

During the loss of all AC power events, the Engineered Safety Feature Actuation System (ESFAS) automatically actuates the AF diesel driven pump and associated controls to ensure an adequate supply to the steam generators during loss of power. Valves which can be manually controlled are provided for each AF line to control the AF flow to each steam generator during loss of all AC power events.

The AF System satisfies the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii).

ATTACHMENT B-4

INCORPORATED PROPOSED CHANGES
TYPED PAGES

BRAIDWOOD STATION

<u>REVISED TS PAGES</u>

1.1-3
1.1-6
2.0-1
3.4.1-1
3.4.1-2
5.5-12
5.5-24

<u>REVISED BASES PAGES</u>

B 3.3.1-22
B 3.4.1-2
B 3.4.1-3
B 3.4.1-5
B 3.4.13-2
B 3.4.13-3
B 3.4.13-4
B 3.6.1-3
B 3.6.2-2
B 3.6.4-2
B 3.6.6-5
B 3.6.6-6
B 3.6.8-1
B 3.6.8-2
B 3.7.3-1
B 3.7.5-3

1.1 Definitions

DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977, or ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."
$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies (in MeV) per disintegration for non-iodine isotopes, with half lives > 10 minutes, making up at least 95% of the total non-iodine activity in the coolant.
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

1.1 Definitions

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits including heatup and cooldown rates, and the pressurizer Power Operated Relief Valve (PORV) lift settings for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Unit operation within these limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3586.6 MWt.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded.

2.1.1.1 In MODE 1, the Departure from Nucleate Boiling Ratio (DNBR) shall be maintained ≥ 1.24 for the WRB-2 DNB correlation for a thimble cell and ≥ 1.25 for the WRB-2 DNB correlation for a typical cell.

2.1.1.2 In MODE 2, the DNBR shall be maintained ≥ 1.17 for the WRB-2 DNB correlation, and ≥ 1.30 for the W-3 DNB correlation.

2.1.1.3 In MODES 1 and 2, the peak fuel centerline temperature shall be maintained ≤ 4700°F.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature (T_{avg}), and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure within the limit specified in the COLR;

b. RCS average temperature (T_{avg}) within the limit specified in the COLR; and

c. RCS total flow rate ≥ 380,900 gpm and within the limit specified in the COLR.

---------------------------NOTE---------------------------
Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is within the limit specified in the COLR.	12 hours
SR 3.4.1.2	Verify RCS average temperature (T_{avg}) is within the limit specified in the COLR.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is ≥ 380,900 gpm and within the limit specified in the COLR.	12 hours
SR 3.4.1.4	-------------------NOTE-------------------- Not required to be performed until 7 days after ≥ 90% RTP. --- Verify by precision heat balance that RCS total flow rate is ≥ 380,900 gpm and within the limit specified in the COLR.	18 months

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

e. Acceptance Criteria

1. Imperfection means an exception to the dimensions, finish or contour of a tube or sleeve from that required by fabrication drawings or specifications. Eddy current testing indications < 20% of the nominal tube or sleeve wall thickness, if detectable, may be considered as imperfections;

2. Degradation means a service induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube or sleeve;

3. Degraded Tube means a tube or sleeve containing unrepaired imperfections ≥ 20% of the nominal tube or sleeve wall thickness caused by degradation;

4. % Degradation means the percentage of the tube or sleeve wall thickness affected or removed by degradation;

5. Defect means an imperfection of such severity that it exceeds the plugging or repair limit. A tube or sleeve containing an unrepaired defect is defective;

6. Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving in the affected area. The plugging or repair limit imperfection depth for the tubing is equal to 40% of the nominal wall thickness. The plugging limit imperfection depth for laser welded sleeves is equal to 38.7% of the nominal wall thickness. The plugging limit imperfection depth for TIG welded sleeves is equal to 32% of the nominal wall thickness;

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an OBE, LOCA, or a steam line or feedwater line break as specified in Specification 5.5.9.d.4;

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, September 1995 and NEI 94-01, Revision 0.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 42.8 psig for Unit 1 and 38.4 psig for Unit 2.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.10% of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C tests and $< 0.75\ L_a$ for Type A tests; and

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

In MODE 1, when there is a potential for overfilling the pressurizer, the Pressurizer Water Level - High trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 interlock. On decreasing power, this trip Function is automatically blocked below P-7. Below the P-7 setpoint, transients that could raise the pressurizer water level will be slow and the operator will have sufficient time to evaluate unit conditions and take corrective actions.

10. Reactor Coolant Flow - Low

The Reactor Coolant Flow-Low Function ensures that protection is provided against violating the DNBR limit due to low flow in the RCS loops, while avoiding reactor trips due to normal variations in loop flow. Each RCS loop has three flow detectors to monitor flow. The flow signals are not used for any control system input.

The LCO requires three Reactor Coolant Flow-Low channels per loop to be OPERABLE in MODE 1 above P-7. Each loop is considered a separate Function. The channel Allowable Values are specified in percent of loop minimum measured flow. The minimum measured flow is 95,225 gpm.

The Reactor Coolant Flow-Low Function encompasses a single loop and a two loop trip logic. In MODE 1 above the P-7 setpoint and below the P-8 setpoint, a loss of flow in two or more loops will initiate a reactor trip. Above the P-8 setpoint, which is approximately 30% RTP, a loss of flow in any one RCS loop will actuate a reactor trip because of the higher power level and the reduced margin to the design limit DNBR. Below the P-7 setpoint, all reactor trips on low flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level.

BASES

APPLICABLE SAFETY ANALYSES

The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will result in meeting the DNBR criterion. This is the acceptance limit for the RCS DNB parameters. Changes to the unit that could impact these parameters must be assessed for their impact on the DNB criteria. The transients analyzed for include loss of coolant flow events and dropped or stuck rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.6, "Control Bank Insertion Limits;" LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD);" and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

Safety Analyses assumed a value of 2207 psia (2192.3 psig). This value is bounded by the limit specified in the COLR assuming a measurement accuracy of less than 26.7 psi.

Safety Analyses assumed a value of 588.0°F for the vessel average temperature.

Safety Analyses assumed a total RCS flow rate of 368,000 gpm. This value is bounded by the LCO value of 380,900 gpm and the limit specified in the COLR assuming a flow measurement uncertainty of 3.5%. This 3.5% flow measurement uncertainty assumed in the analyses included errors from known sources.

The RCS DNB parameters satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO specifies limits on the monitored process variables – pressurizer pressure, RCS average temperature (T_{avg}), and RCS total flow rate – to ensure the core operates within the limits assumed in the safety analyses. These variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. However, the minimum RCS flow, based on maximum analyzed steam generator tube plugging, is retained in the LCO to assure that a lower flow rate than reviewed by the NRC will not be used. Operating within these limits will result in meeting the DNB design criterion in the event of a DNB limited transient.

A Note has been added to indicate the limit on pressurizer is not applicable during short term operational transients such as a THERMAL POWER ramp increase > 5% RTP per minute or a THERMAL POWER step increase > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, since they typically represent transients initiated from power levels < 100% RTP, an increased Departure from Nucleate Boiling Ratio (DNBR) margin exists to offset the temporary pressure variations.

The DNBR limit is provided in SL 2.1.1, "Reactor Core SLs." LCO 3.4.1 represents the initial conditions of the safety analysis which are far more restrictive than the conditions which define the DNBR limit. Should a violation of this LCO occur, the operator must check whether or not an SL may have been exceeded.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for pressurizer pressure is sufficient to ensure the pressure can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.2

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for RCS average temperature (T_{avg}) is sufficient to ensure the temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.3

The 12 hour Surveillance Frequency for RCS total flow rate is performed using the installed flow instrumentation. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation within safety analysis assumptions.

FOR INFORMATION ONLY

BASES

BACKGROUND (continued)

This LCO deals with protection of the Reactor Coolant Pressure Boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a Loss Of Coolant Accident (LOCA). However, the ability to monitor leakage provides advance warning to permit unit shutdown before a LOCA occurs. This advantage has been shown by "leak before break" studies.

APPLICABLE SAFETY ANALYSIS

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes 1 gpm primary to secondary LEAKAGE as the initial condition.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a Locked Rotor with a Concurrent Steam Generator (SG) Power Operated Relief Valve (PORV) Failure accident because such leakage contaminates the secondary fluid. Other accidents or transients involve secondary steam release to the atmosphere, such as a Steam Generator Tube Rupture (SGTR). The SGTR is more limiting than the Locked Rotor with a Concurrent SG PORV Failure for site radiation releases.

The UFSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is released for a limited time via the SG PORV. After a tube rupture occurs, reactor coolant immediately begins flowing from the primary system into the secondary side of the ruptured SG causing the RCS pressure to decrease until a reactor trip occurs on low pressurizer pressure. The analysis assumes a Loss of Offsite Power occurs coincident with the reactor trip causing the Reactor Coolant Pumps to trip and the main condenser to become unavailable when the circulating water pumps are lost.

APPLICABLE SAFETY ANALYSES (continued)

After the reactor trips, the core power quickly decreases to decay heat levels. The steam dump system cannot be used to dissipate the core decay heat due to the unavailable condenser. Therefore, the secondary pressure increases in the SGs until the SG PORVs open at which time the ruptured SG PORV is assumed to fail in the open position. The ruptured SG failed PORV is isolated when the block valve is manually closed twenty minutes after the PORV first opened. The 1 gpm primary to secondary LEAKAGE is relatively inconsequential to the results of this analysis.

The dose consequences resulting from the Locked Rotor with a Concurrent SG PORV Failure accident are well within the limits defined in 10 CFR 100.

To support the use of sleeving techniques for SG tube repair, the Unit 1 primary to secondary leakage limits are conservatively reduced from 500 gpd for any single SG and 1 gpm total to 150 gpd for any single SG and 600 gpd total (Ref. 4).

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals, valve seats, and gaskets is not pressure boundary LEAKAGE.

LCO (continued)

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump discharge flow monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled Reactor Coolant Pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through All SGs

Total primary to secondary LEAKAGE amounting to 600 gallons per day through all SGs not isolated from the RCS produces acceptable offsite doses in the Locked Rotor with a Concurrent SG PORV Failure accident analysis. Violation of this LCO could exceed the offsite dose limits for this accident. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.

APPLICABLE SAFETY ANALYSES (continued)

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a LOCA and a steam line break (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA, secondary system pipe break, or fuel handling accident (Ref. 3). In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.10% of containment air weight per day (Ref. 3). This leakage rate, used to evaluate offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting design basis LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.10% per day in the safety analysis at P_a = 42.8 psig for Unit 1 and P_a = 38.4 psig for Unit 2 (Ref. 3).

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Containment OPERABILITY is maintained by limiting leakage to ≤ 1.0 L_a, except prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test. At this time, applicable leakage limits must be met.

Compliance with this LCO will ensure a containment configuration, including the equipment hatch, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

FOR INFORMATION ONLY

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within containment are a Loss Of Coolant Accident (LOCA), secondary system pipe break, and a fuel handling accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.1% of containment air weight per day (Ref. 2). This leakage rate is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as the maximum allowable containment leakage rate at the calculated peak containment internal pressure, at P_a = 42.8 psig for Unit 1 and P_a = 38.4 psig for Unit 2 following a DBA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

The containment air locks satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Each containment air lock forms part of the containment pressure boundary. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into or exit from containment.

APPLICABLE SAFETY ANALYSES (continued)

The initial pressure condition used in the containment analysis was 1.0 psig. This resulted in a maximum peak pressure from a LOCA of 42.8 psig for Unit 1 and 38.4 psig for Unit 2. The containment analysis (Ref. 1) shows that the maximum peak calculated containment pressure, P_a, results from the limiting LOCA. The maximum containment pressure resulting from the worst case LOCA does not exceed the containment design pressure, 50 psig.

The containment was also evaluated for an external pressure load equivalent to -3.5 psig (Ref. 2). The inadvertent actuation of the Containment Spray System was analyzed to determine the resulting reduction in containment pressure. The initial pressure condition used in this analysis was 0.0 psig. This resulted in a minimum pressure inside containment of -3.48 psig, which is less than the design load.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. Therefore, for the reflood phase, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with 10 CFR 50, Appendix K (Ref. 3).

Containment pressure satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

APPLICABLE SAFETY ANALYSES (continued)

The analysis and evaluation show that under the worst case scenario, the highest peak containment pressure is 42.8 psig for Unit 1 and 38.4 psig for Unit 2 (experienced during a LOCA). The analysis shows that the peak containment temperature is 333°F for Unit 1 and 331°F for Unit 2 (experienced during an SLB). Both results meet the intent of the design basis. (See the Bases for LCO 3.6.4, "Containment Pressure," and LCO 3.6.5 for a detailed discussion.) The analyses and evaluations assume a unit specific power level of 3672.6 MWt, one containment spray train and one containment cooling train operating, and initial (pre-accident) containment conditions of 120°F and 1.0 psig. The analyses also assume a response time delayed initiation to provide conservative peak calculated containment pressure and temperature responses.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 4).

The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation results in a −3.48 psig containment pressure and is associated with the sudden cooling effect in the interior of the leak tight containment. Additional discussion is provided in the Bases for LCO 3.6.4.

The modeled Containment Spray System actuation from the containment analysis is based on a response time associated with exceeding the containment High−3 pressure setpoint to achieving full flow through the containment spray nozzles. The Containment Spray System total response time of 88.1 seconds includes Diesel Generator (DG) startup (for loss of offsite power), sequencing of equipment, containment spray pump startup, and spray line filling (Ref. 5).

APPLICABLE SAFETY ANALYSES (continued)

Containment cooling train performance for post accident conditions is given in Reference 6. The result of the analysis is that each train can provide 100% of the required peak cooling capacity during the post accident condition. The train post accident cooling capacity under varying containment ambient conditions, required to perform the accident analyses, is also shown in Reference 7.

The modeled Containment Cooling System actuation from the containment analysis is based upon a response time associated with exceeding the containment High-3 pressure setpoint to achieving full Containment Cooling System air and safety grade cooling water flow. The Containment Cooling System total response time of 65 seconds, includes signal delay, DG startup (for loss of offsite power), and service water pump startup times (Ref. 5).

The Containment Spray System and the Containment Cooling System satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

During a DBA, a minimum of one containment cooling train and one containment spray train are required to maintain the containment peak pressure and temperature below the design limits (Ref. 7). Additionally, one containment spray train is also required to remove iodine from the containment atmosphere and maintain concentrations below those assumed in the safety analysis. To ensure that these requirements are met, two containment spray trains and two containment cooling trains must be OPERABLE. The chemical aspects of iodine removal capability are addressed in LCO 3.6.7. Therefore, in the event of an accident, at least one train in each system operates, assuming the worst case single active failure occurs.

Each Containment Spray System includes a spray pump, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an ESF actuation signal and manually transferring suction to the containment sump.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.8 Hydrogen Recombiners

FOR INFORMATION ONLY

BASES

BACKGROUND

The function of the hydrogen recombiners is to eliminate the potential breach of containment due to a hydrogen oxygen reaction.

Per 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Reactors" (Ref. 1), and GDC 41, "Containment Atmosphere Cleanup" (Ref. 2), hydrogen recombiners are required to reduce the hydrogen concentration in the containment following a Loss Of Coolant Accident (LOCA) or Steam Line Break (SLB). The recombiners accomplish this by recombining hydrogen and oxygen to form water vapor. The vapor remains in containment, thus eliminating any discharge to the environment. The hydrogen recombiners are manually initiated since flammable limits would not be reached until several days after a Design Basis Accident (DBA).

Two 100% capacity independent hydrogen recombiner systems are provided and shared between the units. Each consists of controls located in the auxiliary building, a power supply and a recombiner. Recombination is accomplished by heating a hydrogen air mixture to 1325°F. The resulting water vapor and discharge gases are cooled prior to discharge from the recombiner. A single recombiner is capable of maintaining the hydrogen concentration in containment below the 4.0 volume percent (v/o) flammability limit. Two recombiners are provided to meet the requirement for redundancy and independence. Each recombiner is powered from a separate Engineered Safety Features bus, and is provided with a separate power panel and control panel.

The hydrogen recombiners are described in UFSAR, Section 6.2.5 (Ref. 3).

BASES

APPLICABLE SAFETY ANALYSES

The hydrogen recombiners provide for the capability of controlling the bulk hydrogen concentration in containment to less than the lower flammable concentration of 4.0 v/o following a DBA. This control would prevent a containment wide hydrogen burn, thus ensuring the pressure and temperature assumed in the analyses are not exceeded. The limiting DBA relative to hydrogen generation is a LOCA. Hydrogen may accumulate in containment following a LOCA as a result of:

a. A metal steam reaction between the zirconium fuel rod cladding and the reactor coolant;

b. Radiolytic decomposition of water in the Reactor Coolant System (RCS) and the containment sump;

c. Hydrogen in the RCS at the time of the LOCA (i.e., hydrogen dissolved in the reactor coolant and hydrogen gas in the pressurizer vapor space); or

d. Corrosion of metals exposed to containment spray and Emergency Core Cooling System solutions.

To evaluate the potential for hydrogen accumulation in containment following a LOCA, the hydrogen generation as a function of time following the initiation of the accident is calculated. Conservative assumptions recommended by Reference 4 are used to maximize the amount of hydrogen calculated.

Based on the conservative assumptions used to calculate the hydrogen concentration versus time after a LOCA, the hydrogen concentration in the primary containment would reach 2.6 v/o about 20 hours after the LOCA and 4.0 v/o about 4 days later if no recombiner was functioning (Ref. 3). Initiating a hydrogen recombiner when the primary containment hydrogen concentration reaches 2.6 v/o will maintain the hydrogen concentration in the primary containment below flammability limits.

B 3.7 PLANT SYSTEMS

B 3.7.3 Secondary Specific Activity

FOR INFORMATION ONLY

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.13, "RCS Operational LEAKAGE") of primary coolant at the limit of 1.0 μCi/gm (LCO 3.4.16, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives, (i.e., < 20 hours). I-131, with a half life of 8.04 days, concentrates faster than it decays, but does not reach equilibrium because of blowdown and other losses.

With the specified activity limit, the resultant 2 hour thyroid dose to a person at the exclusion area boundary (EAB) would be about 6.5 Rem if the Main Steam Safety Valves (MSSVs) open for 2 hours following a trip from full power.

Operating a unit at the allowable limits could result in a 2 hour EAB exposure of a small fraction of the 10 CFR 100 (Ref. 1) limits.

APPLICABLE SAFETY ANALYSES (continued)

In addition, the minimum available AF flow and system characteristics are serious considerations in the analysis of a small break Loss Of Coolant Accident (LOCA) and loss of offsite power (Ref. 3).

The AF System design is such that it can perform its function following an FWLB between the main feedwater isolation valves and containment, combined with a loss of offsite power following turbine trip, and a single active failure of one AF pump. The AF lines to the SGs are orificed such that at least 328 gpm is delivered to the non faulted SGs. Reactor trip is assumed to occur when the faulted SG reaches the low-low level setpoint. Sufficient flow would be delivered to the intact steam generators by the other AF pump.

During the loss of all AC power events, the Engineered Safety Feature Actuation System (ESFAS) automatically actuates the AF diesel driven pump and associated controls to ensure an adequate supply to the steam generators during loss of power. Valves which can be manually controlled are provided for each AF line to control the AF flow to each steam generator during loss of all AC power events.

The AF System satisfies the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii).

ATTACHMENT C

INFORMATION SUPPORTING A FINDING OF NO SIGNIFICANT HAZARDS CONSIDERATION

According to 10CFR 50.92 (c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

> Involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated;
>
> Create the possibility of a new or different kind of accident from any previously analyzed, or;
>
> Involve a significant reduction in the margin of safety.

In support of this determination, an evaluation of each of the three criteria set forth in 10 CFR 50.92 is provided below regarding the proposed license amendment.

Overview

Commonwealth Edison (ComEd) Company is requesting changes to Facility Operating License Nos. NPF-72, NPF-77, NPF-37 and NPF-66, and Appendix A, Technical Specifications (TS), for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively. The proposed change will revise the maximum power level specified in each unit's license, and the TS definition of rated thermal power. In addition, other TS changes associated with this power uprate request are proposed. The specific changes requested are:

- increasing the maximum power level specified in each unit's license;
- revising the value of Rated Thermal Power (RTP) in the TS definitions;
- revising the reference source for conversion factors in the calculation of Dose Equivalent Iodine (I) - 131 as noted in the TS definitions;
- adding a Departure from Nucleate Boiling Ratio (DNBR) limit specifically for a thimble cell;
- increasing the minimum limit for Reactor Coolant System (RCS) total flow;
- revising the steam generator laser welded sleeve plugging limit; and
- reducing, the peak calculated containment internal pressure $P_{a,}$ for the design basis loss of coolant accident (LOCA),

Byron Station and Braidwood Station have completed a comprehensive uprate program to increase the licensed reactor power from 3411 Megawatts-thermal (MWt) to 3586.6 MWt for Units 1 and 2 at each station. Note that the Large Break Loss of Coolant Accident (LBLOCA) analysis to specifically address 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," at uprated power conditions, will be submitted in a separate Byron Station and Braidwood Station license amendment request in December 2000. This analysis will be performed using the NRC approved Westinghouse Best Estimate LOCA model WCOBRA/TRAC.

The uprate program included a reanalysis or evaluation of all other aspects of LBLOCA, Small Break Loss of Coolant Accidents (SBLOCA), non-LOCA accidents, and Nuclear Steam Supply System (NSSS) and balance-of-plant (BOP) structures, systems, and components. Major NSSS components (e.g., reactor pressure vessel, pressurizer, reactor coolant pumps, and steam generators); BOP components (e.g., turbine, generator, and condensate and feedwater pumps); and major systems and sub-systems (e.g., safety injection, auxiliary feedwater, residual heat removal, electrical distribution, emergency diesel generators, containment cooling, and the ultimate heat sink) have been assessed with respect to the bounding conditions expected for operation at the uprated power level. Control systems (e.g., rod control, pressurizer pressure and level, turbine overspeed, steam generator level, and steam dump) have been evaluated for operation at uprated power conditions. Reactor trip and Engineered Safety Feature (ESF) actuation setpoints have been assessed and no needed changes were identified as a result of uprated power operations. The results of all of the above analyses and evaluations have yielded acceptable results and demonstrate that all design basis acceptance criteria will continue to be met during uprated power operations. This detailed analysis is presented in the "Power Uprate Licensing Report for Byron Station and Braidwood Stations," dated June 27, 2000, submitted as an attachment to this license change request.

The power uprate analyses for Byron Station and Braidwood Station were performed consistent with the guidelines set forth in Westinghouse Energy Systems Report, WCAP-10263, "A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Power Plant," dated 1983. The WCAP was submitted to the NRC for review in a letter from E. P. Rahe (Westinghouse Electric Corporation) to C. O. Thomas (NRC), "WCAP-10263 Power Uprating Topical Report Review," dated February 11, 1983. The methodology in WCAP-10263 established the ground rules and criteria for power uprate projects, including the broad categories that must be addressed, such as NSSS performance parameter, design transients, systems, components, accidents and nuclear fuel, as well as the interfaces between the NSSS and the BOP fluid systems.

The proposed Technical Specifications (TS) changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

A. Evaluation of the Probability of Previously Evaluated Accidents

Plant systems and components have been verified to be capable of performing their intended design functions at uprated power conditions. Where necessary, some components will be modified prior to implementation of uprated power operations to accommodate the revised operating conditions. The analysis has concluded that operation at uprated power conditions will not adversely affect the capability or reliability of plant equipment. Current TS surveillance requirements ensure frequent and adequate monitoring of system and component operability. All systems will continue to be operated in accordance with current design requirements under uprated conditions, therefore no new components or system interactions have been identified that could lead to an increase in the probability of any accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR). No changes were required to the Reactor Trip or Engineered Safety Features (ESF) setpoints.

B. Evaluation of the Consequences of Previously Evaluated Accidents

The radiological consequences were reviewed for all design basis accidents (DBAs) (i.e., both Loss of Coolant Accident (LOCA) and non-LOCA accidents) previously analyzed in the UFSAR. The analysis showed that the resultant radiological consequences for both LOCA and non-LOCA accidents remain either unchanged or have not significantly increased due to operation at uprated power conditions. The radiological consequences of all DBAs continue to meet established regulatory limits.

The proposed addition of Table E-7 of NRC Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, 1977, or International Commission on Radiological Protection (ICRP) 30, "Limits for Intakes of Radionuclides by Workers," Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity," for thyroid dose conversion factors, will not significantly increase the consequences of an accident previously evaluated. If Regulatory Guide 1.109, or ICRP 30, Supplement to Part 1, are used to calculate maximum dose equivalent iodine specific activity, the total RCS iodine activity may increase, depending on the iodine nuclide mix, and this activity is used to calculate the doses resulting from a Main Steam Line Break (MSLB) or other analyzed accident. The calculated thyroid doses resulting from an MSLB or other analyzed accident would not increase as the corresponding dose conversion factors would be used to calculate the offsite thyroid doses. For a given Dose Equivalent I-131 concentration in the RCS, the offsite dose predicted using the dose conversion factors in either Table E-7 of Regulatory Guide 1.109, or ICRP 30, Supplement to Part 1, is less than that predicted by Table III of Atomic Energy Commission (AEC) Technical Information Document TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," which is currently referenced in the TS definition of Dose Equivalent I-131.

ICRP-30 is the updated reference for thyroid dose conversion factors used in the power uprate accident analysis radiological evaluation. The current version of 10 CFR 20, "Standards for Protection Against Radiation," also utilizes ICRP-30 data.

Based on the analysis, it is concluded that the proposed TS changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The configuration, operation and accident response of the Byron Station and the Braidwood Station systems, structures or components are unchanged by operation at uprated power conditions or by the associated proposed TS changes. Analyses of transient events have confirmed that no transient event results in a new sequence of events that could lead to a new accident scenario.

The effect of operation at uprated power conditions on plant equipment has been evaluated. No new operating mode, safety-related equipment lineup, accident scenario, or equipment failure mode was identified as a result of operating at uprated conditions. In addition, operation at uprated power conditions does not create any new failure modes that could lead to a different kind of accident. Minor plant modifications, to support implementation of uprated power conditions, will be made as required to existing systems and components. The basic design of all systems remains unchanged and no new equipment or systems have been installed which could potentially introduce new failure modes or accident sequences. No changes have been made to any Reactor Trip or ESF actuation setpoints.

Based on this analysis, it is concluded that no new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed changes. The proposed TS changes do not have an adverse effect on any safety-related system. Therefore, the proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed TS changes do not involve a significant reduction in a margin of safety.

A comprehensive analysis was performed to support the power uprate program at the Byron Station and the Braidwood Station. This analysis identified and defined the major input parameters to the NSSS, reviewed NSSS design transients, and reviewed the capabilities of the NSSS fluid systems, NSSS/BOP interfaces, NSSS control systems, and NSSS and BOP components. All appropriate NSSS accident analysis was reperformed to confirm acceptable results were maintained and that the radiological consequences remained within regulatory limits. The nuclear and thermal hydraulic performance of nuclear fuel was also reviewed to confirm acceptable results. The analysis confirmed that all NSSS and BOP systems and components are capable, some with minor modifications, to safely support operations at uprated power conditions.

To support the operation of Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2 at uprated power conditions, nuclear fuel Departure from Nucleate Boiling Ratio (DNBR) reanalysis was required to define new core limits, axial offset limits, and Condition II, "Faults of Moderate Frequency," acceptability. This analysis included review of the following events: loss of RCS flow, reactor coolant pump locked rotor, feedwater malfunction, dropped control rod, steamline break, and control rod withdrawal from a subcritical condition. DNB design criteria was met for all events.

NUREG 1431, "Standard Technical Specifications, Westinghouse Plants," Revision 1, dated April 1995, allows Dose Equivalent I-131 to be calculated using any one of three dose conversion factors; Table III of TID-14844, 1962, Table E-7 of NRC Regulatory Guide 1.109, Revision 1, 1977, or ICRP 30, Supplement to Part 1. Using thyroid dose conversion factors other than those given in TID-14844 results in lower doses and higher allowable activity but is justified by the discussion given in the Federal Register (i.e., Federal Register (FR) page 23360 Vol. 56, May 21, 1991). This discussion accompanied the final rulemaking on 10 CFR 20, "Standards for Protection Against Radiation," by the NRC. In that discussion, the NRC stated that it was incorporating modifications to existing concepts and recommendations of the ICRP into NRC regulations. Incorporation of the methodology of ICRP-30 into the 10 CFR 20 revision was specifically mentioned with the changes being made resulting from changes and updates in the scientific techniques and parameters used in calculating dose. This FR reference clearly shows that the NRC was updating 10 CFR 20 to incorporate ICRP-30 recommendations and data. Regulatory Guide 1.109 thyroid dose conversion factors are higher than the ICRP-30 thyroid dose conversion factors for all five iodine isotopes of concern. Therefore, using Regulatory Guide 1.109 thyroid dose conversion factors to calculate Dose Equivalent I-131 is more conservative than ICRP-30 and is therefore acceptable. For a given Dose Equivalent I-131 concentration in the Reactor Coolant, the offsite dose predicted using the dose conversion factors in either Table E-7 of Regulatory Guide 1.109, Revision 1, NRC, 1977, or ICRP 30, Supplement to Part 1, is less than that predicted by Table III of TID-14844 which is currently referenced in the TS definition of Dose Equivalent I-131.

ICRP-30 is the updated reference source used in the power uprate accident analysis radiological evaluation. All regulatory acceptance criteria continue to be met and adequate safety margin is maintained.

Revising the minimum limit for RCS total flow from ≥ 371,400 gpm to ≥ 380,900 gpm does not represent a significant reduction in the margin of safety. The reactor coolant pumps run at full flow and have a total flow capacity greater than 380,900 gpm. The analysis has shown that DNBR criteria has been met for all normal operational transients and loss of flow accident scenarios.

The margin of safety of the reactor coolant pressure boundary is maintained under uprated power conditions. The design pressure of the reactor pressure vessel and reactor coolant system will not be challenged as the pressure mitigating systems were confirmed to be sufficiently sized to adequately control pressure under uprated power conditions.

The proposed change revises the plugging limit for laser welded sleeves from 40% to 38.7% of nominal wall thickness. The analysis performed in support of the power uprate effort, indicated that it is necessary to remove steam generator (SG) tubes with laser welded sleeves from service upon discovering an imperfection depth of 38.7% wall thickness to ensure the structural integrity of SG tubes which have been sleeved thereby precluding the occurrence of an SG tube rupture of sleeved tubes under all operating conditions. The previous laser welded sleeve plugging limit was based on an analysis that used lower tolerance limit material strength values. The new analysis methodology, required for laser welded sleeves, uses minimum strength properties from the American Society of Mechanical Engineers Code. As determined by the new analysis, reducing the plugging limit from 40% to 38.7% maintains a comparable margin of safety to the previous analysis.

Reanalysis of containment structural integrity under Design Basis Accident (DBA) conditions indicated that the safety margin improved, even though the mass and energy release due to a LOCA under uprated power conditions increases. Based on new and improved analytical methodologies, P_a, the peak calculated containment internal pressure for the design basis LOCA, is 42.8 psig as compared to the current value of 47.8 psig for Unit 1; and is 38.4 psig as compared to the current value of 44.4 psig for Unit 2, for both Byron Station and Braidwood Station.

Radiological consequences of the following accidents were reviewed: Main Steamline Break, Locked Reactor Coolant Pump (RCP) Rotor, Locked RCP Rotor with Power-Operated Relief Valve Failure, Rod Ejection, Small Line Break Outside Containment, Steam Generator Tube Rupture, Large Break Loss of Coolant Accident, Small Break Loss of Coolant Accident, Waste Gas Decay Tank Rupture, Liquid Waste Tank Failure, and Fuel Handling Accident. The resultant radiological consequences for each of these accidents did not show a significant change due to uprated power conditions and 10 CFR 100 limits continue to be met.

The analyses supporting the power uprate program have demonstrated that all systems and components are capable of safely operating at uprated power conditions. All design basis accident acceptance criteria will continue to be met. Therefore, it is concluded that the proposed TS changes do not involve a significant reduction in the margin of safety.

Conclusion

Based upon the above analyses and evaluations, we have concluded that the proposed changes to the operating licenses and TS involve no significant hazards consideration.

ATTACHMENT D

INFORMATION SUPPORTING AN ENVIRONMENTAL ASSESSMENT

Commonwealth Edison (ComEd) Company is requesting a change to Facility Operating License Nos. NPF-72, NPF-77, NPF-37 and NPF-66, and Appendix A, Technical Specifications (TS), for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively. The proposed change will revise the maximum power level specified in each unit's license, and the TS definition of rated thermal power. In addition, other TS changes associated with this power uprate request are requested.

ComEd has evaluated this proposed operating license amendment consistent with the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21, "Criteria for and identification of licensing and regulatory actions requiring environmental assessments." ComEd has determined that these proposed changes meet the criteria for a categorical exclusion set forth in paragraph (c)(9) of 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," and as such, has determined that no irreversible consequences exist in accordance with paragraph (b) of 10 CFR 50.92, "Issuance of amendment." This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, "Standards for Protection Against Radiation," or that changes an inspection or a surveillance requirement, and the proposed amendment meets the following specific criteria.

(i) The proposed changes involve no significant hazards consideration.

As demonstrated in Attachment C, the proposed changes do not involve a significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

Non-Radiological Effluent Releases

The Environmental Impacts Review evaluated the environmental effluent discharge permit limits to determine any impacts due to uprating the Nuclear Steam Supply System (NSSS) power to 3600.6 MWt. This includes a core thermal power of 3586.6 MWt plus 14 MWt from reactor coolant pump heat.

The assessment included determining whether the power uprate will cause the plant to exceed any effluent discharge permit limitations or other conditions associated with the operation of the plant. This review is based upon information contained in the Environmental Report and in the latest National Pollutant Discharge Elimination System (NPDES) Permits.

ATTACHMENT D

INFORMATION SUPPORTING AN ENVIRONMENTAL ASSESSMENT

Byron Station NPDES Permit Requirements

The Byron Station NPDES Permit requires, in part, that the effluent from various discharge points be monitored and limited at all times. The following NPDES discharge points were evaluated for potential effects due to operating Byron Station at uprated power conditions.

a. Cooling tower blowdown
b. Non-Essential SW blowdown and strainer backwash
c. Essential SW blowdown and strainer backwash
d. Demineralizer regenerant waste
e. Sewage treatment plant effluent
f. Wastewater treatment plant effluent
g. Radwaste treatment system effluent
h. Stormwater runoff basin
i. Secondary steam systems (i.e., non-radiological systems) process water

Additionally, the discharge of wastewater from the facility must not, alone or in combination with other sources, cause the effluent stream to violate the following thermal limitations at the edge of the mixing zone.

1. The maximum temperature rise above natural temperature must not exceed 5°F (2.8°C). The natural temperature is considered to be the ambient or upstream intake river temperature.

2. Water temperature at representative locations in the main river shall not exceed the following maximum limits during more than one percent of the hours in the 12-month period ending with any month (i.e., 87.6 available excursion hours in one year). Moreover, at no time shall the water temperature at such locations exceed the maximum limits in the following table by more than 3°F (1.7°C).

Table 11.2.1-1

Rock River Temperature Limits

Parameters	Jan.	Feb.	Mar.	Apr. – Nov.	Dec.
Temp. - °F	60	60	60	90	60
Temp. - °C	16	16	16	32	16

Braidwood Station NPDES Permit Requirements

The current Braidwood Station NPDES Permit requires, in part, that the effluent of the Cooling Pond Blowdown Line that discharges to the Kankakee River be monitored and limited at all times. The following NPDES discharge points were evaluated for potential effects due to operating Braidwood Station at uprated power conditions.

a. Condenser cooling water

b. House service water
c. Essential service water
d. Demineralizer regenerant waste
e. Wastewater treatment plant effluent
f. Radwaste treatment plant effluent
g. House service water strainer backwash
h. Essential service water strainer backwash
i. Sewage treatment plant effluent
j. Water treatment system filter backwash
k. River intake screen backwash
l. Cooling pond intake screen backwash

Additionally the discharge of wastewater from the facility must not, alone or in combination with other sources, cause the effluent stream to violate the following thermal limitations at the edge of the mixing zone.

1. The maximum temperature rise above natural temperature must not exceed 5°F (2.8°C). The natural temperature is considered to be the ambient or upstream intake temperature.

2. Water temperature at representative locations in the main river shall not exceed the following maximum limits during more than one percent of the hours in the 12-month period ending with any month. Moreover, at no time shall the water temperature at such locations exceed the maximum limits in the following table by more than 3°F (1.7°C).

Table 11.2.2-1

Kankakee River Temperature Limits

Parameters	Jan.	Feb.	Mar.	Apr. – Nov.	Dec.
Temp. - °F	60	60	60	90	60
Temp. - °C	16	16	16	32	16

As noted above, normal blowdown is via the cooling pond blowdown line to the Kankakee River.

Byron Station Effluent Analysis and Evaluation

The Circulating Water (CW) System at Byron Station is a closed loop cooling system designed to dissipate waste heat from the turbine cycle to the atmosphere using natural draft cooling towers; one tower for each unit. Tower blowdown is accomplished by diverting flow from the circulating water system downstream of the CW pumps and upstream of the condenser and tower and discharging it to the Rock River.

The increase in heat associated with the power uprate will primarily affect the CW system and will be approximately 5 percent higher than the heat at the present power level. This will result in a 1°F CW temperature increase. The current CW temperature rise is approximately 22°F at 100% power. Although the NPDES Permit does not specify a maximum cooling tower blowdown temperature, it controls temperature at the edge of the mixing zone in the river. It has been determined that under a worst-case scenario, the tower blowdown temperature would be approximately 120°F and has set this value as the administrative limit. Assuming a nominal summer river supply temperature of 70°F - 90°F and a cooling tower blowdown temperature of 96°F, the proposed power uprate will not impact the 120°F administrative limit. Continuous blowdown from the cooling tower basin to the Rock River maintains control of dissolved solids in the tower basin.

Under most circumstances, the two-unit Byron Station is capable of operating at full load with cooling tower consumption losses supplied by a net withdrawal rate no greater than 10% of the Rock River flow. The limits are as follows.

1. Limit water withdrawal for make-up to a maximum of 125 cfs.

2. Limit net water consumption to no more than 9% of the Rock River's flow when the flow is at or below 679 cfs, the one-day ten-year low.

The average cooling tower makeup rate is between 30,000 - 35,000 gpm (i.e., both towers combined) while the average blowdown rate is 13,000 gpm. This makeup rate is approximately 6.6% of the two-day average minimum river water flow rate of 1,187 cfs (i.e., 533,000 gpm) and less than 2% of the two-day average rate of 4,575 cfs (i.e., 2.05 E6 gpm). Additionally, Byron Station, must discharge less than 0.5 billion BTUs/hour, in accordance with the Illinois Administration Code Title 35, "Environmental Protection," Subtitle C, "Water Pollution," Chapter 1, Section 302.211(f) regulations. The requirement will continue to be met following the power uprate.

Braidwood Station Effluent Analysis and Evaluation

The CW System at Braidwood Station is a closed loop cooling system similar to that at the Byron Station except that waste heat is rejected from the turbine cycle to a cooling lake. Three CW pumps per unit pump cooling water from the lake to the main condenser. Discharge from the condenser is returned to the lake, where it is separated from the intake supply by a dike.

Makeup water to the lake is pumped from the Kankakee River. Under most circumstances, the two-unit Braidwood Station is capable of operating at full load with cooling lake consumption losses supplied by a maximum withdrawal rate no greater than 160 cfs of the Kankakee River. The limits are as follows.

1. Limit withdrawal of Kankakee River water to a maximum of 160 cfs.

2. Stop withdrawing water from the Kankakee River when the flow in the river is 442 cfs (i.e., the seven-day 10 year low flow) or less; and not to withdraw water such that the flow of the river is diminished below 442 cfs.

The plant currently operates at a withdrawal rate of approximately 110 cfs for makeup and blows down at the rate of approximately 28 cfs. Water chemistry is controlled by continuous blowdown of supply water to the condenser and makeup to the cooling lake.

The heat duty increase associated with power uprate is mainly associated with the CW System and will be approximately 5% higher than at the present power level. This will result in a 1°F increase to the CW temperature rise, which is now approximately 21.8°F at 100% power. The increase will nominally increase the lake temperature as the lake temperature is primarily influenced by climatic conditions. Current cooling lake makeup and blowdown requirements should remain acceptable and within NPDES limits following power uprate. The NPDES permit contains a blowdown heat rejection limit to the river that requires blowdown discharge be less than 0.5 billion BTUs/hr, in accordance with Title 35, Subtitle C Chapter 1 Section 302.211(f) regulations.

Noise Evaluation

The noise effects due to operation of Byron Station and Braidwood Station at uprated power conditions were reviewed. No increase in noise from the turbine or reactor building will result due to uprated power operations. In addition, the turbine and the reactor building supply and exhaust fans will continue to operate at current speeds and the associated noise levels will also be unaffected by uprated power operations. In summary, the overall noise levels at Byron Station and Braidwood Station will not increase due to the power uprate.

Conclusions

The non-radiological environmental impacts related to the proposed power uprate at Byron Station and Braidwood Station have been reviewed and there are no adverse impacts or significant changes required to the current NPDES Permits or other plant administrative limits. There is no significant change in the types or a significant increase in the amounts of non-radiological effluents that may be released offsite.

Radiological Effluent Releases

Liquid and gaseous effluents released to the environment during normal plant operations contain small quantities of radioactive materials. As noted in the Byron Station and Braidwood Station Updated Final Safety Analysis Report (UFSAR), Chapter 11, "Radioactive Waste Management," the original plant analyses demonstrated that the radioactive releases from the site are within the radioactive release and dose limits set by 10 CFR 20 and 10 CFR 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low As Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents." The impact of the power uprate on these releases has been evaluated to confirm continued operation within regulatory limits.

In evaluating the impact of the core uprate on radwaste effluents, the methodology found in NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors," was used to establish the relative change in expected reactor coolant radiological activity. The expected percentage change in the reactor coolant activity is estimated to be less than or equal to the percentage change in core power.

For liquid releases, the magnitude of the releases after the power uprate is directly proportional to the increase in reactor coolant activity caused by operations at uprated power operations. For the gaseous releases, the analysis is more complex as gaseous effluents are composed of two components:

1. non-containment leakage from the reactor coolant system or secondary side steam, and normal operation gaseous waste effluents, which are reactor coolant concentration based; and

2. effluents from the gaseous waste system during shutdown sequences and reactor coolant leakage into containment, which are also coolant inventory based.

An upper bound analysis for the potential impact of the power uprate indicates that the increase in radiological releases and resultant dose impact is bounded by the percentage increase in the reactor core power. Note that the original analyses were performed at a reactor core power level of 3565 MWt (i.e., approximately 0.6% less than the proposed uprated power level of 3586.6 MWt).

An evaluation was performed, using scaling techniques, to assess the impact of the power uprate on normal operation annual effluent releases and Appendix I doses.

Expected Reactor Coolant Source Terms

Based on a comparison between the original input parameters and the power uprate input parameters, using the methodology outlined in NUREG 0017 the maximum expected increase in the reactor coolant source term is limited to the percentage increase of the power uprate, i.e., 0.6%. With the exception of the long-lived isotope, Kr-85, the noble gases actually have a lower reactor coolant concentration for the power uprate case. This is primarily due to the conservatively low water mass used in the original analysis vice the more realistic values utilized for the power uprate evaluation. Considering the accuracy and error bounds of the operational data utilized in NUREG 0017, this small percentage increase (i.e., 0.6%) in reactor coolant source term, is well within the uncertainty of the existing NUREG 0017 based expected reactor coolant isotopic inventory used for radwaste effluent analyses.

Gaseous and Liquid Effluent Releases

There was approximately a 0.6% increase in the liquid effluent release concentrations, as this activity is based on the long-term RCS activity which is proportional to the power uprate percentage increase and on waste volumes, which are essentially independent of power level within the applicability range of NUREG 0017. Tritium releases in liquid effluents remain unchanged due to uprated power conditions as the coolant concentration is set by the NUREG 0017 methodology.

Gaseous releases for Kr-85 increased proportionally by the 0.6% power increase. However, isotopes with shorter half-lives can have either reduced releases or slight increases, as compared to the percentage change in power level. For example, releases of Xe-133 will increase about 0.2%. The impact of the power uprate on iodine releases is limited to the 0.6% power level increase. There will be an approximate 0.8% increase in tritium production due to the power uprate. It is assumed that this incremental tritium increase will be released via the normal gaseous effluent pathway. The other components of the gaseous release (i.e., particulates via the building ventilation systems and water activation gases) are not impacted by the power uprate.

Considering the accuracy and error bounds of the operational data utilized in NUREG 0017, these small percentage changes are well within the uncertainty of the calculated results in the existing NUREG 0017 based gaseous and liquid release isotopic inventory presented in UFSAR Tables 11.2-4 and 11.3-7.

10 CFR 50 Appendix I Doses

Since the maximum increase due to the power uprate, relative to the liquid releases, is approximately 0.6%, the increase in the estimated Appendix I doses via the liquid pathway will also be bounded by 0.6%.

With respect to the gaseous pathway, the noble gases, iodines and particulates contribute to over 90% of the Appendix I dose, whereas the tritium contribution to dose is less than 10%. The impact of uprate on the noble gas, iodine and particulate contribution to the Appendix I doses will be less than 0.6%. Though the incremental dose contribution due to the increase in tritium releases due to uprate is 0.8%, it is a small contributor (i.e., <10%) to the total Appendix I dose from gaseous effluents. Therefore, the overall increase in offsite dose due to the gaseous pathway is estimated to be less than 0.6%.

Considering the accuracy and error bounds of the operational data utilized in NUREG 0017, this small percentage change is well within the uncertainty of the calculated results in the existing NUREG 0017 based Appendix I dose estimates documented in UFSAR Tables 11.2-3 and 11.3-9.

Solid Waste Generation

Per regulatory guidance for a "new" facility, the estimated volume and activity of solid waste is linearly related to the core power level. However, for an existing facility that is undergoing power uprate, the volume of solid waste would not be expected to increase proportionally since the power uprate neither appreciably impacts installed equipment performance, nor does it require drastic changes in system operation. Only minor, if any, changes in waste generation volume are expected.

Since the estimated reactor coolant activity does not change appreciably, the calculated specific activity of the solid waste would also not be expected to significantly change as maintenance and operational practices are not affected by the power uprate. Therefore, the power uprate has no significant impact on solid waste generation.

Conclusions

This evaluation has shown that the power uprate will not cause a significant change in the types or a significant increase in the amounts of any radiological effluent that may be released offsite. The liquid and gaseous radwaste system design remains capable of maintaining the normal operational offsite releases and doses within the limits of 10 CFR 20 and 10 CFR 50, Appendix I.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The potential effects of power uprate conditions on the radiation sources within the plant and the radiation levels during normal and post-accident conditions were evaluated. The original calculations for determining the normal operational doses and radiation shielding requirements were very conservative and had additional margin assumed in the calculations. It was determined that these margins are sufficient to accommodate any increases attributed to the five percent increase in rated thermal power. For post-accident conditions, the resulting radiation levels were determined to be within current regulatory limits, and that there would be no effect on the plant equipment, access to vital areas, or habitability of the control room envelope and the Technical Support Center.

There will be no significant change in the level of controls or methodology used for the processing of radioactive effluents or handling of solid radioactive waste, nor will the proposed changes result in a significant change in the normal radiation levels within the plant. Therefore, there will be no significant increase in individual or cumulative occupational radiation exposure resulting from this change.

The calculated whole body and thyroid doses at the exclusion area boundary that might result from a postulated design basis LOCA were evaluated. All offsite doses evaluated at uprated power conditions remain below established regulatory limits.

It is therefore concluded that there will be no significant increase in individual or cumulative occupational radiation exposure resulting from this change.

ATTACHMENT E

POWER UPRATE LICENSING REPORT

FOR

BYRON STATION

AND

BRAIDWOOD STATION

POWER UPRATE LICENSING REPORT

For

Byron Station
and
Braidwood Station

ComEd

A Unicom Company

Commonwealth Edison Company

Revision 1
June 27, 2000

POWER UPRATE LICENSING REPORT

FOR

BYRON STATION
AND
BRAIDWOOD STATION

UNITS 1 & 2

Revision 1
June 27, 2000

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LIST OF ACRONYMS

AB	Boric Acid Processing
ADV	Atmospheric Dump Valve
AE	Auxiliary Equipment
AF	Auxiliary Feedwater
ANC	Advanced Nodal Code
ANS	American Nuclear Society
ARI	All Rods In
ARO	All Rods Out
ART	Adjusted Reference Temperature
AS	Auxiliary Steam
ASME	American Society of Mechanical Engineers
AVB	Anti-Vibration Bar
B&PV	Boiler and Pressure Vessel
B/B	Byron/Braidwood
BE	Best Estimate
BELOCA	Best Estimate Loss of Coolant Accident
BOC	Beginning of Cycle
BOL	Beginning of Life
BOP	Balance of Plant
BRS	Boron Recycle System
BTRS	Boron Thermal Regeneration System
BWI	Babcock & Wilcox International
CAOC	Constant Axial Offset Control
CB	Condensate Booster
CD	Condensate
CC	Component Cooling Water
CD & FW	Condensate and Feedwater
CHF	Critical Heat Flux
COLR	Core Operating Limits Report

LIST OF ACRONYMS (cont.)

COMED	Commonwealth Edison Company
COMS	Cold Overpressure Mitigation System
CR	Control Room
CRDM	Control Rod Drive Mechanism
CRSD	Core Radiation Source Data
CS	Containment Spray
CS	Control Systems
CSAU	Code Scaling, Applicability, and Uncertainty
CST	Condensate Storage Tank
CV	Centrifugal Charging Pump
CW	Circulating Water
CVCS	Chemical and Volume Control System
DBD	Design Basis Document
DBE	Design Basis Earthquake
DC	Downcomer
DCF	Dose Conversion Factor
DECLG	Double-Ended Cold Leg Guillotine
DEH	Digital Electro-Hydraulic
DEHL	Double-Ended Hot Leg
DEPS	Double-Ended Pump Suction
DER	Double Ended Rupture
DF	Decontamination Factor
DFBN	Debris Filter Bottom Nozzle
DG	Diesel Generator
DNB	Departure From Nucleate Boiling
DNBR	Departure From Nucleate Boiling Ratio
DRP	Design Review Package
DT	Design Transients
ECCS	Emergency Core Cooling System

LIST OF ACRONYMS (cont.)

EDG	Emergency Diesel Generator
EFPY	Effective Full Power Years
EH	Electro-Hydraulic
EOC	End of Cycle
EOL	End of Life
EOP	Emergency Operating Procedures
EQ	Environmental Qualification
ERG	Emergency Response Guidelines
ES	Extraction Steam
ESF	Engineered Safety Features
ESW	Essential Service Water
FA	Fuel Assemblies
FAC	Flow Accelerated Corrosion
FBS	Feedwater Bypass System
FC	Spent Fuel Pool Cooling
FCV	Flow Control Valve
FF	Fouling Factor
FHA	Fuel Handling Accident
FIV	Feedwater Isolation Valves
FIV	Flow-Induced Vibrations
FLB	Feedwater Line Break
FS	Fluid Systems
FSAR	Final Safety Analysis Report
FTI	Framatome Technologies, Incorporated
FW	Feedwater
FWP	Feedwater Pump
FWRV	Feedwater Regulator Valve
GDC	General Design Criteria
GDT	Gas Decay Tanks

LIST OF ACRONYMS (cont.)

GPM	Gallons Per Minute
GSC	Gland Steam Condensers
HEI	Heat Exchanger Institute
HD	Heater Drain
HDKP	Heavy Disc and Key-Plate
HFF	Hydraulic Forcing Functions
HFP	Hot Full Power
HHSI	High Head Safety Injection
HL	Hot Leg
HLSO	Hot Leg Switch Over
HVAC	Heating, Ventilating & Air Conditioning
HX	Heat Exchanger
HZP	Hot Zero Power
ICRP	International Commission on Radiological Protection
IFBA	Integral Fuel Burnable Absorbers
IFMs	Intermediate Flow Mixing Grids
LBB	Leak Before Break
LBLOCA	Large Break Loss of Coolant Accident
LDKP	Light Disc and Key-Plate
LHSI	Low-Head Safety Injection
LOCA	Loss of Coolant Accident
LOF	Loss of Flow
LOL/TT	Loss of Load/Turbine Trip
LONF	Loss of Normal Feedwater
LOOP	Loss-of-Offsite Power
LP	Loading Pattern
LPZ	Low Population Zone
LR	Licensing Report
LSIV	Loop Stop Isolation Valve

LIST OF ACRONYMS (cont.)

LTCC	Long Term Core Cooling
LTOP	Low Temperature Overpressure Protection
MD	Motor Driven
M&E	Mass and Energy
MFW	Main Feedwater
MOL	Middle of Life
MOVs	Motor Operated Valves
MS	Main Steam
MSIBV	Main Steamline Isolation Bypass Valve
MSIV	Main Steamline Isolation Valve
MSLB	Main Steam Line Break
MSR	Moisture Separator Reheater
MSSV	Main Steam Safety Valve
MTC	Moderator Temperature Coefficient
MW	Megawatt
MW_e	Megawatt Electric
MW_t	Megawatt Thermal
NF	Neutron Fluence
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NSR	Normalized Stability Ratio
NSSS	Nuclear Steam Supply System
NUPPSCO	Nuclear Power Plant Standard Committee
ODSCC	Outside Diameter Stress Corrosion Cracking
OG	Off Gas
OPΔT	Overpower Delta Temperature
OTΔT	Overtemperature Delta Temperature
P&I	Proportional plus Integral
P/L	Part Length

LIST OF ACRONYMS (cont.)

PBSR	Pressure Boundary Summary Report
PCT	Peak Clad Temperature
PCWG	Performance Capability Working Group
PLS	Precautions, Limitations and Setpoints
PMTC	Positive Moderator Temperature Coefficient
PORV	Power Operated Relief Valve
PRT	Pressurizer Relief Tank
PS	Process Sampling
PSARV	Pressurizer Safety and Relief Valve
PSS	Process Sampling System
P-T	Pressure-Temperature
PTS	Pressurized Thermal Shock
PW	Primary Water
PWR	Pressurized Water Reactor
PWSCC	Primary Water Stress Corrosion Cracking
RAOC	Relaxed Axial Offset Control
RC	Reactor Coolant
RCCA	Rod Cluster Control Assembly
RCFC	Reactor Containment Fan Cooler
RCL	Reactor Coolant Loop
RCLP&S	Reactor Coolant Loop Piping and Supports
RCP	Reactor Coolant Pump
RCPON	Reactor Coolant Pump Outlet Nozzle
RCS	Reactor Coolant System
RD	Radiation Doses
RG	Regulatory Guide
RH	Residual Heat
RHR	Residual Heat Removal
RIP	Rod Internal Pressure

LIST OF ACRONYMS (cont.)

RMCS	Reactor Makeup Control System
RP	Rod Performance
RPV	Reactor Pressure Vessel
RPVIN	Reactor Pressure Vessel Inlet Nozzle
RPVS	Reactor Pressure Vessel System
RSAC	Reload Safety Analysis Checklist
RSE	Reload Safety Evaluation
RSG	Replacement Steam Generator
RSR	Relative Stability Ratio
RTDP	Revised Thermal Design Procedure
RTSR	Reload Transition Safety Report
RV	Reactor Vessel
RWST	Refueling Water Storage Tank
SAL	Safety Analysis Limit
SBLOCA	Small Break Loss of Coolant Accident
SCC	Stress Corrosion Cracking
SDM	Shutdown Margin
SER	Safety Evaluation Report
SFP	Spent Fuel Pool
SFPCS	Spent Fuel Pool Cooling System
SG PORV	Steam Generator Power Operated Relief Valves
SG	Steam Generator
SGTP	Steam Generator Tube Plugging
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SIS	Safety Injection System
SLB	Steam Line Break
SPIL	Safety Parameter Interaction List
SSE	Safe Shutdown Earthquake

LIST OF ACRONYMS (cont.)

STDP	Standard Thermal Design Procedure
SX	Essential Service Water
T_{avg}	RCS Average Temperature or Vessel Average Temperature
T_{cold}	Reactor Coolant System Cold Leg Temperature
TD	Turbine Driven
TDF	Thermal Design Flow
TDH	Total Developed Head
T_{hot}	Reactor Coolant System Hot Leg Temperature
TG	Turbine Generator
T&H	Thermal & Hydraulics
TMI	Three Mile Island
UFSAR	Updated Final Safety Analysis Report
UHS	Ultimate Heat Sink
USE	Upper Shelf Energy
V5	VANTAGE 5
VAR	Volt-Ampere Reactance
VCT	Volume Control Tank
W	Westinghouse
WOG	Westinghouse Owners Group
WS	Non-Essential Service Water

EXECUTIVE SUMMARY

This report summarizes the evaluations performed that justify uprating the licensed thermal power at Byron and Braidwood Generating Stations by approximately 5% to 3586.6 MWt, with an equivalent increase (i.e., approximately 5%) in electrical generation. This report follows previously NRC approved power uprate report formats and content for other uprate projects including North Anna, Callaway, Farley, Vogtle and Wolf Creek.

This Licensing Report provides a description of the analyses and evaluations performed for the project, and provides the information required by the NRC to approve the power uprate license change for the Byron and Braidwood Generating Stations. The methodology and acceptance criteria utilized in this report are described in Section 1.2.

Based upon the various evaluations and analyses performed for this project, there are no Nuclear Steam Supply System (NSSS) modifications required to achieve the uprated core thermal power conditions, as the projected power increase can be accomplished within the existing design margins for the systems and components. Minor Balance of Plant (BOP) modifications are required to accommodate the increased system operating conditions. Major High Pressure Turbine modifications are required to meet the increased volumetric steam flow conditions as well as to restore operating margin.

Where judged to be appropriate, the lessons learned from prior power uprate licensing reports and the responses to previous NRC requests for additional information have been incorporated into this report.

This licensing report is arranged as follows:

Section 1 - Introductory material related to the uprate effort.

Sections 2 through 7 – Description of NSSS, safety and fuel analyses and dose consequences.

Section 8 – Description of Turbine Generator analyses and modifications.

Sections 9 through 12 – Description of BOP analyses, including environmental and radiological evaluations and review of impact on Station procedures.

This report does not include a description of the Best Estimate LOCA (BELOCA) program activities for the Large Break analyses, which are being conducted by Westinghouse and Commonwealth Edison Company as a separate effort. Commonwealth Edison will address the BELOCA program in a separate licensing submittal that will be submitted in support of the power uprate effort.

A power uprate to 105% of current rated thermal power is evaluated in this report. The applicable plant licensing issues have been addressed, and this uprate can be accommodated without a significant increase in the probability or consequences of an accident previously evaluated, without creating the possibility of a new or different type of accident from any accident previously evaluated, and without exceeding any presently existing regulatory limits applicable to the plants, which may cause a significant reduction in the margin of safety.

1.0 INTRODUCTION

1.1 Purpose and Scope

The purpose of performing the Byron/Braidwood Power Uprate Project Nuclear Steam Supply Systems (NSSS) and Balance of Plant (BOP) analyses and evaluations is to demonstrate that the NSSS and BOP will remain in compliance with applicable licensing criteria and requirements and can operate acceptably at the increased thermal/electrical power conditions. This report summarizes the analyses and evaluations and their results.

Like most nuclear units, the Byron and Braidwood units were originally designed with equipment and systems capable of accommodating operating conditions above the original licensed power rating including higher pressures, flows, and temperatures. In addition, continuing improvements in the analytical techniques, plant performance feedback, and improved fuel and core designs have resulted in a significant increase in the difference between the calculated safety analyses results and the licensing limits. These available safety margins, combined with the excess margin in the as-designed equipment, system and component capabilities, provide the Byron and Braidwood plants with the potential for an increase in their thermal power rating of 5% without major NSSS or BOP hardware modifications and with no significant increase in the hazards presented by the plant as approved by the NRC at the original license stage.

Unless otherwise stated, all analyses within this report are applicable to Byron Station Units 1 and 2, and Braidwood Station Units 1 and 2. With the exception of features such as the Ultimate Heat Sinks and steam generators, the units' physical systems configurations, instrumentation and control systems, and reactor protection systems are virtually identical such that the analyses adequately address all related issues stemming from the change to uprated power conditions. The thermal hydraulic characteristics of all four units are very similar.

In support of the Byron/Braidwood Power Uprate Project, the following organizations have performed analyses and evaluations to demonstrate that the Byron/Braidwood units will remain in compliance with applicable licensing criteria and requirements at the uprated power levels.

- Westinghouse Electric Company
- Babcock & Wilcox Canada Ltd.
- Stone & Webster Engineering Corporation

- Siemens Westinghouse Power Corporation
- Commonwealth Edison Company

The scope of each of the organizations is as follows:

Westinghouse Electric Company

Westinghouse analyses and evaluations included the NSSS performance parameters, design transients, systems, components, accidents and nuclear fuel. The results of the Westinghouse NSSS analyses and evaluations satisfy the project purpose and demonstrate that applicable licensing criteria and requirements are satisfied for the NSSS performance parameters, design transients, systems, components, accidents and nuclear fuel at the uprated power conditions. The power uprate analyses and evaluations described in this report were based on the parameters listed in Section 2 of this report.

Babcock & Wilcox Canada Ltd.

Babcock & Wilcox performed the requalification activities for the unit 1 replacement steam generators for the uprated conditions.

Stone & Webster Engineering Corporation

Stone & Webster analyses and evaluations included the Balance of Plant (BOP) systems and components, including radiological and environmental evaluations. Stone & Webster also reviewed the impact on Station procedures.

Siemens Westinghouse Power Corporation

Siemens Westinghouse Power Corporation performed analyses and evaluations of the turbine-generator and accessories for the uprated conditions.

Commonwealth Edison Company

Commonwealth Edison Company performed limited analyses such as steam generator tube rupture and transmission and distribution analysis for the uprated conditions.

1.2 Methodology and Acceptance Criteria

Throughout this report, reference is made to "original" and "current" parameters and design values. In general, the use of the term "original" refers to the parameters and design values which were used in the original (circa 1972-78) analyses of the Byron and Braidwood Units 1 and 2 safety and systems analyses, and components design analyses. In Section 2, Tables 2.1-1 and 2.1-2 list the "original" plant parameters.

Since initial operation, the Byron and Braidwood Units 1 and 2 have undergone several major reanalysis campaigns, including the T-Hot Reduction program, (in 1987), which introduced an operating temperature range for the units. Subsequently, in 1994 and 1995, additional analyses were performed to support operations with increased steam generator tube plugging (SGTP) and a reduced thermal design flowrate (TDF Reduction), as well as operating with a positive moderator temperature coefficient (PMTC). Most recently, in 1997, additional analyses were performed to support operations with the replacement steam generators (RSG) in Byron and Braidwood Units 1. The parameters used for each of these major reanalysis campaigns would be referred to as current if the parameters resulted in a change in the basis for each of the particular analysis areas.

Analyses and evaluations performed for this uprate project were based on the parameters generated for the uprate conditions, and when necessary, comparisons against original or current were made to assess the changes due to the uprate.

For the majority of the safety analyses, the most current parameters (those generated for the RSG or TDF Reduction programs) were used since they formed the bases for the analyses on record at Westinghouse. For the majority of the systems and components evaluations and analyses, the current parameters were the parameters generated for the T-Hot Reduction program (since for the RSG and TDF Reduction programs, little changes were effected for systems and components). And for several of the components, (e.g., the reactor coolant pump (RCP), loop stop isolation valve (LSIV), etc.) the evaluations were based on a comparison to the original design parameters since the methods employed for those evaluations considered the overall change from original to uprate parameters.

In all cases for all uprate analyses and evaluations, the uprate parameters contained in Section 2.0 were used as the basis for analyses and evaluations described in this report.

1.2.1 Nuclear Steam Supply Systems

The methodology utilized in evaluating the impact on the NSSS has been structured consistent with the methodology established in Westinghouse WCAP-10263, "A Review Plan for Uprating the Licensed Power of a PWR Power Plant", dated 1983. Since submittal of WCAP-10263 to the NRC, the methodology has been used successfully as a basis for power uprate projects on over twenty pressurized water reactor (PWR) units, including Farley Units 1 and 2, Vogtle Units 1 and 2 and Turkey Point Units 3 and 4.

The methodology in WCAP-10263 established the ground rules and criteria for power uprate projects, including the broad categories that must be addressed, such as NSSS performance parameters, design transients, systems, components, accidents and nuclear fuel as well as the interfaces between the NSSS and the BOP fluid systems. Inherent in this methodology are key points that promote correctness, consistency and licensability. The key points include the use of well-defined analysis input assumptions/parameter values, use of currently approved analytical techniques (e.g., methodologies and computer codes) and use of currently applicable licensing criteria and standards.

The power uprate analyses and evaluations were performed in accordance with Westinghouse quality assurance requirements defined in the Westinghouse Quality Management System (QMS) procedures, which comply with 10 CFR 50 Appendix B criteria. These analyses and evaluations are in conformance with Westinghouse and industry codes, standards and regulatory requirements applicable to Byron/Braidwood Units 1 and 2. Assumptions and acceptance criteria are provided in the appropriate sections of this report.

1.2.2 Balance of Plant

The methodology utilized in evaluating the impact upon the BOP is consistent with the methodology used for previously approved power uprate projects for other utilities. The initial task was to identify the parameters and design inputs that were to be used in the various BOP system, structure and component evaluations. Detailed evaluations and analyses were then performed based upon the following categories:

- Bounded by existing analysis and design conditions
- Bounded by design with reanalysis
- Not bounded by analysis or design

This methodology is discussed in greater detail in Section 9.2 of this document.

1.2.3 Computer Codes Utilized in Uprate Analyses

Westinghouse utilized various computer codes in the analyses and evaluations for the Byron/Braidwood Power Uprate Project. Only the major codes that were used to obtain safety and operating limits are listed in the table below.

Table 1.2.3-1 Computer Codes

Evaluation Subject	Computer Code	Previously Used on Byron/Braidwood	Reference Section
Non-LOCA SLB Mass and Energy Control Systems	LOFTRAN	Yes	6.2 6.5.2/6.5.1 4.3
Non-LOCA	FACTRAN	Yes	6.2.13, 6.2.19
Non-LOCA	TWINKLE	Yes	6.2.19
Non-LOCA	THINC IV	Yes	6.2.13
Steam Generator Tube Rupture	RETRAN	Yes	6.3
Small Break LOCA	NOTRUMP	Yes	6.1.1
Small Break LOCA	LOCTA-IV	Yes	6.1.1
Containment	COCO	Yes	6.4
Containment	COMPACT	No – see note 1 below	6.5
LOCA Mass and Energy	SATAN VI	No – see note 2 below	6.4
LOCA Mass and Energy	WREFLOOD	No – see note 2 below	6.4
LOCA Mass and Energy	FROTH	No – see note 2 below	6.4
LOCA Mass and Energy	EPITOME	No – see note 2 below	6.4
Radiation Source Terms	ORIGEN 2	No – see note 3 below	7.6
Heat Generation Rates	DORT	No – see note 3 below	7.4
Reactor Vessel Internals	THRIVE	Yes	5.2
Dose Analysis	TITAN 5	Yes	6.7

1. This is the first application for the COMPACT computer program to Byron/Braidwood's outside containment MSLB analysis. The COMPACT code was initially approved for use in equipment qualification applications for the Sequoyah units (Youngblood, B. J. of the NRC, "Transmittal of Draft Copy of the Evaluation on Equipment Qualification Under Superheat Conditions for Sequoyah Units 1&2," letter to S. A. White, Tennessee Valley Authority, 11-25-86.), and was most recently used for the steam generator replacement analysis for the Farley Units.
2. This is the first application for these computer programs to the Byron/Braidwood LOCA mass and energy release analysis; the methodology, presented in WCAP-10326-A, has been approved by the NRC.
3. This is the first application for these computer programs to the Byron/Braidwood units; however, these are industry standard programs, developed by the Oak Ridge National Laboratory and based on the latest industry experimental data.

1.3 Technical Basis for Significant Hazards Evaluation

This report provides the technical basis for the significant hazards evaluation included with the associated License Amendment Request for Byron and Braidwood Stations.

1.4 Updated Final Safety Analysis Report Revisions

The Byron and Braidwood Stations Updated Final Safety Analysis Report (UFSAR) has been reviewed for necessary revisions prompted by the power uprate effort. The associated Design Review Packages (DRPs) (i.e., UFSAR revisions) are currently being developed and will be formally issued for use concurrent with the NRC approval of the proposed power uprate license amendment. These DRPs will be incorporated into the UFSAR hard copy with the next UFSAR update following the anticipated NRC approval of the proposed power uprate license amendment. This UFSAR hard copy revision will be published in December 2002. Until that time, an electronic change log is maintained to identify all approved UFSAR changes not yet incorporated into the UFSAR hard copy. Hard copies of all DRPs are also available at the station for reference.

1.5 Plant Impacts Due to Power Uprate

No hardware modifications are required to the NSSS systems or components.

Only minor hardware changes are required in the BOP systems. These include an increase in valve trim in a few of the heater drain control valves, replacement of several pressure transmitters, a piping modification to increase HP turbine gland steam removal capability, and an increase in the turbine driven feedwater pump operating speed setpoint. Other changes in the BOP include setpoint/scaling and station procedure changes.

HP and LP turbine enhancements are required to accommodate the increased steam flows for the full uprate program. Post-uprate related activities will be formally tracked in the stations' commitments tracking program.

1.6 Conclusions

The applicable plant licensing issues have been addressed, and this uprate can be accommodated without a significant increase in the probability or consequences of an accident previously evaluated, without creating the possibility of a new or different type of accident from any accident previously evaluated, and without exceeding any presently existing regulatory limits applicable to the plants, which may cause a significant reduction in the margin of safety.

No hardware modifications are required to the NSSS systems or components. Only minor hardware changes are required in the BOP systems to maintain acceptable margins. These include an increase in valve trim in a few of the heater drain control valves, replacement of several pressure transmitters, a piping modification to increase HP turbine gland steam removal capability, and an increase in the turbine driven feedwater pump operating speed setpoint. Other changes in the BOP include setpoint/scaling and station procedure changes.

2.0 NUCLEAR STEAM SUPPLY SYSTEM (NSSS) PARAMETERS

The power uprate project included NSSS performance analyses to develop bounding NSSS Performance Capability Working Group (PCWG) Parameters for use in the analyses and evaluations of the NSSS, including NSSS design transients, systems, components, accidents, and nuclear fuel.

2.1 PCWG Parameters

2.1.1 Introduction and Background

The NSSS PCWG parameters are the fundamental parameters which are used as input in all the NSSS analyses. They provide the Reactor Coolant System (RCS) and secondary system conditions (temperatures, pressures, flow) that are used as the basis for the design transients, systems, components, accidents, and fuel analyses and evaluations.

The PCWG parameters are established using conservative assumptions in order to provide bounding conditions to be used in the NSSS analyses. For example, the RCS flow assumed in generating the primary and secondary side conditions is the Thermal Design Flow (TDF), which is a conservatively low flow that accounts for flow measurement uncertainty and assumes the maximum steam generator tube plugging (SGTP) level. The PCWG parameters were determined such that Commonwealth Edison would have operating flexibility. A range of conditions was therefore established for the vessel average temperature (T_{avg}) (i.e., RCS average temperature) and the SGTP level. The T_{avg} range was specified between 575° and 588°F, while the SGTP level can vary from 0% to 5% for Byron Unit 1 and Braidwood Unit 1 and 0% to 10% for Byron Unit 2 and Braidwood Unit 2. An uprated NSSS power level of 3600.6 MWt and a TDF value of 92,000 gpm/loop were also used to generate the PCWG parameters.

2.1.2 Input Parameters and Assumptions

The major input parameters and assumptions used in the calculation of the PCWG parameters established for the uprate project are summarized below:

- The power level for the uprating was set at 3600.6 MWt NSSS (3586.6 MWt core). This is approximately 5% higher than the current NSSS power rating of 3425 MWt (3411 MWt core).

- The TDF of 92,000 gpm/loop incorporates sufficient margin to support SGTP consistent with the values described below. This flow was applied for all cases, even those which assumed 0% SGTP, in order to be consistent and for conservatism.

- The following values of SGTP were assumed: 0%-5% for Unit 1 and 0%-10% for Unit 2.

- Design core bypass flow was assumed to be 8.3% with thimble plugs not installed.

- A range of full power normal operating T_{avg} from 575° to 588°F was selected for the analyses. This temperature range is sufficiently wide enough to cover the potential operation of all units.

2.1.3 Discussion of Parameter Cases

Tables 2.1-1 and 2.1-2 provide the NSSS PCWG parameter cases which were generated and used as the basis for the uprating project. The original design parameters are also shown for comparison purposes. A description of the uprated cases follows:

Table 2.1-1 lists the parameters for Byron Unit 1 and Braidwood Unit 1 which have BWI Replacement Steam Generators. The Unit 1 parameters incorporate 0% and 5% Steam Generator Tube Plugging (SGTP). The parameters for Byron Unit 2 and Braidwood Unit 2 are given in Table 2.1-2 and reflect the original Model D5 Steam Generators and 0% and 10% SGTP.

These performance capability parameters were used by Westinghouse in all the analytical efforts. Westinghouse performed the analyses and evaluations based on the parameter set or sets which were most limiting, so that the analyses would support operation of the Byron and Braidwood units over the range of conditions specified.

2.1.4 Acceptance Criteria for Determination of Parameters

The primary acceptance criteria for the determination of the PCWG parameters was that they would pose as few potential feasibility issues as possible for the uprate project from an analysis perspective, and that they provide Commonwealth Edison with adequate flexibility and margin in the operation of the plant.

2.1.5 Results/Conclusions

The resulting PCWG parameters have been evaluated throughout this document.

2.1.6 References

None.

Table 2.1-1
Design Power Capability Parameters
for Byron Unit 1 and Braidwood Unit 1

BASIC COMPONENTS			
Reactor Vessel, ID, in.	173	Isolation Valves	Yes
Core		Number of Loops	4
Number of Assemblies	193	Steam Generator	
Rod Array	17x17 V5(1)	Model	BWI
Rod OD, in.	.360	Shell Design Pressure, psia	1200
Number of Grids	6Z/2I/3IFM	Reactor Coolant Pump	
Active Fuel Length, in.	144	Model/Weir	93A/Yes
Number of Control Rods, FL	53	Pump Motor, hp	7000
Internals Type	CAE	Frequency, Hz	60

			UPRATING		
THERMAL DESIGN PARAMETERS	**Original(6)**	**Case 1**	**Case 2**	**Case 3**	**Case 4**
NSSS Power, %	100	105	105	105	105
MWt	3425	3600.6	3600.6	3600.6	3600.6
10^6 BTU/hr	11687	12286	12286	12286	12286
Reactor Power, MWt	3411	3586.6	3586.6	3586.6	3586.6
10^6 BTU/hr	11639	12238	12238	12238	12238
Thermal Design Flow, Loop gpm	94,400	92,000	92,000	92,000	92,000
Reactor 10^6 lb/hr	140.3	139.8	139.8	137.2	137.2
Reactor Coolant Pressure, psia	2250	2250	2250	2250	2250
Core Bypass, %	5.8	8.3(5)	8.3(5)	8.3(5)	8.3(5)
Reactor Coolant Temperature, °F					
Core Outlet	621.7	613.3	613.3	625.4	625.4
Vessel Outlet	618.4	608.0	608.0	620.3(2)	620.3(2)
Core Average	591.8	579.5	579.5	592.7	592.7
Vessel Average	588.4	575.0	575.0	588.0	588.0
Vessel/Core Inlet	558.4	542.0(2)	542.0(2)	555.7	555.7
Steam Generator Outlet	558.1	541.7	541.7	555.4	555.4
Steam Generator					
Steam Temperature, °F	543.3	533.5	532.8	547.5	546.8
Steam Pressure, psia	990	912(2,4)	907(2,4)	1024(2,4)	1019(2,4)
Steam Flow, 10^6 lb/hr total	15.13	15.98	15.98	16.07	16.06
Feed Temperature, °F	440.0	446.6	446.6	446.6	446.6
Moisture, % max.	0.25	0.10	0.10	0.10	0.10
Apparent FF, hr.sq.ft.°F/BTU	0.00005	0.00005	0.00005	0.00005	0.00005
Tube Plugging, %	0	0	5	0	5
Zero Load Temperature, °F	557.0	557.0	557.0	557.0	557.0
HYDRAULIC DESIGN PARAMETERS					
Pump Design Point, Flow (gpm)/Head (ft.)	100,000/290				
Mechanical Design Flow, gpm/loop	104,000	107,000(3)	107,000(3)	107,000(3)	107,000(3)
Minimum Measured Flow, gpm total	390,400	380,900(7)	380,900(7)	380,900(7)	380,900(7)

FOOTNOTES:

1. Vantage 5 fuel features: IFBA, AB, IFMs, DFBN, Protective Bottom Grid.
2. Plant operation limited to a minimum steam pressure of 827 psia, maximum T_{hot} of 618.4°F, and minimum T_{cold} of 538.2°F.
3. Mechanical Design Flow is being increased as part of the Uprate Program.
4. 13.5 psi SG internal ΔP.
5. Core bypass increased by 2% for thimble plug removal and .5% for IFMs.
6. N-loop parameters based on original Model D4 steam generators.
7. MMF based on TDF of 92,000 gpm/loop and 3.5% flow measurement uncertainty.

Table 2.1-2
Design Power Capability Parameters
for Byron Unit 2 and Braidwood Unit 2

BASIC COMPONENTS					
Reactor Vessel, ID, in.	173		Isolation Valves		Yes
Core			Number of Loops		4
Number of Assemblies	193		Steam Generator		
Rod Array	17x17 V5(2)		Model		D5
Rod OD, in.	.360		Shell Design Pressure, psia		1200
Number of Grids	6Z/2I/3IFM		Reactor Coolant Pump		
Active Fuel Length, in.	144		Model/Weir		93A/Yes
Number of Control Rods, FL	53		Pump Motor, hp		7000
Internals Type	CAE		Frequency, Hz		60
			UPRATING		
THERMAL DESIGN PARAMETERS	**Original**	**Case 1**	**Case 2**	**Case 3**	**Case 4**
NSSS Power, %	100	105	105	105	105
MWt	3425	3600.6	3600.6	3600.6	3600.6
10^6 BTU/hr	11687	12286	12286	12286	12286
Reactor Power, MWt	3411	3586.6	3586.6	3586.6	3586.6
10^6 BTU/hr	11639	12238	12238	12238	12238
Thermal Design Flow, Loop gpm	94,400	92,000	92,000	92,000	92,000
Reactor 10^6 lb/hr	140.3	139.8	139.8	137.2	137.2
Reactor Coolant Pressure, psia	2250	2250	2250	2250	2250
Core Bypass, %	5.8	8.3(5)	8.3(5)	8.3(5)	8.3(5)
Reactor Coolant Temperature, °F					
Core Outlet	621.7	613.3	613.3	625.4	625.4
Vessel Outlet	618.4(1)	608.0	608.0	620.3(3)	620.3(3)
Core Average	591.8	579.5	579.5	592.7	592.7
Vessel Average	588.4	575.0	575.0	588.0	588.0
Vessel/Core Inlet	558.4	542.0(3)	542.0(3)	555.7	555.7
Steam Generator Outlet	558.1	541.7	541.7	555.4	555.4
Steam Generator					
Steam Temperature, °F	543.3	523.6	519.8	538.8	534.9
Steam Pressure, psia	990	838(3)	811(3)	953(3)	923(3)
Steam Flow, 10^6 lb/hr total	15.13	15.96	15.95	16.04	16.02
Feed Temperature, °F	440.0	446.6	446.6	446.6	446.6
Moisture, % max.	0.25	0.25	0.25	0.25	0.25
Apparent FF, hr.sq.ft.°F/BTU	0.00005	0.00005	0.00005	0.00005	0.00005
Tube Plugging, %	0	0	10	0	10
Zero Load Temperature, °F	557.0	557.0	557.0	557.0	557.0
HYDRAULIC DESIGN PARAMETERS					
Pump Design Point, Flow (gpm)/Head (ft.)	100,000/290				
Mechanical Design Flow, gpm/loop	104,000	107,000(4)	107,000(4)	107,000(4)	107,000(4)
Minimum Measured Flow, gpm total	390,400	380,900(6)	380,900(6)	380,900(6)	380,900(6)

FOOTNOTES:

1. T_{hot} 0.6°F higher than nominal to offset 90/10 feedwater flow split.
2. Vantage 5 fuel features: IFBA, AB, IFMs, DFBN, Protective Bottom Grid.
3. Plant operation limited to a minimum steam pressure of 827 psia, maximum T_{hot} of 618.4°F, and minimum T_{cold} of 538.2°F.
4. Mechanical Design Flow is being increased as part of the Uprate Program
5. Core bypass increased by 2% for thimble plug removal and .5% for IFMs.
6. MMF based on TDF of 92,000 gpm/loop and 3.5% flow measurement uncertainty.

3.0 NSSS DESIGN TRANSIENTS

This chapter discusses the generation of NSSS and Auxiliary Equipment Design Transients for the uprated power conditions. Current NSSS design transients were analyzed for their continued applicability at uprated power and the resulting transient curves were provided to all system and component designers for use in their specific analyses. Section 3.1 describes the evaluation performed. Auxiliary equipment design transients were also evaluated to determine whether they remain applicable for use in the uprating analysis of all the auxiliary equipment in the NSSS. The results of this evaluation are presented in Section 3.2.

3.1 NSSS Design Transients

3.1.1 Introduction and Background

As part of the original design and analyses of the NSSS components for the Byron/Braidwood Plants, NSSS design transients (i.e., temperature and pressure transients) were specified for use in the analyses of the cyclic behavior of the NSSS components. To provide the necessary high degree of integrity for the NSSS components, the transient parameters selected for component fatigue analyses were based on conservative estimates of the magnitude and frequency of the temperature and pressure transients resulting from various plant operating conditions. The transients selected for use in component fatigue analyses were representative of operating conditions which would be considered to occur during plant operations and were considered to be sufficiently severe or frequent to be of possible significance to component cyclic behavior. The transients were selected to be conservative representations of transients which, when used as a basis for component fatigue analysis, would provide confidence that the component was appropriate for its application over the operating license period of the plant. For purposes of analysis, the number of transient occurrences were based on an operating license period of 40 years.

3.1.2 Input Parameters and Assumptions

NSSS design transients are based primarily on the NSSS Performance Capability Working Group (PCWG) parameters as discussed in Chapter 2 of this report. The NSSS PCWG parameters upon which the original NSSS design transients were based were compared to the NSSS PCWG parameters for power uprate and shown to be different in only a few instances such as steam pressure and feedwater temperature. These differences are sufficient to reassess the original NSSS design transients and to require, if necessary, that revised NSSS design transients be specified for power uprate.

3.1.3 Description of Analyses/Evaluations

The PCWG parameters for the original and proposed uprate parameters were compared and the results were that except for one instance (i.e., loss of load transient), the current design transients remain applicable. These B/B specific design transients have been used in the NSSS component and fatigue analyses and evaluations presented in Chapter 5 of this report.

3.1.4 Acceptance Criteria

There are no specific acceptance criteria. See Section 5.0 for component criteria.

3.1.5 Results and Conclusions

There are no specific results or conclusions for this section. See Section 5.0 for component results and conclusions.

3.2 Auxiliary Equipment Design Transients

3.2.1 Introduction and Background

As part of the original design and analyses of the NSSS auxiliary components (i.e., NSSS auxiliary pumps, valves, and heat exchangers) for Byron/Braidwood, auxiliary equipment design transients (i.e., temperature and pressure transients) were specified for use in the analyses of the cyclic behavior of the NSSS auxiliary components. To provide the necessary high degree of integrity for the NSSS auxiliary components, the transient parameters selected for component fatigue analyses were based on conservative estimates of the magnitude and frequency of the temperature and pressure transients resulting from various plant operating conditions. The transients selected for use in component fatigue analyses are representative of operating conditions which would be considered to occur during plant operations and are considered to be sufficiently severe or frequent to be of possible significance to component cyclic behavior. The transients were selected to be conservative representations of transients which, when used as a basis for component fatigue analysis, would provide confidence that the component was appropriate for its application over the operating license period of the plant. For purposes of analysis, the number of transient occurrences were based on an operating license period of 40 years.

3.2.2 Input Parameters and Assumptions

The review of the NSSS auxiliary equipment design transients was based on a comparison between the NSSS PCWG parameters for power uprate as discussed in Chapter 2 of this report and the parameters which make up the original auxiliary equipment design transients.

3.2.3 Description of Analyses/Evaluations

A review of the current auxiliary equipment transients determined that the only transients that could be potentially impacted by the uprating are those temperature transients that are impacted by full load NSSS operating temperatures, namely T_{hot} and T_{cold}. These transients are currently based on an assumed full load NSSS worst case T_{hot} of 630°F and worst case T_{cold} of 560°F. These NSSS temperatures were originally selected to ensure that the resulting design transients would be conservative for a wide range of NSSS operating temperatures.

3.2.4 Acceptance Criteria

There are no specific acceptance criteria for this section. See Section 5.0 for component acceptance criteria.

3.2.5 Results and Conclusions

Since the PCWG parameter ranges for T_{hot} (608.0° - 620.3°F) and T_{cold} (542.0° - 555.7°F), are less limiting than the temperature ranges which established the current auxiliary equipment design transients, it is concluded that the actual temperature transients (that is, the change in temperature from T_{hot} or T_{cold} dictated by the uprated parameters to a lower auxiliary system related temperature or vice versa) are less severe than the current design temperature transients, and these design transients remain applicable at uprated power.

3.2.6 References

None.

4.0 NSSS SYSTEMS

This chapter describes the results of the efforts performed in the NSSS systems area to support the uprating. Evaluations and analyses were performed to confirm that the NSSS systems continue to perform their intended functions under the uprated conditions. The systems addressed in this chapter are as follows:

Fluid Systems:

- Reactor Coolant System
- Chemical and Volume Control System
- Residual Heat Removal System
- Emergency Core Cooling System
- Boron Thermal Regeneration System
- Component Cooling Water System
- Boron Recycle System
- Sampling System
- Waste Processing System

NSSS/BOP Interface Systems:

- Main Steam System
- Steam Dump System
- Condensate Feedwater System
- Auxiliary Feedwater System
- Steam Generator Blowdown System

NSSS Control Systems:

- Pressure Relief Component Sizing
- Control Systems Setpoints Analysis
- Low Temperature Overpressure Protection System Setpoint Analysis

Detailed results and conclusions are presented within each subsection of this chapter.

4.1 NSSS Fluid Systems

4.1.1 Reactor Coolant System

4.1.1.1 Introduction

The Reactor Coolant System (RCS) consists of four heat transfer loops connected in parallel to the reactor vessel. Each loop contains a Reactor Coolant Pump (RCP), which circulates the water through the loops and reactor vessel, and a Steam Generator (SG), where heat is transferred to the Main Steam (MS) System. In addition, the RCS contains a pressurizer which controls RCS pressure through electrical heaters, water sprays, power-operated relief valves (PORVs), and spring-loaded safety/relief valves. The steam discharged from the PORVs and safety/relief valves flows through interconnecting piping to the Pressurizer Relief Tank (PRT).

This section identifies the key functions of the RCS and which functions are potentially impacted by the uprate project. The key RCS functions are as follows:

1. The RCS transfers heat generated in the reactor core to the MS System via the SGs.

2. When the core is subcritical and RCS temperature is below approximately 350°F, the RCS provides means to transfer decay and sensible heat to the Residual Heat Removal (RHR) System.

3. The RCS fluid acts as a moderator of neutrons by slowing the neutrons to lower thermal energy states and increasing the probability of thermal fission.

4. The RCS fluid is a solvent and carrier of boric acid which is used as a neutron poison. Neutrons are absorbed by boron, which decreases the probability of thermal fission.

5. The RCS is the second of three barriers against fission product release to the environment. (The fuel cladding is the first barrier, and the containment building is the third.)

6. The RCS provides means for pressure control via use of pressurizer heaters and spray flow.

The calculated RCS design operating conditions at the uprated power conditions are presented in Chapter 2 of this report. The primary changes in PCWG parameters impacting the RCS functions include the increase in core power, the allowable operating range for average RCS temperature (T_{avg}), and reduced Thermal Design Flow (TDF). The potential impact of the uprated conditions on the RCS functions are described below.

1. The core power increase will affect the total amount of heat transferred to the MS System. Verification that the major components can support this increased heat removal function is addressed in Section 4.2 of this report.

2. During the second phase of plant cooldown, the RHR System will be required to remove larger amounts of decay heat from the RCS. Section 4.1.3 of this report addresses the RHR System cooldown capability at uprated conditions.

3. The increased thermal power can change the RCS transient response to normal and postulated design basis events. The acceptability of the RCS, with respect to control and protection functions, is addressed in Section 4.3 of this report.

4. With reduced TDF, RCS loop flows can decrease. The reduction in RCS loop flows can reduce pressurizer spray flow capability since loop velocity head is used for driving head. In addition, a range of steady-state full-power RCS operating temperatures is established. This range can cause changes in nominal pressurizer level, which can change the steam release potential to the PRT. The "systems" impact of these changes is discussed within this section.

5. Changes in Best Estimate (BE) RCS flows will affect the Low Temperature Overpressure Protection (LTOP) System analysis (Section 4.3). The LTOP System setpoints are dependent upon the pressure differential between the core midplane (pressure of

interest) and the pressure measured by the RCS wide-range pressure instrumentation. This pressure differential is primarily a function of RCS BE flows at cold conditions.

6. Changes in BE RCS flows at 100% power impact RCP horsepower and therefore heat input to the RCS. All parameters affecting net heat input to the RCS were reviewed and revised, as applicable, to reflect the uprated operating parameters at all Byron/Braidwood Units.

4.1.1.2 Input Parameters and Assumptions

As noted earlier, the general acceptability of the changes to RCS operating conditions is justified by acceptable plant transient and safety analysis results, which are discussed in Chapter 6 of this report.

For the uprate project, various "systems" assessments were performed. Key input parameters used in the "systems" assessments are listed below.

1. Reduced TDF reduces available loop flows. For RCS loops used for pressurizer spray flow, lower RCS flows reduce the available driving head for spray. To support RCS transient response and plant safety analyses, a range of pressurizer spray flows, with full spray valve operation, was calculated. For these calculations, the piping layouts for Byron/Braidwood Units 1 and 2 were used along with pressurizer spray valve hydraulic performance.

2. The range of RCS operating temperatures provided in Chapter 2 of this report was used as a basis to evaluate RCS design temperatures.

3. The Byron/Braidwood-specific equipment elevations and piping layouts were used for the LTOP ΔP analysis. RCS conditions of 500 psia, 70°F were assumed for consistency with previous LTOP analyses. In addition, 0% power applies, since LTOP is activated for Modes 4-6. No steam generator tube plugging is assumed, which maximizes RCS flow and therefore RCS ΔP. Finally, RCS BE flows are used. This is more conservative (higher) than thermal design flow, which maximizes ΔP.

4. Operation at the lower range of RCS T_{avg} conditions increases the available pressurizer steam space volume that may have to be condensed in the PRT under limiting RCS

transient conditions (e.g., loss of load event). The existing Westinghouse PRT design basis sizing calculation was used as a basis for this evaluation.

5. No steam generator tube plugging is assumed for determining the net heat input. This maximizes RCP flow and heat input. Plant-specific actual operating parameters, such as letdown, seal injection, charging flows and temperatures, pressurizer spray flow and temperature, and surge line temperature were used as input to the analysis.

4.1.1.3 Description of Analysis/Evaluation

To evaluate pressurizer spray flow capability, the existing calculations, which defined expected minimum pressurizer spray flow, were reviewed and design inputs were compared with the corresponding uprated values. The available ΔP due to RCS flow between the spray line connections on the cold legs and the pressurizer was compared to the actual pressure drop required to provide 900 gpm total spray flow based on system resistances and elevation head.

In the assessment of system operation at the higher RCS T_{avg} condition, the maximum expected RCS T_{hot} temperature (provided in Chapter 2) was compared to RCS design temperatures. In the assessment of system operation at the lower RCS T_{avg} condition, the available steam space volume in the pressurizer was compared to that assumed in the PRT design basis calculation to assess available margin.

Pressure drop from the core midplane to the RCS wide-range pressure instrumentation is determined by calculating ΔP due to flow from the core through the hot leg to the pressure transmitter in the RHR suction piping. In addition, elevation head between the core and transmitter is considered. Finally, the effects of RHR flow on RCS flow which affects the ΔP due to piping and equipment resistance, are estimated.

The net heat input analysis is basically a detailed heat balance on the RCS. Convective heat losses from RCS equipment and enthalpy losses due to letdown and pressurizer spray are subtracted from heat inputs from the RCPs, and enthalpy inputs from charging, seal injection, and surge line flow, to determine the net heat input.

4.1.1.4 Acceptance Criteria

The design basis pressurizer spray flow (total) was established at 900 gpm. The calculated flow (considering RCS process conditions and reduced RCS loop flow) should be at or above this value.

In the assessment of system operation at the higher range of RCS T_{avg}, the maximum expected RCS T_{hot} must be less than or equal to the applicable RCS design temperature. This ensures pressure boundary integrity.

In the assessment of the PRT relief capability, the desirable acceptance criteria is "successful" operation following a maximum expected pressurizer discharge condition. The PRT nominal liquid and gas volumes specified for full power operation are inherently based on the following Westinghouse PRT design criteria:

1. The PRT initial water volume was selected to limit final water temperature (following a steam discharge) to 200°F. This is the maximum allowable temperature for discharge to the Liquid Waste Disposal System without external cooling.

2. The PRT initial gas volume was selected to limit maximum internal pressure (following a steam discharge) to less than one-half of the tank design pressure. This ensures that the PRT rupture disks (provided for tank overpressure protection) will not rupture.

These design criteria continue to be met under uprated power conditons.

There is no acceptance criteria for the ΔP between the core midplane and the wide-range pressure instrumentation at cold (70°F) conditions.

Net heat input must be at least 14 MWt per unit to support assumptions made in Chapter 2.

4.1.1.5 Results

The calculated available and required ΔP differed by less than 1 psi at 900 gpm total pressurizer spray flow. This difference is judged to be negligible.

The maximum expected RCS Hot Leg (T_{hot}) temperature at uprated conditions is 620.3°F. This temperature is well within the RCS loop design temperature of 650°F.

The LTOP ΔP calculated values were used as input to the LTOP evaluation. Results of the LTOP ΔP calculation are summarized in Table 4.1.1-1.

Table 4.1.1-1
LTOP ΔP Calculation Results

Case	No. RCPs Running	No. RHR Trains	RCS Loops Active	ΔP Core to Press. Trans. (psi)
1	4	0	1,2,3,4	69.25
2	4	1	1,2,3,4	70.15
3	4	2	1,2,3,4	70.94
4	3	0	1,2,3	49.04
5	3	1	1,2,3	50.02
6	3	2	1,2,3	50.63
7	2	0	1,2	35.06
8	2	1	1,2	36.07
9	2	2	1,2	36.83
10	1	0	1	27.29
11	1	1	1	28.74

No changes were made to the current nominal pressurizer level full-power control setpoint of 60%. The PRT level setpoint calculation of record was reviewed to confirm acceptability for the uprating. It was determined that the basis for the PRT level setpoints was conservative for the Byron/Braidwood tank volume. Therefore, new setpoints were calculated based on actual tank volume of 1800 ft^3:

PRT Level Setpoints	Current	Revised
High Alarm (%)	80	88
Low Alarm (%)	69	59

The net heat input results for the Byron/Braidwood Units at uprated conditions are:

Net Heat Input

Unit	MBtu/hr.	MWt
Byron 1	56.90	16.65
Byron 2	56.45	16.53
Braidwood 1	51.29	15.03
Braidwood 2	54.49	15.91

4.1.1.6 Conclusions

1. There is no impact on pressurizer spray flow capacity due to the uprating.

2. The RCS equipment design parameters bound those of the uprating.

3. Since the current PRT level setpoints bound the new setpoints, no changes are required for the uprating. However, use of the revised setpoints will provide additional operating margin.

4. The minimum requirement of 14 MWt per unit is satisfied for net heat input.

4.1.1.7 References

None.

4.1.2 Chemical and Volume Control System

4.1.2.1 Introduction

The RCS fluid interfaces with the Chemical and Volume Control System (CVCS) are the regenerative and excess letdown heat exchangers, RCP seal injection and RCP seal leakoff. Design and operating conditions of the heat exchangers (HX) were reviewed to ensure that the uprating conditions are bounded by the original HX design and operating conditions.

4.1.2.2 Input Parameters and Assumptions

The regenerative and excess letdown HXs take fluid from the RCS cold leg at temperatures of 558.1°F (original) and 541.7 to 555.4°F (uprating). These are bounding design temperatures.

4.1.2.3 Description of Analysis/Evaluation

Regenerative HX

The letdown (shell side) design temperature of 650°F and operating temperature of 560°F both bound the uprating conditions. Charging flow and temperature remain the same at uprating conditions, since the letdown temperature and pressure control systems remain unchanged. Since the bounding design inlet (letdown) temperature at uprating (555.4°F) is lower than the operating temperature from the HX specification sheet (560°F), the outlet temperature will be less than the specification sheet outlet temperature (288.7°F) because cooling flow and temperature (charging) do not change. This results in a lower inlet temperature to the letdown HX.

Letdown HX

The letdown (tube side) design temperature of 400°F exceeds the original operating inlet temperature of 288.7°F, which bounds the uprating condition. The letdown outlet temperature is controlled by an instrument which adjusts Component Cooling Water (CC) flow to the HX. Since the outlet temperature control setpoint has not changed and the inlet temperature under uprated conditions is less than the original inlet temperature, slightly less CC will be required.

Excess Letdown HX

The letdown (tube side) design temperature of 650°F bounds the uprating bounding design conditions of 541.7 to 555.4°F.

The operating inlet temperature is 560°F (same as regenerative HX). For calculational purposes, CC temperature and flow were fixed; therefore, a bounding design inlet temperature (555.4°F) less than 560°F will result in a lower outlet temperature (less than the 190°F from the HX specification sheet). This fluid enters the seal water HX, then the charging pump suction, and is used primarily for seal injection when normal letdown flow is not available. Since seal injection temperature can be maintained in the desired operating band of 60°F – 130°F, there is no adverse affect on the RCP seals

RCP Seal Injection & Seal Return

Power uprate has no effect on these systems.

4.1.2.4 Acceptance Criteria

The original CVCS HX design and operating parameters must bound those of the uprating.

4.1.2.5 Results/Conclusions

The original CVCS HX design and operating parameters bound those of the uprating. There is, therefore, no adverse effect on the CVCS system design and operation due to the uprating.

4.1.2.6 References

None.

4.1.3 Residual Heat Removal System

4.1.3.1 Introduction

The Residual Heat Removal (RHR) System is a dual function system. During normal power operation, the system is in stand-by mode to support its Engineered Safeguards function (i.e., safety injection). During the second phase of plant cooldown and during plant shutdown mode the RHR System is used to remove RCS sensible heat and core decay heat. The auxiliary feedwater and main steam systems are used for RCS heat removal during the first phase of plant cooldown and may supplement the second phase of plant cooldown. This section discusses the RHR System normal functions (i.e., heat removal). The Engineered Safeguards function of the RHR System (safety injection) is discussed in Section 4.1.4.

The RHR System is comprised of two centrifugal pumps, two heat exchangers, interconnecting piping and instrumentation. With the RHR System in operation, each RHR pump takes suction from an RCS hot leg and recirculates the flow back to the RCS cold legs. System flow passes through the tube side of the RHR heat exchangers (shell & tube design). Cooling flow to the

RHR heat exchangers (shell side) is provided via the Component Cooling Water (CC) System, which in turn, is cooled by the Essential Service Water (SX) System. The CC System is comprised of five pumps and three heat exchangers. Two CC pumps and two CC heat exchangers are available to support the unit shutting down. Another CC heat exchanger and CC pump are available to support the operating unit.

The maximum heat removal demand on the RHR System occurs during plant cooldown when RCS sensible heat (e.g., metal mass), core decay heat, heat input from one or more Reactor Coolant Pumps (RCPs), and heat for the cooldown must all be removed from the RCS. In addition, operating restrictions are imposed on the maximum allowable CC System temperature and flow during cooldown, which can limit the RHR System heat removal rate.

The overall RHR System heat removal capability can vary significantly depending on system equipment availability, CC system equipment availability, CC system flows, and SX system inlet temperature. In general, RHR System heat removal capability becomes more restricted when operating conditions change as outlined below.

- Higher RCS heat loads
- Lower RHR System flows
- Lower CC flow to the RHR System heat exchanger
- Lower SX flow to the CC System heat exchanger
- Higher CC System auxiliary heat loads

4.1.3.2 Input Parameters and Assumptions

Table 4.1.3-1
Cooldown Input Parameters

Parameter	2-Train	1-Train
Reactor Power (MWt)	3586.6	3586.6
CC HX UA (MBtu/hr/°F) per HX	4.79	4.79
RHR HX UA (MBtu/hr/°F) per HX	2.17	2.17

Table 4.1.3-1 (Cont.) Cooldown Input Parameters		
Parameter	**2-Train**	**1-Train**
Maximum SX Temperature (°F)	100	100
RCS Heat Capacity (MBtu/°F)	2.262	2.262
Auxiliary Heat Load @4 Hr. after shutdown (MBtu/hr.) *	9.998	10.173
Auxiliary Heat Load @ end of cooldown (MBtu/hr.) *	7.238	8.963
RCP Heat @ 350°F RCS temperature (MBtu/hr.)	19.21	19.21
Number of RHR HX in service	2	1
Number of CC HX in service	2	1
RCP Stop Temperature (°F)	160	160
Maximum CC Temperature (°F)	120	120
Maximum Cooldown Rate (°F/hr.)	50	50
RHR Initiation Time after shutdown (hr.)	4	4
Initial RCS Temperature (°F)	350	350
Final RCS Temperature (°F)	140	200
Spent Fuel Pool HX Heat Load (MBtu/hr.) Cases 1 and 3	0	0
Spent Fuel Pool HX Heat Load (MBtu/hr.) Case 2	19.3	NA

* Excluding Spent Fuel Pool HX heat load

4.1.3.3 Description of Analysis/Evaluation

Of the specified changes in RCS operating conditions addressed by this project, only the increase in reactor core power level has a significant effect on RHR System thermal performance capability. Specifically, higher core power levels will increase RCS decay heat loads, which must be removed during plant cooldown and shutdown conditions. As such, detailed thermal analyses were performed. From a hydraulic (flow) perspective, the revised RCS operating conditions have no direct impact on the flow delivery capability of the RHR System. As such, no hydraulic evaluations were performed. Likewise, existing instrumentation and controls are independent of uprated conditions and were not evaluated.

RHR System thermal performance was calculated for each of the following three cooldown scenarios:

1. The ability of the RHR System to accept the RCS heat removal function during the second phase of plant cooldown (i.e., RHR System Cut-In).

2. The ability of the RHR System to cool down the RCS, with all equipment operating, to a cold shutdown condition (200°F) and a refueling condition (140°F). Note, RHR System operation with all equipment available (including support systems) is referred to as a "normal" plant cooldown within the context of this section.

3. The ability of the RHR System to cool down the RCS under limiting equipment availability to a cold shutdown condition (200°F). Note, RHR System operation with one subsystem of equipment available (including support systems) is referred to as a "single train" plant cooldown within the context of this section.

For scenario 2, the RHR System major components were originally sized to achieve a targeted (desired) overall cool down from system "Cut-In" (which occurs 4 hours after reactor shutdown) to a refueling RCS temperature (140°F). This "normal" cooldown (with all cooling equipment available) was reanalyzed at the higher core power level.

For scenario 3, the Byron/Braidwood UFSAR (Reference 1) Section 5.4.7.2.7 requires that the RHR System cool the RCS from 350°F at 4 hours after shutdown to 200°F by 72 hours after plant shutdown with one CC System train unavailable.

4.1.3.4 Acceptance Criteria

Per UFSAR Section 5.4.7.2.7 (Reference 1), the RHR System must cool the plant from 350°F to 200°F in 72 hours with a limiting single failure. This is assumed as loss of one CC pump/train. Since one CC pump cannot provide design flow to both RHR HXs, loss of one CC pump also eliminates one RHR cooling train.

4.1.3.5 Results/Conclusions

Results of the analysis/evaluation are listed in Table 4.1.3-2.

Table 4.1.3-2 Cooldown Analysis/Evaluation Results				
Case	**Trains**	**SFP Heat Load (MBtu/hr.)**	**Final RCS Temperature (°F)**	**Cooldown Time (hr.)**
1	2	0	140	39.9
2	2	19.3	140	43.6
3	1	0	200	47.6

The results in Table 4.1.3-2 show that single train cooldown for the uprating can be achieved within 72 hours after shutdown for case 3 and thus meets the acceptance criteria. Case 3 is supported by the CC System design basis WCAP-12232 (Reference 2). The design basis normal cooldown for the uprating is defined by cases 1 and 2.

4.1.3.6 References

1. "Byron & Braidwood Station, Updated Final Safety Analysis Report," Revision 7, Docket Nos. STN-454/455/456/457, as amended through December 1998.

2. WCAP-12232, "Commonwealth Edison Company Byron/Braidwood Plants Component Cooling Water System Design Basis Document," Revision. 0.

4.1.4 Emergency Core Cooling System

The Safety Injection System (SIS) is an Engineered Safeguards System that is used to mitigate the effects of postulated design basis events. The basic functions of this system include providing short and long-term core cooling and maintaining core shutdown reactivity margin. The SIS is generally referred to as the Emergency Core Cooling System (ECCS).

At Byron and Braidwood Units 1 and 2, the ECCS is comprised of four subsystems. The first is a passive portion, comprised of four accumulator vessels with one connected to each of the RCS cold leg pipes. Each accumulator contains borated water under pressure (nitrogen cover gas). The borated water automatically injects into the RCS when the pressure in the RCS drops below the operating pressure of each of the accumulators.

The active portion of the ECCS is comprised of three subsystems, all of which automatically start following the generation of a Safety Injection (SI) signal. High head safety injection flow is delivered by the two centrifugal charging (CV) pumps as soon as the RCS pressure falls below the "cut-in" pressure (CV pump shut off head is adjusted for the effects of pump miniflow, RCP seal injection flow, and system resistance).

Intermediate head safety injection flow is delivered by the two safety injection (SI) pumps when the RCS pressure falls below the SI pump "cut-in" pressure (SI pump shut off head is adjusted for the effects of miniflow and system resistance).

Low head safety injection flow is provided by the two residual heat removal (RHR) pumps when RCS pressure falls below the RHR pump "cut-in" pressure (RHR pump shut off head is adjusted for the effects of miniflow and system resistance).

As the design basis event proceeds, the Refueling Water Storage Tank (RWST) borated water inventory decreases as water is transferred to the RCS and/or containment building. Upon depletion of sufficient RWST inventory on the affected Unit, the suction of operating CV, SI and RHR pumps are required to be realigned to cold leg recirculation mode. In this mode, the RHR pumps are realigned to take suction from the containment sump and the CV and SI pumps are realigned to take suction from the RHR pumps. Long-term core cooling is provided by the RHR system heat exchangers.

In an immediate response to a design basis event, the ECCS is designed to perform its safety functions despite the assumptions of a loss of offsite power and a limiting single active failure. An example of a postulated single failure is the failure of an emergency diesel generator to start. This failure would result in one train of ECCS being inoperable (one CV, SI and RH pump). In the long-term, a postulated failure would not prevent adequate core cooling, since one train of ECCS is sufficient for long-term core cooling. Several alternate flow paths are available for delivery of flow in the long-term.

4.1.4.1 Input Parameters and Assumptions

In general, the specified changes in RCS operating conditions due to thermal uprating (e.g., higher core power, hot full power T_{avg} range) have no direct effect on the overall performance capability of the ECCS. These systems will continue to provide flow performance (minimum and maximum) as determined by ECCS system parameters (e.g., pump performance and system resistance) and boundary conditions (e.g., RCS and containment pressure).

The acceptability of a given range of ECCS performance is justified by acceptable plant safety analysis results. For this project, the plant safety analyses were reanalyzed and evaluated. The calculation of ECCS flow used bounding plant performance data to calculate system resistance, which was then used to calculate the flow delivered to the RCS under accident conditions. It includes the effects of miniflow, seal injection flow, suction source, RCS and containment pressure, instrument uncertainties, and future pump degradation or enhancement.

4.1.4.2 Results

Different cases of ECCS flow rates, various RCS pressures, and different combinations of ECCS equipment were evaluated. The minimum safeguards ECCS flow rates (CV, SI and RHR pumps) for this report were calculated based on conservative pump degradation allowances. Maximum safeguards ECCS flow rates included a conservative allowance for pump enhancement. The ECCS flow rates versus RCS pressure were provided as input to the various plant safety analyses. All flow rates are applicable to all four Byron and Braidwood Units.

4.1.4.3 Conclusions

The acceptability of the calculated ECCS operating performance (flow delivered to the RCS versus RCS pressure) defined for this project is documented in the various discussions of individual plant safety analysis results as summarized in Chapter 6 of this report. All results show that the performance of the ECCS is sufficient to meet all plant safety analysis acceptance criteria.

4.1.5 Boron Thermal Regeneration System

This system is not used at Byron or Braidwood Station and; therefore, will not be discussed in this report.

4.1.6 Component Cooling Water System

4.1.6.1 Introduction and Evaluation

The Component Cooling Water (CC) System is designed to remove residual and sensible heat from the RCS via the RHR System during plant shutdown, cool heat exchangers in various NSSS systems, and provide cooling to Engineered Safeguards pumps after an accident.

The effect of uprating on the CC System is indirect, through the RHR System and spent fuel pool cooling system. As described in Section 4.1.3 (RHR System), the maximum operating temperature for the CC System (120°F) is assured by reducing RHR flow to the RHR HX during initial stages of cooldown when decay heat is maximum. Thus, the temperature design limits are maintained administratively.

The performance of components, such as the letdown and excess letdown HXs, which are cooled by CC, is not affected by the uprating because the RCS cold leg temperature at the uprating condition is lower than the current value. The fluid inlet temperature to the excess letdown HX and to the letdown HX (via the regenerative HX) is thus reduced such that the current HX design and operating parameters bound the requirements for uprating. Detailed evaluation of the CC System is contained in BOP Section 9.3.9.

4.1.6.2 Conclusions

There is no effect on the CC System design or equipment performance due to the uprating.

4.1.6.3 References

None.

4.1.7 Boron Recycle System

4.1.7.1 Introduction and Evaluation

The Boron Recycle System (BRS) is designed to accept and process all effluent which can readily be recycled to the RCS for reuse. It has no direct interface with the RCS, but receives letdown flow from the CVCS downstream of the letdown HX. The letdown HX outlet temperature instrumentation controls component cooling water flow to maintain a constant letdown outlet temperature of approximately 110°F.

The RCS cold leg temperature for the uprating is 541.7 to 555.4°F. This range is below the previous cold leg temperature of 558.1°F. Therefore, it can be expected that the letdown HX outlet temperature will be no higher than the current value. Since only a higher fluid temperature would cause a concern for the AB (i.e., reduced ion exchange capacity), there is no effect due to the uprating.

The volume of liquid to be processed by the AB is a function of plant operations, such as load follow, shutdowns, and startups. It is not a direct function of plant power rating.

4.1.7.2 Conclusions

There is no effect on the BRS system design or equipment performance due to the uprating.

4.1.7.3 References

None.

4.1.8 Sampling System

4.1.8.1 Introduction and Evaluation

The sampling system (PS) connections to the RCS include the pressurizer gas and liquid spaces, Loops 1 and 3 hot legs, and Loops 1-4 cold legs. Flow from these connections enters sample HXs prior to the sample sink. The RCS fluid temperatures are:

Table 4.1.8-1
Sampling System Temperatures

	Original	Uprating
Cold leg (°F)	558.1	541.7 to 555.4
Hot leg (°F)	618.4	608.0 to 620.3

The sample HX specification sheet shows a design temperature of 680°F and an operating inlet temperature of 653°F. Both of these values exceed all expected operating temperatures because they are based on pressurizer conditions, which exceed those of the RCS loops. Since the pressurizer conditions do not change for the uprating, the HXs remain adequate for all applications.

4.1.8.2 Conclusions

There is no impact on the PS system design or operation due to the uprating.

4.1.8.3 References

None.

4.1.9 Waste Processing Systems

4.1.9.1 Introduction and Evaluation

The liquid waste processing system collects, among other types of waste, reactor coolant leakage from piping and equipment. Plant operations and the size and number of leaks dictate the volume of liquid entering the system. The effect of the uprating is on the typical isotopic concentrations in the RCS, and therefore, the leakage. A slight increase in the total activity is expected, but it will not affect the manner in which the waste is treated.

Unlike liquid wastes, the volume of fission gasses generated by the core is proportional to power. Although this volume may increase due to the uprate, the actual volume is very small and will not directly affect waste gas system operation (nearly all the waste gas volume is nitrogen and hydrogen). For example, a 3600 MWt core generates approximately 1.7 ft^3/yr./Unit of Krypton and 10.6 ft^3/yr./Unit of Xenon. Very little of this volume actually reaches the RCS unless there is fuel leakage, and even less reaches the waste gas system. If it is assumed that the same volume of fission gasses enters the waste system, then the activity concentration and total curies will increase. This may require a slightly longer holdup time (a few days) for decay prior to release to the environment. The holdup is on the order of 45-60 days, which provides several half-lives of decay for Xe-133, the primary fission gas component. If the total number of curies is very low initially, there may be no need for additional holdup to meet discharge requirements on either activity concentration or total curies released.

4.1.9.2 Conclusions

There is no significant effect on the waste systems design or operation due to the uprating. A slight increase in activity concentration or total curies can be accomodated without affecting normal system operation.

4.1.9.3 References

None.

4.2 NSSS/BOP Fluid Systems Interfaces

4.2.1 Introduction and Background

As part of the Byron/Braidwood Units 1 & 2 Power Uprate Project, the following Balance-of-Plant (BOP) fluid systems were reviewed to assess compliance with Westinghouse Nuclear Steam Supply Systems (NSSS)/BOP interface guidelines (Reference 2):

- Main Steam System
- Steam Dump System
- Condensate and Feedwater System
- Auxiliary Feedwater System
- Steam Generator Blowdown System

The review was performed based on the range of NSSS operating parameters developed to support an NSSS power level of 3600.6 MWt (Section 2.0). The various interface systems were reviewed with the purpose of providing interface information which could be used in the more detailed Balance of Plant (BOP) analyses. The results of those analyses are provided later in the BOP sections of this report.

4.2.2 Input Parameters and Assumptions

A comparison of the power uprate Performance Capability Working Group (PCWG) parameters to the original PCWG parameters, previously evaluated for systems and components, indicates differences that could impact the performance of the above BOP systems. For example, the increase in NSSS power of approximately 5.1 percent (to 3600.6 MWt) and the upper limit on T_{avg} (588°F) would result in about a 6.2 percent increase in steam/feedwater mass flowrates. Additionally, the average steam generator tube plugging (SGTP) level of 10% in combination with the lower limit on T_{avg} (575°F) would result in a reduction in full-load steam pressure from 990 psia to 827 psia.

4.2.3 Description of Analyses, Acceptance Criteria and Results

Evaluations of the above BOP systems relative to compliance with Westinghouse NSSS/BOP interface guidelines were performed to address the parameters for power uprate analyses which include ranges for parameters such as T_{avg} (575° to 588°F) and steam generator tube plugging (0% to 10% average). These ranges on NSSS operating parameters result in ranges on BOP parameters such as steam generator outlet steam pressure (827 psia to 1024 psia). The NSSS/BOP interface evaluations were performed to address these ranges on NSSS and BOP parameters even though the Byron/Braidwood units are not expected to operate at the low end of the steam pressure range due to turbine volumetric flow limits. The results of the NSSS/BOP interface evaluations are delineated below.

4.2.3.1 Main Steam System

The uprating coupled with the potential reduction in full-load steam pressure to the average minimum value of 827 psia adversely impacts main steam line pressure drop. At the average minimum steam generator pressure of 827 psia, the full-load steam mass flowrate would increase about 5.4 percent. However, due to the reduced operating pressure and the lower-density steam, the volumetric flowrate would increase by approximately 28 percent and the steam line pressure drop would increase by approximately 35 percent.

Note that the original NSSS operating parameters for the NSSS power of 3425 MWt resulted in a steam line pressure drop of about 30 psi and a pressure of about 960 psia at the turbine inlet valves. Based on the range of NSSS operating parameters for the uprating to 3600.6 MWt, the lowest steam generator pressure would result in a pressure at the turbine inlet valves of approximately 786 psia.

The following summarizes the Westinghouse evaluation of the major steam system components relative to the power uprate conditions. The major components of the Main Steam (MS) System are the Steam Generator Main Steam Safety Valves (MSSVs), the SG Power Operated Relief Valves (PORVs), and the Main Steam Isolation Valves (MSIVs).

Steam Generator Main Steam Safety Valves

The setpoints of the MSSVs are determined based on the design pressure of the steam generators (1185 psig) and the requirements of the ASME Boiler and Pressure Vessel (B&PV) Code. Since the design pressure of the SGs has not changed with the power uprate, there is no need to revise the setpoints of the safety valves.

The MSSVs must have sufficient capacity so that main steam pressure does not exceed 110 percent of the steam generator shell-side design pressure (the maximum pressure allowed by the ASME B&PV Code) for the worst-case loss-of-heat-sink event (Reference 2, Section 15.2). Based on this requirement, Westinghouse applies the conservative criterion that the valves should be sized to relieve 105% of the maximum calculated steam flow at an accumulation pressure not exceeding 110% of the MS System design pressure (Reference 1).

Each Byron and Braidwood operating unit has twenty safety valves with a total capacity of 17.25×10^6 lb/hr, which provides about 107.3 percent of the maximum calculated steam flow of the 16.07×10^6 lb/hr for the uprating. Therefore, based on the range of NSSS parameters for the uprating, the capacity of the installed MSSVs meets the Westinghouse sizing criterion.

The original design requirements for the MSSVs (as well as the SG PORVs and steam dump valves) included a maximum flow limit per valve of 970,000 lb/hr at 1185 psig. Since the actual capacity of any single MSSV, SG PORVs or steam dump valve (926,000 lb/hr) is less than the maximum flow limit per valve, the maximum capacity criterion is satisfied.

Steam Generator Power Operated Relief Valves (PORVs)

The SG PORVs, which are located upstream of the main steam isolation valves (MSIVs) and adjacent to the MSSVs, are automatically controlled by steam line pressure during plant operations. The SG PORVs automatically modulate open and exhaust to atmosphere whenever the steam line pressure exceeds a predetermined setpoint to minimize safety valve lifting during steam pressure transients. As the steam line pressure decreases, the SG PORVs modulate closed and reseat at a pressure below the opening pressure. The SG PORV set pressure for these operations is between zero-load steam pressure and the setpoint of the lowest-set MSSVs. Since neither pressure changes for the proposed range of NSSS operating parameters, there is no need to change the SG PORV setpoint.

The primary function of the SG PORVs is to provide a means for decay heat removal and plant cooldown by discharging steam to the atmosphere when either the condenser, the condenser circulating water pumps, or steam dump to the condenser is not available. Under such circumstances, the SG PORVs, in conjunction with the Auxiliary Feedwater (AF) System, permit the plant to be cooled down from the pressure setpoint of the lowest-set MSSVs to the point where the Residual Heat Removal (RHR) System can be placed in service. During cooldown, the SG PORVs are either automatically or manually controlled. In automatic, each SG PORV Proportional plus Integral (P&I) controller compares steam line pressure to the pressure setpoint, which is manually set by the plant operator.

During a tube rupture event in conjunction with loss of offsite power, the SG PORVs are used to cool the RCS to a temperature that permits equalization of the primary and secondary pressures at a pressure below the lowest-set MSSVs. RCS cooldown and depressurization is required to preclude steam generator overfill and to terminate activity release to the atmosphere (Reference 2, Section 15.6.3).

The four SG PORVs are sized for a capacity of approximately 10 percent of the steam flow used for plant design, at no-load steam pressure. At uprated power, this capacity permits a plant cooldown to RHR System operating conditions in 4 hours (at an assumed cooldown rate of at least 50°F/hr) assuming 2 hours at hot standby. This sizing is compatible with normal cooldown capability and minimizes the water supply required by the AF System (see Section 4.2.3.4, Auxiliary Feedwater System). This is based on one train of auxiliary feedwater (AF) operating and flow going through all four SGs.

The design capacity of a single SG PORV is 415,250 lb/hr/valve (full open) at 1107 psia. Based on the range of parameters for the uprated power level, the total SG PORV design capacity, which is 1.661×10^6 lb/hr total at 1107 psia, is about 10.3 percent of the required maximum steam flow (16.07×10^6 lb/hr). Therefore the SG PORVs are adequate based on the range of conditions for power uprate.

Main Steam Isolation Valves and Main Steam Isolation Bypass Valves

The MSIVs are located outside the containment and downstream of the MSSVs. The valves function to prevent the uncontrolled blowdown of more than one steam generator and to minimize the RCS cooldown and containment pressure increase to within acceptable limits following a main steam line break. To accomplish this function, the original design requirements specified that the MSIVs must be capable of closure within 5 seconds of receipt of a closure signal against steam break flow conditions in either the forward or reverse direction.

Rapid closure of the MSIVs following postulated steam line breaks causes a significant differential pressure across the valve seats and a thrust load on the main steam system piping and piping supports in the area of the MSIVs. The worst cases for differential pressure increase and thrust loads are controlled by the steam line break area (i.e., mass flowrate and moisture content), throat area of the steam generator flow restrictors, valve seat bore, and no-load operating pressure. Since these variables and no-load operating pressure are not impacted by the uprating, the design loads and associated stresses resulting from rapid closure of the MSIVs will not change. Consequently, power uprate has no significant impact on the interface requirements for the MSIVs.

The MSIV bypass valves are used to warm up the main steam lines and equalize pressure across the MSIVs prior to opening the MSIVs. The MSIV bypass valves perform their function at no-load and low power conditions where power uprate has no significant impact on main steam conditions (e.g., steam flow and steam pressure). Consequently, power uprate has no significant impact on the interface requirements for the MSIV bypass valves.

4.2.3.2 Steam Dump System

The NSSS Reactor Control Systems and the associated equipment (pumps, valves, heaters, control rods, etc.) are designed to provide satisfactory operation (automatic in the range of 15 to 100 percent power) without a reactor trip when subjected to the following load transients:

- Loading at 5 percent of full power per minute with automatic reactor control.
- Unloading at 5 percent of full power per minute with automatic reactor control.

- Instantaneous load transients of plus or minus 10 percent of full power (not exceeding full power) with automatic reactor control.

- Load reductions of 50 percent of full power with automatic reactor control and steam dump.

The steam dump system creates an artificial steam load by dumping steam from before the turbine valves to the main condenser. The Westinghouse sizing criterion recommends that the steam dump system (valves and pipe) be capable of discharging 40 percent of the rated steam flow at full-load steam pressure to permit the NSSS to withstand an external load reduction of up to 50 percent of plant rated electrical load without a reactor trip. To prevent a trip, this transient requires all NSSS control systems to be in automatic, including the Reactor Control System, which accommodates 10% of the load reduction. A steam dump capacity of 40 percent of rated steam flow at full load steam pressure also prevents MSSV lifting following a reactor trip from full power.

Steam Dump System Major Components

Each Byron and Braidwood unit is provided with twelve condenser steam dump valves and each valve is specified to have a flow capacity of 816,337 lb/hr at a valve inlet pressure of 1107 psia. The total capacity for all twelve valves provides a steam dump capability of about 52.8 percent of the original maximum guaranteed steam flow (15.13×10^6 lb/hr), at a full load steam generator pressure of 990 psia. This exceeds the Westinghouse sizing criterion of 40 percent of rated steam flow by 12.8 percent flow.

NSSS operation within the range of operating parameters at lower steam generator pressures and higher steam flows will result in a reduced steam dump capability relative to the original Westinghouse sizing criterion. The Westinghouse evaluation for the uprate indicates that the total steam dump capacity could be as low as 41.5 percent of rated steam flow (15.95×10^6 lb/hr), or 6.620×10^6 lb/hr, at a full-load steam pressure equal to 827 psia. These operating conditions are based on an NSSS power level of 3600.6 MWt, an average SGTP level of 10%, and a T_{avg} in the lower end of the operating range (575°F). However, this reduced steam dump capability still exceeds the Westinghouse sizing criterion. Note at the upper end of the T_{avg} operating range and with a full-load steam generator pressure of 1019 psia, the

Westinghouse evaluation for the uprate indicates steam dump capacity is about 51.2 percent of rated flow (16.06 x 10^6 lb/hr), or 8.22 x 10^6 lb/hr.

The NSSS control systems analysis (Section 4.3) provides evaluations of the adequacy of the steam dump capacity and steam dump control system at the uprated conditions.

To provide effective control of flow on large step-load reductions or plant trips, the steam dump valves are required to operate from full-closed to full-open in 3 seconds at any pressure between 50 psi less than full load pressure and steam generator design pressure. The steam dump valves are also required to modulate to control flow. Positioning response may be slower with a maximum full stroke time of 20 seconds. These requirements are still applicable for the NSSS operating conditions for power uprate.

4.2.3.3 Condensate and Feedwater System

The Condensate and Feedwater (CD & FW) System must automatically maintain steam generator water levels during steady-state and transient operations. The range of NSSS PCWG parameters will result in a required feedwater volumetric flow increase of up to 6.8% during full-power operation. The higher feedwater flow and higher feedwater temperatures will have an impact on system pressure drop, which may increase by as much as 13.4 percent. Also, a comparison of the uprated PCWG parameters to the original PCWG parameters indicates that the SG full-power operating steam pressure may be decreased by as much as 163 psi (990 psia - 827 psia).

The main feedwater lines to the Byron/Braidwood Unit 2 pre-heater steam generators incorporate a Feedwater Bypass System (FBS). The functions of the FBS are to minimize the potential occurrence of water hammer in the steam generators, and to mitigate flow induced tube vibration in the steam generators. The FBS includes the feedwater bypass line, which connects the main feedwater line to the feedwater pre-heater bypass line outside containment. The feedwater pre-heater bypass line delivers flow to the steam generator auxiliary feedwater nozzles. The feedwater bypass line is designed to provide a flow-split ratio of approximately 90 percent to 10 percent between the main feedwater nozzle and the auxiliary feedwater nozzle at 100 percent of rated flow. This flow-split ratio was prescribed, based on the reference parameters, to preclude rapid wear of certain tubes in the preheater.

The proposed increase in NSSS power of approximately 5.1 percent (to 3600.6 MWt) and the upper limit on T_{avg} (588°F) would result in an approximate 6.2 percent increase in Unit 2 steam/feedwater mass flowrates. The bulk of the increased flow would enter the steam generator through the main feedwater nozzle and the remainder would enter through the auxiliary nozzle. An evaluation of pre-heater tube wear concluded that the increased feedwater flow rate would cause a modest increase in the rate of tube wear and the increase would be acceptable. Therefore, no action is recommended in terms of plant modifications to maintain the existing pre-heater flow limit. In light of this recommendation, the pre-heater high flow alarm set point will be adjusted based on the results of the tube wear study to maintain adequate margin between the alarm set point and normal full-load feedwater flow.

Based on the range of NSSS operating parameters approved for power uprate, the high flow rate alarm set point will be increased to a value no lower than 90 percent of maximum expected feedwater flow (3.61×10^6 lb/hr) to provide adequate margin for plant operations. This minimum alarm set point (3.61×10^6 lb/hr) permits a maximum combined instrumentation channel uncertainty of ±1.6 percent, since the pre-heater tube wear study concluded that a maximum pre-heater flow rate of 3.672×10^6 lb/hr is acceptable.

The major components of the CD & FW System are the Feedwater Isolation Valves, the Feedwater Control Valves, and the Condensate and Feedwater System Pumps.

Feedwater Isolation Valves/Feedwater Control Valves

The feedwater isolation valves (FIVs) are located outside containment and upstream of the feedwater control valves (FCVs). The valves function in conjunction with the primary isolation signals to the FCVs and backup trip signals to the feedwater pumps to provide redundant isolation of feedwater flow to the steam generators following a steam line break or a malfunction in the steam generator level control system. Isolation of feedwater flow is required to prevent containment overpressurization and excessive reactor coolant system cooldown. To accomplish this function, the FCVs and the backup FIVs must be capable of closure within 5 seconds, following receipt of any feedwater isolation signal.

The quick-closure requirements imposed on the FCVs and the backup FIVs causes dynamic pressure changes that may be of large magnitude and must be considered in the design of the valves and associated piping. The worst loads occur following a steam line break from no load conditions with the conservative assumption that all feedwater pumps are in service providing maximum flow following the break. Since these conservative assumptions are not impacted by the uprating, the design loads and associated stresses resulting from rapid closure of these valves will not change.

Condensate and Feedwater System Pumps

The CD & FW System available head, in conjunction with the FCV characteristics, must provide sufficient margin for feed control to ensure adequate flow to the steam generators during steady-state and transient operation. A continuous, steady feed flow should be maintained at all loads. To assure stable feedwater control, with variable speed feedwater pumps, the pressure drop across the FCVs at rated flow (100 percent power) should be approximately equal to the dynamic losses from the feed pump discharge through the steam generator (i.e., equal to the frictional resistance of feed piping, FIV, high pressure feedwater heaters, feed flow meter, and steam generator). In addition, adequate margin should be available in the FCVs at full load conditions to permit a CD & FW System delivery of 96 percent of rated flow with a 100 psi pressure increase above the full load pressure with the FCVs fully open (Reference 1).

The current turbine pump speed control programs were set to meet the above Westinghouse criterion based on the original PCWG parameters, and resulted in a valve lift of about 75 percent at full load with the pre-heat steam generators. This valve lift was determined to be compatible with the margin required to address the Westinghouse transient criterion, that is, provide 96 percent of rated flow with a 100 psi pressure increase above full load pressure with the FCVs fully open.

The system head losses between the feedwater header and the steam header will increase for the range of NSSS operating parameters approved for uprate. With the existing feedwater pump speed control programs, the lift of the FCVs at full power will increase by up to 13 percent. This increase in valve lift will decrease valve pressure drop and the margin required at full load to meet the Westinghouse transient criterion. Therefore, the feedwater pump speed controller will be re-calibrated to re-establish optimum valve lift (about 75 percent) at 100 percent load based on measured head loss data at the actual operating conditions. As conditions vary within the approved range of NSSS operating parameters over time due to tube plugging etc., it may be necessary to re-calibrate the pump speed controller to maintain optimum valve lift at full load.

To provide effective flow control during normal operation, the FCVs are required to stroke open or closed in 20 seconds over the anticipated inlet pressure control range (approximately 0-1600 psig). Additionally, rapid closure of the FCVs is required within five seconds after receipt of a trip close signal to mitigate certain transients and accidents. These requirements remain applicable at the uprated conditions (Reference 1). The NSSS control systems analysis (Section 4.3) provides an evaluation of the adequacy of feedwater control systems at the uprated conditions.

Further evaluation of the CD & FW System, including the feedwater and condensate pumps, is contained in the BOP Engineering Report.

4.2.3.4 Auxiliary Feedwater System

The Auxiliary Feedwater (AF) System supplies feedwater to the secondary side of the steam generators when the normal feedwater system is not available, thereby maintaining the steam generator heat sink. The system provides feedwater to the SGs during normal unit startup, hot standby, and cooldown operations, and also functions as an Engineered Safeguards System. In the latter function, the AF System is required to prevent core damage and system overpressurization during transients and accidents, such as a loss of normal feedwater or a secondary system pipe break. The minimum flow requirements of the AF System are dictated by accident analyses, and since the uprating impacts these analyses, evaluations of the limiting transients and accidents were performed to confirm that the AF System performance will be acceptable at the uprated conditions. Refer to Section 6.0 for a discussion of accident analyses.

<u>Auxiliary Feedwater Storage Requirements</u>

The AF System pumps are normally aligned to take suction from the condensate storage tank (CST). To fulfill the Engineered Safety Features (ESF) design functions, sufficient feedwater must be available during transient or accident conditions to enable the plant to be placed in a safe shutdown condition.

The limiting transient with respect to CST inventory requirements is the loss-of-offsite power (LOOP) transient. The Byron/Braidwood licensing basis specifies that in the event of a LOOP, sufficient CST useable inventory must be available to bring the unit from full power to hot standby conditions, maintain the plant at hot standby for 4 hours, and then cool the RCS to the residual heat removal system cut-in temperature (350°F) in 4 hours. In light of these design bases requirements, the Byron/Braidwood analysis-of-record concluded that the CST requires a minimum useable inventory of 200,000 gallons. Accordingly, the plant Technical Specifications for Units 1 and 2 ensure this minimum useable inventory is maintained.

Since the required CST inventory is a function of plant rated power and other NSSS operating parameters, a new analysis was performed as part of the BOP evaluations to determine the required inventory at uprated conditions. This new analysis for the LOOP scenario is based on the following conservative assumptions:

- Reactor trip occurs from 102 percent of rated core power (3586.6 MWt), from a low low water level in the steam generators. A two-second delay is assumed before the reactor trip following loss of offsite power.
- Steam is released from the steam generators at the first safety valve set point (including setting tolerance for drift and accumulation pressure).
- The steam generators are refilled to no-load programmed level.
- The CST operating fluid temperature is at the maximum value (i.e., 120°F for power uprate analyses).
- The plant has been at hot standby for four hours.
- The cooldown from hot standby temperature to RHR System cut-in temperature (RCS T_{hot} equal to 350°F) is accomplished in 4 hours.

The results of the Unit 1 and Unit 2 analysis (198,619 gallons and 197,666 gallons respectively) confirmed that a minimum useable inventory of 200,000 gallons is adequate to meet the plant design bases requirements for the range of NSSS operating conditions approved for plant uprate. Therefore, no change is required to the plant Technical Specifications covering CST inventory requirements for operation at the uprated conditions.

4.2.3.5 Steam Generator Blowdown System

The Steam Generator Blowdown System is used in conjunction with the Chemical Addition System to control the chemical composition of the steam generator shell water within specified limits. The blowdown system also controls the buildup of solids in the steam generator water.

The Byron/Braidwood steam generators and the SG blowdown system are designed to handle a maximum continuous blowdown rate of 90 gpm per steam generator. The actual blowdown flows required during plant operation are based on chemistry control and tube-sheet sweep requirements to control the buildup of solids. These required flow rates are not expected to be significantly impacted by power uprate, since neither the addition of dissolved solids nor the rate of addition of particulates into the steam generators will be significantly impacted by power uprate.

Since the range of NSSS PCWG parameters permits a large variation in full load steam pressure (827 to 1024 psia), the inlet pressure to the steam generator blowdown system can also vary accordingly. Therefore, the control range of the steam generator blowdown system valves was reviewed and verified to have adequate margin to maintain design blowdown capability with the variation in system inlet pressure.

4.2.4 Conclusions

The following is a brief summary of the NSSS/BOP interface evaluation conclusions for the Byron/Braidwood Units 1 and 2 Power Uprate Program. Refer to the identified sections for a more detailed discussion.

4.2.4.1 Main Steam System

- Operation at reduced steam pressures and corresponding higher pressure drops will have a negative effect on plant heat rate.

- The capacity of the installed MSSVs meets the Westinghouse sizing criterion for the proposed range of NSSS operating conditions.

- The capacity of the installed SG PORVs meets the Westinghouse sizing criterion for the proposed range of NSSS operating conditions.

- The MSIVs and MSIV bypass valves are not adversely impacted by the uprating.

4.2.4.2 Steam Dump System

- The capacity of the steam dump system meets the Westinghouse sizing criterion for the proposed range of NSSS operating conditions.

4.2.4.3 Condensate and Feedwater System

- The increased feedwater flow that enters the Unit 2 steam generator pre-heaters was determined to be acceptable in terms of tube wear. Based on the range of NSSS operating parameters approved for power uprate, the high flow rate alarm set point will be increased to a value no lower than 90 percent of maximum expected feedwater flow (3.61×10^6 lb/hr) to provide adequate margin for plant operations. This minimum alarm set point (3.61×10^6 lb/hr) will permit a maximum combined instrumentation channel uncertainty of ± 1.6 percent, since the pre-heater tube wear study concluded that a maximum pre-heater flow rate of 3.672×10^6 lb/hr is acceptable. Refer to the BOP Engineering Report for evaluation of the required alarm set point change.

- The existing feedwater pump speed control program may result in unacceptable main feedwater control valve lift for the range of NSSS operating conditions approved for uprate. Therefore the feedwater pump speed controller will be re-calibrated to achieve optimum valve lift based on measured head loss data at the actual uprate operating conditions. Refer to the BOP Engineering Report for evaluation of the required speed control program changes.

4.2.4.4 Auxiliary Feedwater System

- The minimum flow requirements of the AF System are dictated by accident analysis and since the uprating impacts these analyses, evaluations of the limiting transients and accidents have been performed to confirm that the AF System performance will remain acceptable at the uprated operating conditions. These analyses are described in Section 6 of this report and show acceptable results.

- The CST minimum useable inventory of 200,000 gals is adequate to meet the plant design bases requirements for the range of NSSS operating conditions approved for plant uprate. Therefore, no change is required to the Technical Specifications covering CST inventory requirements.

4.2.4.5 Steam Generator Blowdown System

- The actual required blowdown flow rates during plant operation will not be significantly impacted by power uprate, since neither the rate of addition of dissolved solids nor the rate of addition of particulates into the steam generators will be significantly impacted by power uprate.

- To maintain a design blowdown capability, the control range of the steam generator blowdown system control valves has been verified to have adequate margin to accommodate the proposed variation in system inlet pressure.

4.2.5 References

1. Westinghouse Steam Systems Design Manual, WCAP-7451, Revision 2, August 1973.

2. Byron and Braidwood UFSAR, Rev. 7, 12/98.

4.3 NSSS Control Systems

4.3.1 Pressure Relief Component Sizing

Based on the operating conditions for the Byron/Braidwood uprating program (Section 2.0), the installed capacities of the following NSSS pressure control components have been evaluated at the uprated conditions.

- Pressurizer Power Operated Relief Valves (PORVs)
- Pressurizer Spray Valves
- Pressurizer Heaters
- Steam Dump Relieving Capacity

The following concludes that the installed capacity of the listed pressure control components is still acceptable for the uprated conditions. This analysis confirms that the design basis load rejection transient acceptance criteria can be met with adequate margin.

4.3.1.1 Pressurizer Power Operated Relief Valves (PORVs)

4.3.1.1.1 Introduction

The sizing basis for the pressurizer PORVs is to prevent the pressurizer pressure from reaching the high pressurizer pressure reactor trip setpoint for the design basis load rejection with steam dump transient. For the Byron/Braidwood plants, this is a step change in the turbine load from 100% to 50% power.

4.3.1.1.2 Input Parameters and Assumptions

The pressurizer PORV sizing analysis was performed at the Byron/ Braidwood uprating design conditions defined in Section 2.1. The analysis was intended to bracket all operating conditions; BWI and Westinghouse D5 steam generators, full power T_{avg} ranging from 575 to 588°F, and zero to maximum steam generator tube plugging (SGTP) levels. The adequacy of the PORV relief capacity is most severely challenged for the operating condition that results in the largest surge flow into the pressurizer. The following assumptions were made in the analysis:

1. Initial power level at 102% of full uprated thermal power

2. Reactor Coolant System (RCS) vessel average temperature (T_{avg}) measurement uncertainty: +4°F.

3. Pressurizer PORV installed capacity: 210,000 lb/hr saturated steam per valve at 2335 psig, total of 2 valves.

4. Pressurizer spray valves total capacity: 900 gpm.

5. Pressurizer heater maximum installed total capacity: 1802 kW, split between 1386 kW in back-up heaters and 416 kW in proportional heaters.

6. Control Systems Actuation Logic and Setpoints: NSSS control systems actuation logic and corresponding setpoints for rod control, steam dump control, pressurizer pressure and level control, and steam generator level control systems are included.

7. Setting the setpoints or trip delay time artificially high defeated protection system actuation.

8. Best estimate Nuclear Design Parameters (moderator temperature coefficient, Doppler power defect, control rod worth, and startup data) are assumed. Conservatism was applied by assuming the moderator temperature coefficient was zero at all times. In actuality, the full-power moderator temperature coefficient would be expected to be negative for all times in core life.

The key analysis assumptions for the pressurizer PORV sizing analysis are provided in subsection 4.3.1.1.3.

4.3.1.1.3 Description of Analysis and Evaluations

A 50% load rejection with steam dump transient was analyzed using the Byron/Braidwood model of the LOFTRAN code (Reference 1), used for non-LOCA analyses, as a starting point for plant definition (RCS volumes, power levels, RCS flow mixing coefficients, PORV/safety valve flows, pressurizer heater and spray capacities, etc.). Since the 50% load rejection transient is loop symmetric, a single-loop version of the LOFTRAN code was used (this is the same LOFTRAN code version used in the original PORV sizing analysis). This computer code is a system level program code and models the overall NSSS including the detailed modeling for control and protection systems. The method of analysis used was similar to the standard sizing procedure for pressurizer PORV original sizing calculations. The major analysis assumptions were as follows.

1. A 50% load rejection with steam dump transient is initiated from 102 percent of full uprated thermal power (102 to 50 percent load) in a step manner.

2. Initially, the analysis was performed for the D5 steam generator at all four identified full power conditions (High T_{avg} (588°F) and low T_{avg} (575°F) conditions, and 0% and 10% tube plugging) described in Section 2.1. Based on the results, the worst resulting case was then run at the corresponding condition for the BWI steam generators. The worst (limiting) case was the Low T_{avg}, 0% tube plugging condition for the BWI steam generator.

3. A conservative initial steam generator mass 50-percent lower than nominal is assumed.

4. The initial pressurizer water level is nominal.

5. All normally active control systems; this includes pressurizer pressure and level control, steam dump control, and rod control systems are modeled.

6. To model steam generator level control, feedwater flow was assumed to step to 70% of nominal and remain constant throughout the transient.

7. A conservative 2-second delay is assumed from transient initiation before the steam dump valves start opening.

4.3.1.1.4 Acceptance Criteria

The installed pressurizer PORVs capacity should limit pressurizer pressure to less than the fixed high pressurizer pressure reactor trip setpoint on a design basis full-load rejection with steam dump transient. This criterion is conservatively met if the total PORV capacity is greater than or equal to the peak pressurizer in-surge flow rate during this transient.

4.3.1.1.5 Results

For all cases analyzed (high and low T_{avg}, 0-percent and maximum SGTP level, and BWI and D5 steam generator designs), the results showed that a maximum total relief capacity of 364,789 lb/hr saturated steam at 2350 psia is required at the uprating conditions. The maximum total installed relief capacity of the pressurizer PORVs is 420,000 lb/hr saturated steam at 2350 psia.

4.3.1.1.6 Conclusions

Since the installed total capacity of 420,000 lb/hr saturated steam at 2350 psia is greater than the required relief capacity, the installed PORVs are adequate for the Byron/Braidwood uprating conditions.

4.3.1.2 Pressurizer Spray Valves

4.3.1.2.1 Introduction

The sizing basis for pressurizer spray valves is to prevent challenges to the pressurizer PORVs from a 10 percent step-load decrease transient. For load decreases up to 10 percent power, the spray valves are the sole means of controlling pressure without actuating the pressurizer PORVs when in automatic pressure control mode.

4.3.1.2.2 Input Parameters and Assumptions

The pressurizer spray valves sizing analysis was performed at the Byron/Braidwood uprated design conditions defined in Section 2.1. Other key input parameters and assumptions are listed below.

1. A 10 percent step-load decrease transient is initiated from 102 percent full uprated thermal power (102 to 90 percent power transient). A transient initiated from full power bounds all lower initial power levels.

2. The pressurizer spray valves sizing analysis was performed for the Low T_{avg}, 0% plugging BWI steam generator case. Similar to the PORV sizing analysis described in Section 4.3.1.1, this analysis is an RCS heatup transient. Therefore, the same sensitivity results are expected as for the PORV sizing; the Low T_{avg}, 0% plugging, BWI steam generator case would result in the largest pressurizer insurge and the greatest potential for challenging the pressurizer PORVs.

3. The initial T_{avg} is assumed to be nominal with 4°F uncertainty.

4. A conservative initial steam generator mass of 90 percent nominal is assumed.

5. Rod control, pressurizer spray control, pressurizer level control, and steam generator level control systems are assumed operational in automatic mode. Pressurizer level control and steam generator level control systems are not explicitly modeled in the analysis since it is not critical to the analytical results. The transient pressurizer pressure response of concern (reaching the pressurizer PORV setpoint) happens too rapidly for pressurizer level to impact the results. Forcing feedwater flow to match steam flow simulates automatic steam generator level control.

6. Best estimate reactor kinetics (i.e., moderator temperature coefficient, Doppler power defect, control rod worth, etc.) at BOL conditions are used.

4.3.1.2.3 Description of Analysis and Evaluations

A 10 percent step-load decrease (10 percent nominal plus 2 percent uncertainty) from full power transient was analyzed using the Byron/Braidwood model of the LOFTRAN code. The method of analysis is similar to the standard sizing procedure for pressurizer spray valves original sizing calculations.

4.3.1.2.4 Acceptance Criteria

The design capacity (900 gpm total) for the pressurizer spray valves should limit pressurizer pressure to less than the pressurizer PORV actuation setpoint on a 10 percent step-load decrease transient.

4.3.1.2.5 Results

For the limiting case analyzed (low T_{avg}, zero-percent SGTP level and BWI steam generator), the results showed a maximum peak pressurizer pressure of 2337 psia at uprating design conditions.

4.3.1.2.6 Conclusions

Since the peak pressurizer pressure is less than the PORV actuation setpoint, the installed capacity of not less than 900 gpm is adequate.

4.3.1.3 Pressurizer Heaters

Pressurizer heater total capacity is proportional to pressurizer volume. Therefore, the pressurizer heater requirements are unaffected by power uprate.

4.3.1.4 Post-Trip Steam Dump Capacity

4.3.1.4.1 Introduction

There are two sizing bases for the steam dump system:

a. To allow the plant to accept sudden large load decreases without incurring a reactor trip or challenging the steam generator safety valves, and

b. To remove stored energy and residual heat following a turbine/reactor trip to bring the plant to equilibrium no-load conditions without actuation of steam generator safety valves

Criterion (a) is analyzed as part of the NSSS control systems evaluation described in Section 4.3.2. Criterion (b) is evaluated here. The acceptance criteria is that following a reactor trip from 100% uprated power, steam pressure does not rise to the lowest steam generator safety valve nominal setpoint of 1190 psia, minus the 5% setpoint tolerance.

4.3.1.4.2 Input Parameters and Assumptions

The post-trip steam dump sizing analysis was performed at the Byron/Braidwood uprated design conditions defined in Section 2.1. Other key input parameters and assumptions are listed below.

1. The transient is initiated by a turbine trip occurring at 102% nominal power. Stepping the steam flow to zero simulates this.

2. A reactor trip is assumed generated 2.5 seconds after turbine trip.

3. All normally functioning control systems are active. This includes pressurizer pressure and level control, steam dump control, and steam generator level control. As described in Section 4.3.1.2, steam generator level control is not explicitly modeled. For this analysis, the feedwater flow is assumed to step to 70% of nominal at turbine trip and it then remains at this value until the feedwater is isolated at 564°F.

4. The analysis was performed for the High T_{avg}, 0% tube plugging, BWI steam generator case. This case is limiting since it has the highest full-power steam pressure and the lowest margin to the steam generator safety valve setpoint.

5. A conservative initial steam generator mass 10 percent below nominal is assumed.

6. The initial T_{avg} is nominal. The initial pressurizer pressure and water volume are nominal.

7. Best estimate reactor kinetics (i.e., moderator temperature coefficient, Doppler power defect, control rod worth, etc.) at BOL conditions are used (limiting for heatup transient).

8. Decay heat is set to the best-estimate value expected for long-term plant operation prior to the reactor trip.

4.3.1.4.3 Description of Analysis and Evaluations

A reactor trip from 100% power was analyzed using the Byron/Braidwood model of the LOFTRAN code.

4.3.1.4.4 Acceptance Criteria

Peak steam pressure should be limited to less than the nominal steam generator main steam safety valve lowest setpoint of 1130.5 psia (nominal setpoint of 1190 psia minus 5% tolerance).

4.3.1.4.5 Results

For the limiting case analyzed, the peak steam pressure was 1139 psia.

4.3.1.4.6 Conclusions

The acceptance criteria is that a reactor trip from nominal 100% power should not challenge the steam generator safety valves. The lower valve setpoint is 1190 psia nominally; with a 5% setpoint tolerance, the safety valves may actuate as low as 1130.5 psia. On a reactor trip for the limiting case (high T_{avg}, 0% tube plugging and BWI steam generators), the safety valves could be challenged if the setpoint is at maximum tolerance below the nominal value. For the safety valves opening at their nominal setpoint of 1190 psia, the safety valve will not be challenged for any plant condition identified in Section 2.1.

As the full-power steam pressure is reduced below the 1024 psia peak value identified in the Section 2.1 operating conditions, the potential for challenging the steam generator safety valves on a reactor trip is reduced. The potential for safety valve actuation at the peak 1024 psia steam pressure conditions is judged minimal based upon the following conservatisms taken in the analysis:

- Initial power level of 102% of nominal uprated value
- maximum reactor trip delay time of 2.5 seconds
- maximum steam dump valve stroke times used
- maximum safety valve tolerance used

Additional evaluations to support operation at steam pressures up to 1035 psig were performed and found to be acceptable. For the reasons stated above, the potential for safety valve actuation at 1035 psia is also judged to be minimal.

4.3.1.5 References

1. WCAP-7878, Rev. 5, "LOFTRAN Code Description," G. E. Heberle, November 1989.

4.3.2 Control Systems Setpoints Analysis

4.3.2.1 Introduction

An evaluation was performed to determine the impact of uprate conditions on the NSSS control system setpoints for the Byron and Braidwood units. The conditions used as inputs are provided in Section 2.0 of this report (PCWG parameters) and encompass a range of primary T_{avg} values. In addition, normal operation at power levels from 94 to 100% was examined, for consistency with earlier analyses and for additional operating flexibility.

4.3.2.2 Reactor Control System Setpoints

The functions of the reactor control system are:

- to maintain a desired reactor coolant average temperature as a function of power, and
- to automatically control the reactor in the power range between 15 and 100 percent rated power for 10% step changes in load, 5% per minute ramp loadings and unloadings, and 50% load rejections with the aid of load rejection steam dump.

The reactor control coolant average temperature program will be modified to maintain the desired programmed reactor coolant temperature. No additional reactor control system setpoint modifications are required for the uprate program, since the system demonstrates response similar to that seen for the current operating temperature and power parameters.

4.3.2.3 Remaining NSSS Control System Setpoints

The following additional control system setpoints were evaluated for changes resulting from the uprate parameters, and were determined to remain applicable:

- Feedwater Pump Speed Controller
- Load Rejection Steam Dump Controller
- Plant trip controller setpoints
- Pressurizer pressure control system
- Pressurizer level control system
- Rod control system
- Turbine loading stop

4.3.2.4 NSSS Control System Alarms

The alarms that will be modified as a result of the uprate program, based on full-power average temperature and delta T for a particular fuel cycle, are the insertion limit alarms and high auctioneered T_{avg} temperature alarm. In addition, the low steamline pressure alarm will be modified as a function of the uprate design steam pressure for a particular fuel cycle.

4.3.2.5 Conclusions

The NSSS control systems setpoints for Byron and Braidwood Units 1 and 2 were reviewed for the uprate conditions. Modifications will be made to the reactor control coolant average temperature program. In addition, the following alarms will be modified:

- insertion limit alarms
- high auctioneered T_{avg} temperature alarm
- low steamline pressure alarm

The remaining NSSS control system setpoints remain applicable for the uprate conditions.

4.3.2.6 References

None

4.3.3 Low Temperature Overpressure Protection (LTOP) System Setpoint Analysis

4.3.3.1 Introduction

At relatively low reactor coolant temperatures (less than approximately 350°F), two potential overpressure transients to the RCS have been defined as the design basis for the Low Temperature Overpressure Protection (LTOP) System. Both transients are assumed to occur when the RCS is water-solid.

The first transient is a heat addition scenario in which a reactor coolant pump in a single loop is started when the RCS temperature is up to 50°F lower than the steam generator secondary side temperature. This results in a sudden secondary to primary heat transfer and rapid increase in primary system pressure.

The second design basis transient is a mass injection event. The influx of fluid into the relatively inelastic RCS also causes a rapid increase in system pressure. The mass input mechanisms considered in the analyses involved operation of one centrifugal charging pump with inadvertent isolation of letdown flow.

This section documents the development of the PORV setpoint program (pressure setpoint vs. RCS temperature) as determined for the LTOP System Setpoint Analysis for the Byron/Braidwood power uprate project. This program will maintain RCS pressure within acceptable limits following all credible overpressurization incidents occurring in the Byron/Braidwood Plants during low temperature, water-solid operation.

4.3.3.2 Input Parameters and Assumptions

The input parameters and assumptions used were consistent with those applied in the current analysis of record. Analysis input was revised to reflect the uprated plant conditions as defined in Section 2.1. Both the Westinghouse D5 and BWI steam generator types were analyzed.

4.3.3.3 Analyses and Evaluation

Setpoints were developed using the current Westinghouse methodology outlined in WCAP-14040, Rev. 2 (Reference 1), with the following exception.

1. The Byron/Braidwood Plants heatup and cooldown curves used were developed using ASME Code Case N-640. Code Case N-640 is the latest approved industry standard pertaining to the development of P-T curves.

The LTOP transient analyses were performed using the Byron/Braidwood models of the special versions of the LOFTRAN Code. A special version single-loop LOFTRAN code (LOFT12) was used for the mass injection transient analysis. This single loop version of the LOFTRAN code combines all RCS loops into a single loop and thus the total RCS volume is maintained. For heat addition transient analysis, a special version multi-loop LOFTRAN code (LOFT4) was used. The mass injection transients are loop-symmetric and, therefore, use of a single-loop version of LOFTRAN code is appropriate. These special versions of the LOFTRAN code have been and are currently used for Westinghouse LTOP transients analysis. The LOFTRAN computer code is a system-level program code that models the overall NSSS. These special versions of LOFTRAN code have the capability to model the pressurizer PORV characteristics and perform automatic data reduction (PORV overshoot and undershoot). The NRC has approved both single-loop and multi-loop versions of the base LOFTRAN code, as documented in WCAP-7907-P (Reference 2).

The 32 EFPY Appendix G limits were provided in Section 5.1. The Appendix G limits were adjusted to account for the differential pressure resulting from Reactor Coolant Pump (RCP) and Residual Heat Removal operation, as shown in Tables 4.3.3-1 and 4.3.3-2. The minimum of the adjusted Appendix G limit and the PORV discharge piping limit of 800 psig form the upper limit for the LTOP System setpoints. The minimum RCS pressure required to start a Reactor Coolant pump (RCP) defines the lower limit. The lower limit was taken from the current analysis of record.

4.3.3.4 Acceptance Criteria

The maximum allowable setpoints are to be selected such that peak RCS pressures will not exceed the steady state Appendix G limits or the pressurizer PORV piping limit of 800 psig, whichever is limiting. In addition, the maximum recommended setpoints should prevent simultaneous actuation of both PORVs during a design basis mass injection or heat addition event and maintain RCS pressure margin to the RCP seal limit.

4.3.3.5 Results

PORV over- and under-shoots, determined with LOFTRAN, were used along with the upper and lower limits to define the maximum allowable LTOP System setpoints. The maximum allowable LTOP System setpoints (without instrumentation uncertainties) are defined in Table 4.3.3-3. These setpoints satisfy the acceptance criteria contained in Section 4.3.3.4.

4.3.3.6 Conclusions

Operation during RCS water-solid low-temperature condition is acceptable since protection is afforded by proper PORV setpoint selection.

4.3.3.7 References

1. WCAP-14040-NP-A, Rev. 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Andrachek et. al., January 1995.

2. WCAP-7907-P-A, "LOFTRAN Code Description," Burnett, T. W. T., et al., April 1984.

Table 4.3.3-1 Transmitter ΔP				
# of RCPs running	ΔP w/o Trains of RHR (psi)	ΔP w/1 Train of RHR (psi)	ΔP w/2 Trains of RHR (psi)	Maximum ΔP Between Vessel and Transmitter (psi)[1]
4	69.25	70.15	70.94	70.94
3	49.04	50.02	50.63	50.63
2	35.06	36.07	36.83	36.83
1	27.29	28.74	N/R[2]	28.74

1 Maximum of 0, 1, or 2 trains of RHR in operation

2 N/R = not reported

Table 4.3.3-2 32 EFPY Steady State Pressure/Temperature Limits				
RCS Temperature °F	Byron Unit 1 psig	Byron Unit 2 psig	Braidwood Unit 1 psig	Braidwood Unit 2 psig
60		0	0	0
60	762.68	1101.78	1082	931
65	779.45	1154.21	1133	965
70	797.98	1212.16	1188	1003
75	818.46	1276.2	1250	1045
80	841.1	1346.97	1318	1092
85	866.11	1425.19	1393	1143
90	893.76	1511.64	1476	1200
95	924.31	1607.18	1568	1263
100	958.08	1712.76	1669	1332
105	995.4	1829.46	1781	1409
110	1036.65	1958.42	1905	1494
115	1082.23	2100.94	2042	1587
120	1132.61	2258.46	2194	1691
125	1188.28	2432.54	2361	1805
130	1249.81			1932
135	1317.81			2071
140	1392.96			2226
145	1476.02			2396
150	1567.81			
155	1669.26			
160	1781.37			
165	1905.28			
170	2042.22			
175	2193.56			
180	2360.81			

Table 4.3.3-3 Maximum Allowable PORV Setpoints (w/o Instrument Uncertainty)				
RCS Temp (°F)	**Byron Station**		**Braidwood Station**	
	Unit 1 (psig)	**Unit 2 (psig)**	**Unit 1 (psig)**	**Unit 2 (psig)**
60	644	747	754	747
70	679	747	754	747
100	754	747	754	747
120	754	747	754	747
125	754	747	754	747
150	754	747	754	747
170	754	747	754	747
175	752	747	754	747
200	745	747	745	747
220	735	746	735	746
250	720	734	720	734
300	697	715	697	715
350	668	702	668	702

5.0 NSSS COMPONENTS

Evaluations were performed to determine the effects of the uprate parameters on the NSSS components. In general, the uprate-related input used for these evaluations were the PCWG parameters (Section 2.0) and the NSSS design transient changes (Section 3.1). Additional input parameters specific to particular components (e.g., NSSS auxiliary equipment design transients for the auxiliary equipment evaluations) were considered and are discussed in the appropriate component evaluation section. The purpose of the evaluations performed for the NSSS components was to confirm that they continue to satisfy the applicable codes, standards, and regulatory guides under the uprate conditions.

Evaluations were performed in the following areas, and are described within the remainder of this chapter:

- Reactor Vessel Structural Evaluations and Integrity
- Reactor Pressure Vessel System
- Fuel Assemblies
- Control Rod Drive Mechanisms
- Reactor Coolant Loop Piping and Supports
- Reactor Coolant Pumps and Motors
- Steam Generators
- Pressurizer
- NSSS Auxiliary Equipment
- Loop Stop Isolation Valves

5.1 Reactor Vessel

5.1.1 Structural Evaluation

5.1.1.1 Introduction

Evaluations were performed for the various regions of the Byron and Braidwood Units 1 and 2 reactor vessels to determine the stress and fatigue usage effects of Nuclear Steam Supply System (NSSS) operation at the revised operating conditions of the Byron and Braidwood Power Uprate Project throughout the current plant operating licenses.

5.1.1.2 Input Parameters and Assumptions

The evaluations assess the effects of the design transients on the most limiting locations with regard to ranges of stress intensity and fatigue usage factors. The evaluations consider a worst-case set of design transients from among the high temperature power uprate conditions, the low temperature power uprate conditions, and the original design basis. The Power Uprate Project (Section 2.0) did not affect the reactor vessel design parameters. The design transients were identified in Section 3.0. Furthermore, no new design interface loads were identified as a result of the Power Uprate Project.

5.1.1.3 Description of Analysis

Reactor vessel operation from plant startup through implementation of the power uprate and any future operation in accordance with the original design basis remain bounded by the stress and fatigue analyses. Where appropriate, revised maximum ranges of stress intensity and maximum usage factors were evaluated for the Power Uprate Project. In all cases, the original design basis stress analysis remains conservative so that no new calculational results were identified. However, the very conservative maximum range of stress intensity and cumulative fatigue usage factor results for the Byron inlet nozzle safe ends were applied to Braidwood to simplify the design transient evaluation.

The evaluations of the Byron Units 1 and 2 and Braidwood Units 1 and 2 reactor vessels show that they are acceptable for plant operation in accordance with the Power Uprate Project. Therefore, the reactor vessel power uprate evaluation addresses reactor operation with the operating temperature ranges and design transients as discussed in Section 3.0. Such operation is shown to be acceptable in accordance with the ASME Boiler and Pressure Vessel Code (Reference 1) for the remainder of the plant licenses.

5.1.1.4 Acceptance Criteria

1. The maximum range of primary plus secondary stress intensity resulting from normal and upset condition design transient mechanical and thermal loads shall not exceed 3 S_m at operating temperature (Reference 1, Paragraph NB-3222.2). NB-3228-3 elastic-plastic analysis is an alternate acceptance criteria if 3 S_m criteria is not being met.

2. The maximum cumulative usage factor resulting from the peak stress intensities due to the normal and upset condition design transient mechanical and thermal loads shall not exceed 1.0 in accordance with the procedure outlined in Paragraph NB-3222.4 of Reference 1.

5.1.1.5 Results

The reactor vessel power uprate evaluation demonstrates that the power uprate does not increase the ranges of stress intensity or cumulative fatigue usage factors for any of the various regions of the reactor vessels, except for the Braidwood inlet nozzles. For the Braidwood reactor vessel inlet nozzles, conservative results for the Byron safe ends were applied to the Braidwood safe ends to simplify the evaluation. The Braidwood reactor vessel inlet nozzles continue to meet the ASME Code design requirements. Otherwise, the maximum ranges of primary plus secondary stress intensity remain as evaluated and justified in the original design basis. Additionally, the maximum cumulative fatigue usage factors reported for the original design basis are otherwise unaffected by the power uprate conditions and remain significantly below the acceptance criterion of 1.00.

Results of the Power Uprate Project evaluation are shown in Tables 5.1.1-1 (Byron) and 5.1.1-2 (Braidwood).

5.1.1.6 Conclusions

The reactor vessel structural evaluation concludes that all acceptance criteria continue to be met for the Power Uprate Project.

5.1.1.7 References

1. ASME Boiler and Pressure Vessel Code, Section III, *Nuclear Power Plant Components,* 1971 Edition with Addenda through the Summer 1973, American Society of Mechanical Engineers, New York.

Table 5.1.1-1 Stress Intensities and Fatigue Usage Factors for the Byron Reactor Vessels		
Location	**$P_L + P_b + Q$ Range**	**U_c**
Outlet Nozzles	Nozzle: 105.5 ksi* > 3 S_m = 80.1 ksi Safe End: 51.5 ksi* > 3 S_m = 50.52 ksi	Nozzle: 0.274 < 1.0
Inlet Nozzles	Nozzle: 91.2 ksi* > 3 S_m = 80.1 ksi Safe End: 84.53 ksi* > 3 S_m = 49.8 ksi	Nozzle: 0.215 < 1.0 Safe End: 0.032 < 1.0
Main Closure Flange Region		
Closure Head Flange	65.5 ksi < 3 S_m = 80.1 ksi	0.083 < 1.0
Vessel Flange	68.8 ksi < 3 S_m = 80.1 ksi	0.086 < 1.0
Closure Studs	80.8 ksi < 3 S_m = 109.9 ksi	0.883 < 1.0
Vessel Shell		
Vessel Wall Transition	47.0 ksi < 3 S_m = 80.1 ksi	0.013 < 1.0
Bottom Head to Shell Juncture	41.8 ksi < 3 S_m = 80.1 ksi	—
Core Support Guides	42.32 ksi < 3 S_m = 80.1 ksi	0.211 < 1.0
CRDM Housings	62.95 ksi < 3 S_m = 69.9 ksi	0.021 < 1.0
Bottom Head Instrumentation Tubes	78.7 ksi* > 3 S_m = 69.9 ksi	0.0528 < 1.0

* Justified by simplified elastic-plastic analysis accordant with Paragraph NB-3228.3 in Section III of the ASME Boiler and Pressure Vessel Code (Reference 1).

Table 5.1.1-2
Stress Intensities and Fatigue
Usage Factors for the Braidwood Reactor Vessels

Location	$P_L + P_b + Q$ Range	U_c
Outlet Nozzles	Nozzle: 106.6 ksi* > 3 S_m = 80.1 ksi Safe End: 51.5 ksi* > 3 S_m = 50.52 ksi	Nozzle: 0.3205 < 1.0
Inlet Nozzles	Nozzle: 106.5 ksi* > 3 S_m = 80.1 ksi Safe End: 84.53 ksi* > 3 S_m = 49.8 ksi	Nozzle: 0.302 < 1.0 Safe End: 0.032 < 1.0
Main Closure Flange Region		
Closure Head Flange	66.9 ksi < 3 S_m = 80.1 ksi	0.289 < 1.0
Vessel Flange	75.3 ksi < 3 S_m = 80.1 ksi	0.363 < 1.0
Closure Studs	82.2 ksi < 3 S_m = 109.9 ksi	0.941 < 1.0
Vessel Shell		
Vessel Wall Transition	43.7 ksi < 3 S_m = 80.1 ksi	0.009 < 1.0
Bottom Head to Shell Juncture	37.3 ksi < 3 S_m = 80.1 ksi	—
Core Support Guides	42.32 ksi < 3 S_m = 80.1 ksi	0.211 < 1.0
CRDM Housings	62.95 ksi < 3 S_m = 69.9 ksi	0.021 < 1.0
Bottom Head Instrumentation Tubes	78.7 ksi* > 3 S_m = 69.9 ksi	0.0528 < 1.0

* Justified by simplified elastic-plastic analysis accordant with Paragraph NB-3228.3 in Section III of the ASME Boiler and Pressure Vessel Code (Reference 1).

5.1.2 Reactor Vessel Integrity (Byron Units)

5.1.2.1 Introduction

Reactor vessel integrity is impacted by any changes in plant parameters that affect neutron fluence levels or temperature/pressure transients. The changes in neutron fluence resulting from the proposed Commonwealth Edison Company Byron Units 1 and 2 Thermal Power Uprate Program have been evaluated to determine the impact on reactor vessel integrity. This assessment included a review of the current material surveillance capsule withdrawal schedules[1, 2], development of new plant heatup and cooldown pressure-temperature (P-T) limit curves, review of the Emergency Response Guideline (ERG)[3] limits, and a revision to the RT_{PTS} values (10 CFR 50.61[4], known as the Pressurized Thermal Shock (PTS) Rule).

The most critical area, in terms of reactor vessel integrity, is the beltline region of the reactor vessel. The beltline region is defined in ASTM E185-82[5] as "the irradiated region of the reactor vessel (shell material including weld regions and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions that are predicted to experience sufficient neutron damage to warrant consideration in the selection of surveillance material." Figures 5.1.2-1 and 5.1.2-2 identify and indicate the location of all beltline region materials of the Byron Units 1 and 2 reactor vessels.

5.1.2.2 Input Parameters and Assumptions

Uprated Fluence Projections

Fluence projections (Section 7.5) on the vessel were evaluated for the uprated power level for input to the reactor vessel integrity evaluations. Typically, fluence values are used to evaluate end-of-life (EOL) transition temperature shift (EOL ΔRT_{NDT}) for development of surveillance capsule withdrawal schedules, determining EOL upper shelf energy (USE) values, adjusted reference temperature (ART) values for determining applicability of heatup and cooldown curves, ERG limits, and RT_{PTS} values.

Inlet Temperature

Due to the uprated power, the reactor vessel inlet temperature may also change. This updated inlet temperature (Section 2.0) has been reviewed to verify its compliance with ASTM E900[6], which is the basis for Regulatory Guide 1.99, Revision 2.

5.1.2.3 Description of Analysis/Evaluations

The reactor vessel integrity evaluation for the Byron Units 1 and 2 uprating included the following objectives:

1. Review reactor vessel surveillance capsule removal schedules for Byron Units 1 and 2 to determine if changes are required as a result of changes in vessel fluence due to the Thermal Power Uprate Program. This evaluation is consistent with the recommended practices of ASTM E185-82 and meets the requirements of Appendix H of 10 CFR Part 50[7].

2. Revised P-T limit curves have been generated for 22 and 32 EFPY at Byron Units 1 and 2. These curves account for the effects of uprated fluence projections and the latest Code changes as described later in this report. The methodology of Regulatory Guide 1.99, Revision 2[8], will be used in any required ART calculations.

3. Revised RT_{PTS} values have been determined accounting for the effects of uprated fluence projections in the Byron Units 1 and 2 reactor vessel at EOL (32 EFPY) and license renewal (48 EFPY). The current PTS Rule, 10 CFR Part 50.61[4], will be used to ensure that the screening criteria is met.

4. Review USE values at EOL for all reactor vessel beltline materials in the Byron Units 1 and 2 reactor vessels to assess the impact of uprated fluence projections.

5. Review reactor vessel inlet temperature for Byron Units 1 and 2 to verify that it maintains an acceptable level after the uprated condition takes affect.

Surveillance Capsule Withdrawal Schedules

A surveillance capsule withdrawal schedule is developed to periodically remove surveillance capsules from the reactor vessel. This procedure is followed to effectively monitor the condition of the reactor vessel materials under actual operating conditions. ASTM E185-82 defines the recommended number of surveillance capsules and the recommended withdrawal schedule, based on the vessel material predicted transition temperature shifts (ΔRT_{NDT}). The surveillance capsule withdrawal schedule is in terms of effective full-power years (EFPY) of plant operation, with a design life of 32 EFPY. Other factors that must be considered in establishing the surveillance capsule withdrawal schedule are maximum fluence values at the vessel inside surface and 1/4-thickness (1/4T) location.

The first surveillance capsule is typically scheduled to be withdrawn early in vessel life to verify initial predictions of surveillance material response to the actual radiation environment. It is generally removed when the predicted shift exceeds expected scatter by sufficient margin to be measurable. Normally, the capsule with the highest lead factor is withdrawn first. Early withdrawal also permits verification of the adequacy and conservatism of reactor vessel pressure-temperature operational limits.

The withdrawal schedule for the maximum number of surveillance capsules to be withdrawn is adjusted by the lead factor so that:

- exposure of the *second* surveillance capsule withdrawn occurs when the accumulated neutron fluence of the capsule corresponds to a value midway between that of the first and third capsules,

- exposure of the *third* surveillance capsule withdrawn does not exceed the peak EOL 1/4T fluence,

- exposure of the *fourth* surveillance capsule withdrawn does not exceed the peak EOL reactor vessel fluence, and

- exposure of the *fifth* surveillance capsule withdrawn does not exceed twice the peak EOL reactor vessel fluence.

Per ASTM E185-82, the four steps used for development of a surveillance capsule withdrawal schedule are as follows:

1. Estimate peak vessel inside surface fluence at EOL and the corresponding transition temperature shift (ΔRT_{NDT}). This identifies the number of capsules required.

2. Obtain the lead factor for each surveillance capsule relative to peak beltline fluence.

3. Calculate the EFPY for the capsule to reach peak vessel EOL fluence at the inside surface and 1/4T locations. These results are used to establish the withdrawal schedule for all but the first surveillance capsule.

4. Schedule surveillance capsule withdrawals at the nearest vessel refueling date.

A current surveillance capsule withdrawal schedule for Byron Units 1 and 2 reactor vessels is documented in WCAP-15123 Rev. 1[1] and WCAP-15176[2]. This schedule has been evaluated for the Power Uprate Project due to increased neutron fluence.

<u>Heatup and Cooldown Pressure-Temperature Limit Curves</u>

New heatup and cooldown curves for Byron Units 1 and 2 have been generated using uprated fluence projections at 22 and 32 EFPY. These heatup and cooldown curves and the subsequent data points are documented in WCAP-15391[9] (Unit 1) and WCAP-15392[10] (Unit 2).

The heatup and cooldown curves documented in these reports were generated using the most limiting ART values and the NRC approved methodology documented in WCAP-14040-NP-A, Revision 2[11], "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" with the following exceptions:

1. The fluence values used in this report are calculated fluence values, not the best estimate fluence values.

2. The K_{Ic} critical stress intensities are used in place of K_{Ia} critical stress intensities. This methodology is taken from approved ASME Code Case N-640[12].

3. The reactor vessel flange temperature requirement has been eliminated. Justification has been provided in WCAP-15315[13].

4. The 1996 Version of Appendix G to Section XI[14] was used rather than the 1989 version.

5. The methodology from approved ASME Code Case N-588[18] was used to consider circumferentially oriented flaws in development of P-T Curves.

Pressurized Thermal Shock (PTS)

Pressurized Thermal Shock (PTS), a limiting condition on reactor vessel integrity, may occur during a severe system transient such as a loss-of-coolant-accident (LOCA) or a steam line break. Such transients may challenge the integrity of a reactor vessel under the following conditions:

- severe overcooling of the inside surface of the vessel wall followed by high repressurization,
- significant degradation of vessel material toughness caused by radiation embrittlement, and
- the presence of a critical-size defect in the vessel wall.

The PTS concern arises if one of these transients acts on the beltline region of a reactor vessel where a reduced fracture resistance exists because of neutron irradiation. Such an event may produce the propagation of flaws postulated to exist near the inner wall surface, thereby potentially affecting the integrity of the vessel.

In 1985, the Nuclear Regulatory Commission (NRC) issued a formal ruling on PTS. It established screening criterion on pressurized water reactor (PWR) vessel embrittlement as measured by nil-ductility reference temperature, termed RT_{PTS}. RT_{PTS} screening criteria values were set (using conservative fracture mechanics analysis techniques) for beltline axial welds, plates, and beltline circumferential weld seams for end-of-life plant operation. All PWR vessels in the United States have been required to evaluate vessel embrittlement in accordance with these criteria through EOL.

The Nuclear Regulatory Commission amended its regulations for light water nuclear power plants to change the procedure for calculating radiation embrittlement. The revised PTS Rule was published in the Federal Register, December 19, 1995 with an effective date of January 18, 1996[4]. This amendment makes the procedure for calculating RT_{PTS} values consistent with the methods given in Regulatory Guide 1.99, Revision 2.

The Rule establishes the following requirements for all domestic, operating PWRs:

- For each pressurized water nuclear power reactor for which an operating license has been issued, the licensee shall have projected values of RT_{PTS}, accepted by the NRC, for each reactor vessel beltline material for the EOL fluence of the material.

- The assessment of RT_{PTS} must use the calculation procedures given in the PTS Rule and must specify the bases for the projected value of RT_{PTS} for each beltline material. The report must specify the copper and nickel contents and the fluence values used in the calculation for each beltline material.

- This assessment must be updated whenever there is a significant change in projected values of RT_{PTS} or upon the request for a change in the expiration date for operation of the facility. Changes to RT_{PTS} values are significant if either the previous value or the current value, or both values, exceed the screening criteria prior to expiration of the operating license, including any renewal term (if applicable), for the plant.

- The RT_{PTS} screening criteria values for the beltline region are:

  270°F for plates, forgings and axial weld materials, and

  300°F for circumferential weld materials.

RT_{PTS} must be calculated for each vessel beltline material using a fluence value, f, which is the EOL fluence for the material. Equation 1 must be used to calculate values of RT_{NDT} for each weld, plate or forging in the reactor vessel beltline.

Equation 1: $RT_{NDT} = RT_{NDT(U)} + M + \Delta RT_{NDT}$

Where,

$RT_{NDT(U)}$ = Reference Temperature for a reactor vessel material in pre-service or non-irradiated condition

M = Margin to be added to account for uncertainties in values of $RT_{NDT(U)}$, copper and nickel contents, fluence, and calculation procedures. M is evaluated from Equation 2.

Equation 2: $M = \sqrt{\sigma_U{}^2 + \sigma_\Delta{}^2}$

σ_U is the standard deviation for $RT_{NDT(U)}$.

σU = 0°F when RTNDT(U) is a measured value.

σU = 17°F when RTNDT(U) is a generic value.

σ_Δ is the standard deviation for RT_{NDT}. σ_Δ is not to exceed one half of ΔRT_{NDT}

For plates and forgings:

σ_Δ = 17°F when surveillance capsule data is not used.

σ_Δ = 8.5°F when surveillance capsule data is used.

For welds:

σ_Δ = 28°F when surveillance capsule data is not used.

σ_Δ = 14°F when surveillance capsule data is used.

ΔRT_{NDT} is the mean value of the transition temperature shift, or change in RT_{NDT}, due to being irradiated and must be calculated using Equation 3.

Equation 3: $\Delta RT_{NDT} = (CF) * f^{(0.28-0.10 \log f)}$

CF (°F) is the chemistry factor, which is a function of copper and nickel content. CF is determined from Tables 1 and 2 of the PTS Rule (10 CFR 50.61). Surveillance data deemed credible must be used to determine a material-specific value of CF. A material-specific value of CF is determined in Equation 5.

f is the calculated neutron fluence, in units of 10^{19} n/cm^2 (E > 1.0 MeV), at the clad-base-metal interface on the inside surface of the vessel at the location where the material in question receives the highest fluence. The 32 EFPY EOL fluence and 48 EFPY license renewal fluences are used in calculating RT_{PTS}.

Equation 4 must be used for determining RT_{PTS} using Equation 3 with EOL fluence values for determining RT_{PTS}

$$\text{Equation 4:} \qquad RT_{PTS} = RT_{NDT(U)} + M + \Delta RT_{PTS}$$

To verify that RT_{NDT} for each vessel beltline material is a bounding value for the specific reactor vessel, licensees shall consider plant-specific information that could affect the level of embrittlement. This information includes but is not limited to reactor vessel operating temperature and any related surveillance program results. Results from the plant-specific surveillance program must be integrated into the RT_{NDT} estimate if the plant-specific surveillance data has been deemed credible. According to Regulatory Guide 1.99, Revision 2, in order to use surveillance data there has to be "...two or more credible data sets...from the reactor in question." A material-specific value of CF for surveillance materials is determined from Equation 5.

$$\text{Equation 5:} \qquad CF = \frac{\sum [A_i * f_i^{(0.28-0.10 \log f_i)}]}{\sum [f_i^{(0.56-0.20 \log f_i)}]}$$

In Equation 5, "A_i" is the measured value of ΔRT_{NDT} and "f_i" is the fluence for each surveillance data point. If there is clear evidence that the copper and nickel content of the surveillance weld differs from the vessel weld, i.e., differs from the average for the weld wire heat number associated with the vessel weld and the surveillance weld, the measured values of RT_{NDT} must be adjusted for differences in copper and nickel content by multiplying them by the ratio of the chemistry factor for the vessel material to that for the surveillance weld.

Emergency Response Guideline (ERG) Limits

Emergency Response Guideline (ERG) pressure-temperature limits[3] were developed to establish guidance for operator action in the event of an emergency, such as a PTS event. Generic categories of limits were developed for the guidelines based on the limiting inside surface RT_{NDT} at EOL. These generic categories were conservatively generated for the Westinghouse Owners Group (WOG) to be applicable to all Westinghouse plants.

The highest EOL RT_{NDT} for which the generic category ERG pressure-temperature limits were developed is 250°F for a longitudinal flaw and 300°F for a circumferential flaw. Thus, if the limiting vessel material has an EOL RT_{NDT} that exceeds 250°F for a longitudinal flaw or 300°F for a circumferential flaw, plant-specific ERG pressure-temperature limits are required.

A comparison of the current RT_{PTS} calculation (which is the EOL RT_{NDT} value (32 EFPY)) has been made to the uprated RT_{PTS} values for Byron Units 1 and 2 to determine if the applicable ERG category would change.

Upper Shelf Energy (USE)

The integrity of the reactor vessel may be affected by changes in system temperatures and pressures resulting from the power uprate. To address this consideration, an evaluation was performed to assess the impact of the power uprate on the USE values for all reactor vessel beltline materials in the Byron Units 1 and 2 reactor vessels. The USE assessment used the results of the neutron fluence evaluation for the power uprate and Figure 2 of Regulatory Guide 1.99, Revision 2[8], to determine if a further decrease in USE at EOL would occur due to the effects of the uprate on fluence projections.

Inlet Temperature

The basis of the equations and tables from Regulatory Guide 1.99, Revision 2 and 10 CFR 50.61, which are used in all the analyses described herein, is ASTM E900. Paragraph 1.1.4 of ASTM E900 stipulates that these equations are valid only in the temperature range of 530°F to 590°F. Therefore, the reactor vessel inlet temperature must be maintained within this range to retain validity of all existing analyses.

5.1.2.4 Acceptance Criteria

Surveillance Capsule Withdrawal Schedule

The surveillance capsule withdrawal schedules developed for Byron Units 1 and 2 following the uprating shall meet the requirements of ASTM-185-82. A satisfactory number of surveillance capsules shall remain in the reactor vessel so that further analysis, such as for life extension, can be completed as necessary.

Heatup and Cooldown Pressure-Temperature Limit Curves

If the fluence projections for Byron Units 1 and 2 increase from those used in the current heatup and cooldown curves, then new applicability dates must be calculated or new heatup and cooldown curves must be generated.

Pressurized Thermal Shock (PTS)

The uprated RT_{PTS} values for all beltline materials shall not exceed the screening criteria of the PTS Rule. Specifically, the RT_{PTS} values of the base metal (plates or forgings) shall not exceed 270°F, while the girth weld metal RT_{PTS} values shall not exceed 300°F through the end-of-license (32 EFPY) or license renewal (48 EFPY).

Emergency Response Guideline (ERG) Limits

The ERG limits shall be developed to establish guidelines for operator action in the event of an emergency, such as a PTS event.

Upper Shelf Energy (USE)

At power uprate conditions, the EOL USE values for all reactor beltline materials shall meet the requirements of 10 CFR 50, Appendix G[15].

Inlet Temperature

The inlet temperature must be maintained in the range of 530°F to 590°F for the current analyses described herein to remain valid.

5.1.2.5 Results

An evaluation of the impact of uprating on reactor vessel integrity was performed for Byron Units 1 and 2. Per Section 7.5, one can determine, by comparison to previous documentation, that the neutron fluence projections for Byron Units 1 and 2 after the 5% power uprating have increased from previous analyses.

Surveillance Capsule Withdrawal Schedule

Surveillance capsule withdrawal schedules have been developed for Byron Units 1 and 2 reactor vessels based on the uprated fluence projections and are presented in Tables 5.1.2-1 and 5.1.2-2. The only difference from those formally documented in WCAP-15123 Rev. 1[1] and WCAP-15176[2] are notes (d) and (e) in each table.

The surveillance capsule withdrawal schedules for Byron Units 1 and 2 are based on ASTM E185-82. Per ASTM E185-82, the withdrawal of a capsule is to be scheduled at the nearest vessel refueling outage to the calculated EFPY established for the particular surveillance capsule withdrawal. The capsules removed from the Byron Units 1 and 2 vessel to date meet the intent of ASTM E185-82 for the removal of the first capsule.

Heatup and Cooldown Pressure-Temperature Limit Curves

Since the uprated fluence projections are higher than those used in the current heatup and cooldown curves, new applicability dates must be calculated or new heatup and cooldown curves must be generated. ComEd chose to have new heatup and cooldown curves generated for Byron Units 1 and 2 at 22 and 32 EFPY. These heatup and cooldown curves, and the subsequent data points, are documented in WCAP-15391 (Unit 1) and WCAP-15392 (Unit 2). Summaries of the adjusted reference temperature values for Byron Units 1 and 2 are presented in Tables 5.1.2-3 and 5.1.2-4.

Pressurized Thermal Shock (PTS)

The 5% power uprating caused calculated neutron fluence values to increase for Byron Units 1 and 2. Therefore, the RT_{PTS} values that were generated for all beltline region materials of the Byron Units 1 and 2 reactor vessels for EOL (32 EFPY) and license renewal (48 EFPY) were recalculated and are presented in Tables 5.1.2-6 and 5.1.2-7 (also documented in

WCAP-15390[16] for Unit 1 and WCAP-15389[17] for Unit 2). These RT_{PTS} values increased primarily due to the 5% power uprating. However, other circumstances, such as updated chemistry factor values and data integration, also had an effect.

All RT_{PTS} values remain below the NRC screening criteria values using the projected fluence values through 32 and 48 EFPY for Byron Units 1 and 2.

Emergency Response Guideline (ERG) Limits

Based on the revised fluence projections after the power uprating, new RT_{PTS} values were determined to be 110°F and 113°F for Unit 1 (WCAP-15390[16]), and 116°F and 123°F for Unit 2 (WCAP-15389[17]). This is well below the 200°F maximum for Category I ERG limits (See Table 5.1.2-5). Thus, Byron Units 1 and 2 are not required to change ERG Plant Specific Limits for EOL and license renewal due to the 5% power uprating.

Upper Shelf Energy (USE)

The calculated neutron fluence values for the uprated condition at Byron Units 1 and 2 have increased. Therefore, the upper shelf energy (USE) values were recalculated and are presented in Tables 5.1.2-8 and 5.1.2-9. These tables demonstrate that all beltline materials still have a USE greater than 50 ft-lb through end of license (EOL, 32 FPY) as required by 10CFR50, Appendix G[15], after the 5% power uprating.

Inlet Temperature

Per Section 2.0, which contains the new PCWG parameters, the inlet temperature is within the range of 530°F to 590°F. Therefore, all current analyses remain valid.

5.1.2.6 Conclusions

The fluence projections under the uprated condition have increased from the previously documented values. Thus, new heatup and cooldown limit curves were developed and are presented in WCAP-15391 (Unit 1) and 15392 (Unit 2). The EOL and License Renewal ΔRT_{PTS} values at both Byron Units 1 and 2 were also recalculated but still remain well below the PTS screening criteria. The Byron Units 1 and 2 surveillance capsule withdrawal schedules from WCAPs-15123 Rev. 1 (Unit 1) and 15176 (Unit 2) have changed slightly in the fluence value of the standby capsules due to the increase in fluence projections. The ERG Limit remains at Category I after the 5% power uprating. Lastly, the Byron Units 1 and 2 USE values are still greater than 50 ft-lb through end of license.

It is concluded that the uprating program for Byron Units 1 and 2 will not have a significant impact on the reactor vessel integrity.

5.1.2.7 References

1. WCAP-15123, Rev. 1, "Analysis of Capsule W from the Commonwealth Edison Company Byron Unit 1 Reactor Vessel Radiation Surveillance Program," T. J. Laubham, et al., January, 1999.

2. WCAP-15176, "Analysis of Capsule X from the Commonwealth Edison Co. Byron Unit 2 Reactor Vessel Radiation Surveillance Program," T. J. Laubham, et al., March 1999.

3. Emergency Response Guidelines - Revision 1B, Westinghouse Owners Group, 2/28/92.

4. 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," Federal Register, Volume 60, No. 243, dated December 19, 1995, effective January 18, 1996.

5. ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," E706 (IF), in ASTM Standards, Section 3, American Society for Testing and Materials, Philadelphia, PA, 1993.

6. ASTM E900, "Standard Guide for Predicting Neutron Radiation Damage to Reactor Vessel Materials, E 706 (IIF)," Reapproved 1994.

7. 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," Federal Register, Volume 60, 243, dated December 19, 1995, effective January 18, 1996.

8. Regulatory Guide 1.99, Revision 2, May 1988, "Radiation Embrittlement of Reactor Vessel Materials."

9. WCAP-15391, "Byron Unit 1 Heatup and Cooldown Curves for Normal Operation," May 2000. (pending)

10. WCAP-15392, "Byron Unit 2 Heatup and Cooldown Curves for Normal Operation," May 2000. (pending)

11. WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," J. D. Andrachek, et al., January 1996.

12. ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1," February 26, 1999.

13. WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation For Operating PWR and BWR Plants," W. Bamford, et al., October 1999.

14. ASME Boiler and Pressure Vessel Code, Section XI, "Rule for Inservice Inspection of Nuclear Power Plant Components," Appendix G, "Fracture Toughness Criteria for Protection Against Failure," 1996.

15. 10 CFR 50, Appendix G, "Fracture Toughness Requirements," Federal Register, Volume 60, No. 243, dated December 19, 1995.

16. WCAP-15390, "Evaluation of Pressurized Thermal Shock for Byron Unit 1," T. J. Laubham, May 2000.

17. WCAP-15389, "Evaluation of Pressurized Thermal Shock for Byron Unit 2," T.J. Laubham, May 2000.

18. ASME Code Case N-588, "Attenuation to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels," Section XI, Division 1, Approved December 12, 1997.

Table 5.1.2-1 Byron Unit 1 Reactor Vessel Surveillance Capsule Withdrawal Schedule				
Capsule	**Location**	**Lead Factor[(a)]**	**Removal Time (EFPY)[(b)]**	**Fluence (n/cm^2,E>1.0 MeV)[(a)]**
U	58.5°	4.22	1.15	4.04×10^{18} (c)
X	238.5°	4.27	5.64	1.57×10^{19} (c)
W	121.5°	4.20	9.24	2.43×10^{19} (c)
Z	301.5°	4.20	Standby	(d)
V	61°	3.97	Standby	(e)
Y	241°	3.97	Standby	(e)

Notes:

(a) Updated in Capsule W dosimetry analysis, See WCAP-15123 Rev. 1.

(b) Effective Full Power Years (EFPY) from plant startup.

(c) Plant specific evaluation.

(d) This capsule will reach a fluence of approximately 3.03×10^{19} (Interpolated from 48 EFPY Peak Fluence, Section 7.5) at approximately 11.5 EFPY.

(e) These capsules will reach a fluence of approximately 3.03×10^{19} (Interpolated from 48 EFPY Peak Fluence, Section 7.5) at approximately 12.2 EFPY.

Table 5.1.2-2 Byron Unit 2 Reactor Vessel Surveillance Capsule Withdrawal Schedule				
Capsule	**Location**	**Lead Factor[(a)]**	**Removal Time (EFPY)[(b)]**	**Fluence (n/cm^2,E>1.0 MeV)[(a)]**
U	58.5°	4.40	1.15	4.05×10^{18} (c)
W	121.5°	4.25	4.634	1.27×10^{19} (c)
X	238.5°	4.25	8.57	2.30×10^{19} (c)
Z	301.5°	4.21	Standby	(d)
V	61°	3.97	Standby	(e)
Y	241°	3.97	Standby	(e)

Notes:

(a) Updated in Capsule X dosimetry analysis, See WCAP-15176.

(b) Effective Full Power Years (EFPY) from plant startup.

(c) Plant specific evaluation.

(d) This capsule will reach a fluence of approximately 3.10×10^{19} (48 EFPY Peak Fluence, Section 7.5) at approximately 11.4 EFPY.

(e) These capsules will reach a fluence of approximately 3.10×10^{19} (48 EFPY Peak Fluence, Section 7.5) at approximately 12.1 EFPY.

Table 5.1.2-3
Summary of Adjusted Reference Temperature (ART) Values at 1/4T and 3/4T Locations for Byron Unit 1

Material	22 EFPY		32 EFPY	
	1/4T ART	3/4T ART	1/4T ART	3/4T ART
Intermediate Shell Forging 5P-5933	89	75	95	80
- Using Surveillance Data	103[(a)]	94[(a)]	106[(a)]	97[(a)]
Lower Shell Forging 5P-5951	59	45	65	50
Circumferential Weld WF-336	71	41	82	52
- Using Credible Surveillance Data	60	42	67	48
Circumferential Weld WF-501	61	42	69	49
- Using Credible Surveillance Data from Braidwood 1 and 2	42	30	47	34
Nozzle Shell Forging 123J218	69	55	75	59

Notes:

(a) These ART values were used to generate the Byron Unit 1 heatup and cooldown curves

Table 5.1.2-4 Summary of Adjusted Reference Temperature (ART) Values at 1/4T and 3/4T Locations for Byron Unit 2				
	22 EFPY		32 EFPY	
Material	1/4T ART	3/4T ART	1/4T ART	3/4T ART
Intermediate Shell Forging [49D329/49C297]-1-1	18	7	22	11
Lower Shell Forging [49D330/49C298]-1-1	49	30	53	37
- Using Surveillance Data	15	5	17	9
Circumferential Weld WF-447	112	82	123	93
- Using Surveillance Data	101	82	107	89
Circumferential Weld WF-562	88	70	96	76
- Using Surveillance Data from Braidwood 1 and 2	70	59	75	63
Nozzle Shell Forging 4P-6107	46[(a)]	33[(a)]	52[(a)]	37[(a)]

Notes:

(a) These ART values were used to calculate the heatup and cooldown curves for Byron Unit 2. They were generated using the '96 App. G Methodology, and were more conservative than the curves generated using the limiting circumferential flaw ART values and Code Case N-588 Methodology.

Table 5.1.2-5 ERG Pressure-Temperature Limits[3]	
Applicable RT_{NDT} (ART) Value[(a)]	ERG Pressure-Temperature Limit Category
RT_{NDT} < 200°F	Category I
200°F < RT_{NDT} < 250°F	Category II
250°F < RT_{NDT} < 300°F	Category IIIb

Notes:

(a) Longitudinally oriented flaws are applicable only up to 250°F; circumferentially oriented flaws are applicable up to 300°F

Table 5.1.2-6 RT_{PTS} Calculation for Byron Unit 1 Beltline Region Materials at EOL (32 EFPY) and License Renewal (48 EFPY)		
Material	**32 EFPY RT_{PTS} (°F)**	**48 EFPY RT_{PTS} (°F)**
Intermediate Shell Forging 5P-5933	102	107
Intermediate Shell Forging 5P-5933 → Using S/C Data	110	113
Lower Shell Forging 5P-5951	72	77
Nozzle Shell Forging 123J218	83	90
Inter. to Lower Shell Circ. Weld Metal	90	95
Inter. to Lower Shell Circ. Weld Metal → Using S/C Data	76	82
Nozzle Shell to Inter. Shell Circ. Weld Metal	81	90
Nozzle Shell to Inter. Shell Circ. Weld Metal → Using S/C Data	54	60

Table 5.1.2-7 RT_{PTS} Calculation for Byron Unit 2 Beltline Region Materials at EOL (32 EFPY) and License Renewal (48 EFPY)		
Material	**32 EFPY RT_{PTS} (°F)**	**48 EFPY RT_{PTS} (°F)**
Inter. Shell Forging [49D329/49C297]-1-1	28	32
Lower Shell Forging [49D330/49C298]-1-1	58	62
Lower Shell Forging [49D330/49C298]-1-1 → Using S/C Data	19	21
Nozzle Shell Forging 4P-6107	61	68
Inter. to Lower Shell Circ. Weld Metal	130	136
Inter. to Lower Shell Circ. Weld Metal → Using S/C Data	116	123
Nozzle Shell to Inter. Shell Circ. Weld Metal	107	116
Nozzle Shell to Inter. Shell Circ. Weld Metal → Using S/C Data	82	88

Table 5.1.2-8 Byron Unit 1 Predicted End-of-License (32 EFPY) USE Calculations for all Beltline Region Materials	
Material	**Projected EOL USE (ft-lb)**
Intermediate Shell Forging 5P-5933 Using S/C Data	133
Lower Shell Forging 5P-5951	120
Nozzle Shell Forging 123J218	117
Intermediate to Lower Shell Forging Circ. Weld Seam WF336 (Heat 442002) Using S/C Data	69
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF501 (Heat 442011)	65

Table 5.1.2-9 Byron Unit 2 Predicted End-of-License (32 EFPY) USE Calculations for all Beltline Region Materials	
Material	**Projected EOL USE (ft-lb)**
Intermediate Shell Forging 49D329/49C297-1-1	119
Lower Shell Forging 49D330/49C298-1-1 Using S/C Data	107
Nozzle Shell Forging 4P-6107	133
Intermediate to Lower Shell Forging Circ. Weld Seam WF447 (Heat 442002) Using S/C Data	78
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF562 (Heat 442011)	69

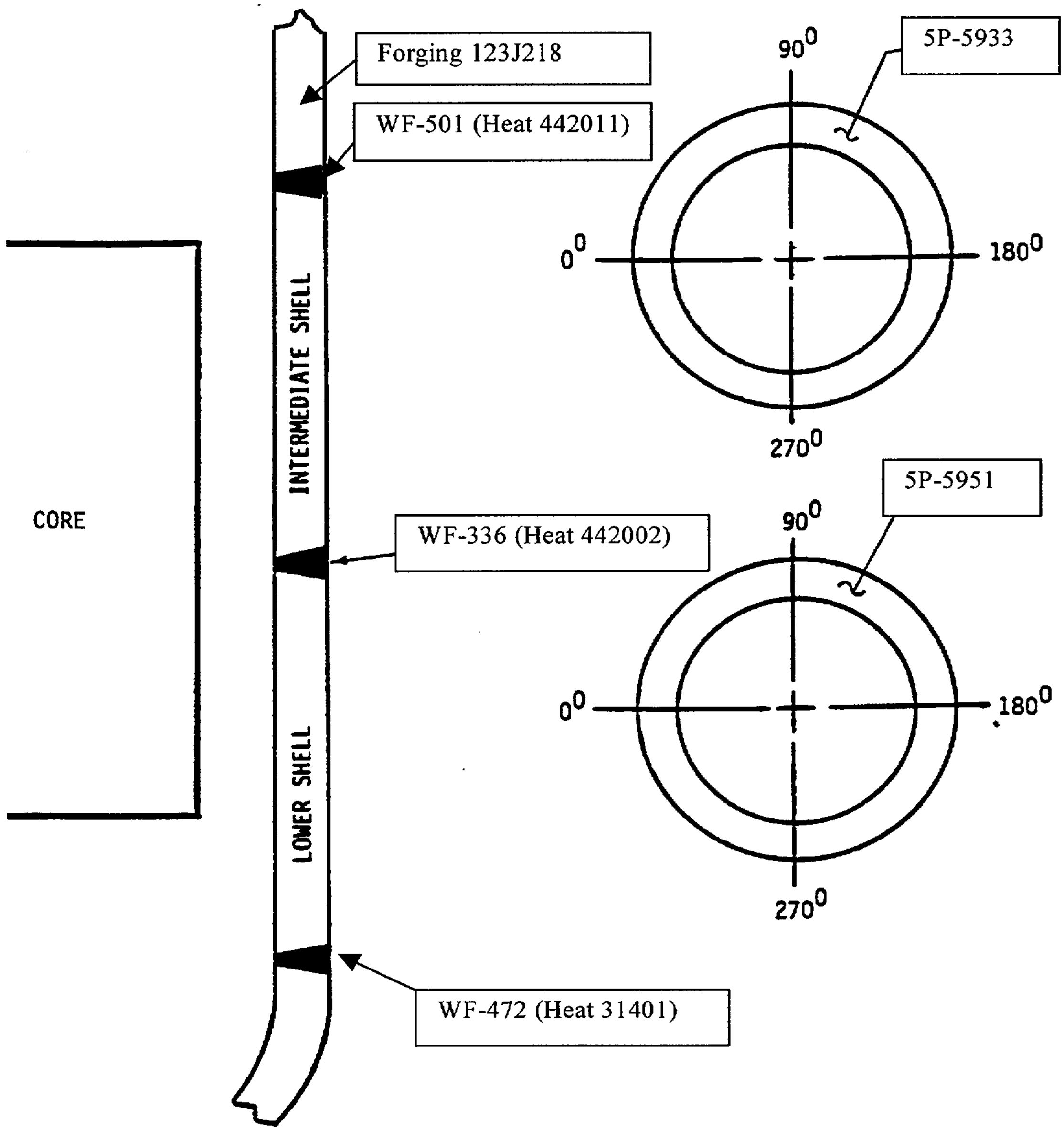

Figure 5.1.2-1

Identification and Location of Beltline Region Material for

Byron Unit 1 Reactor Vessel

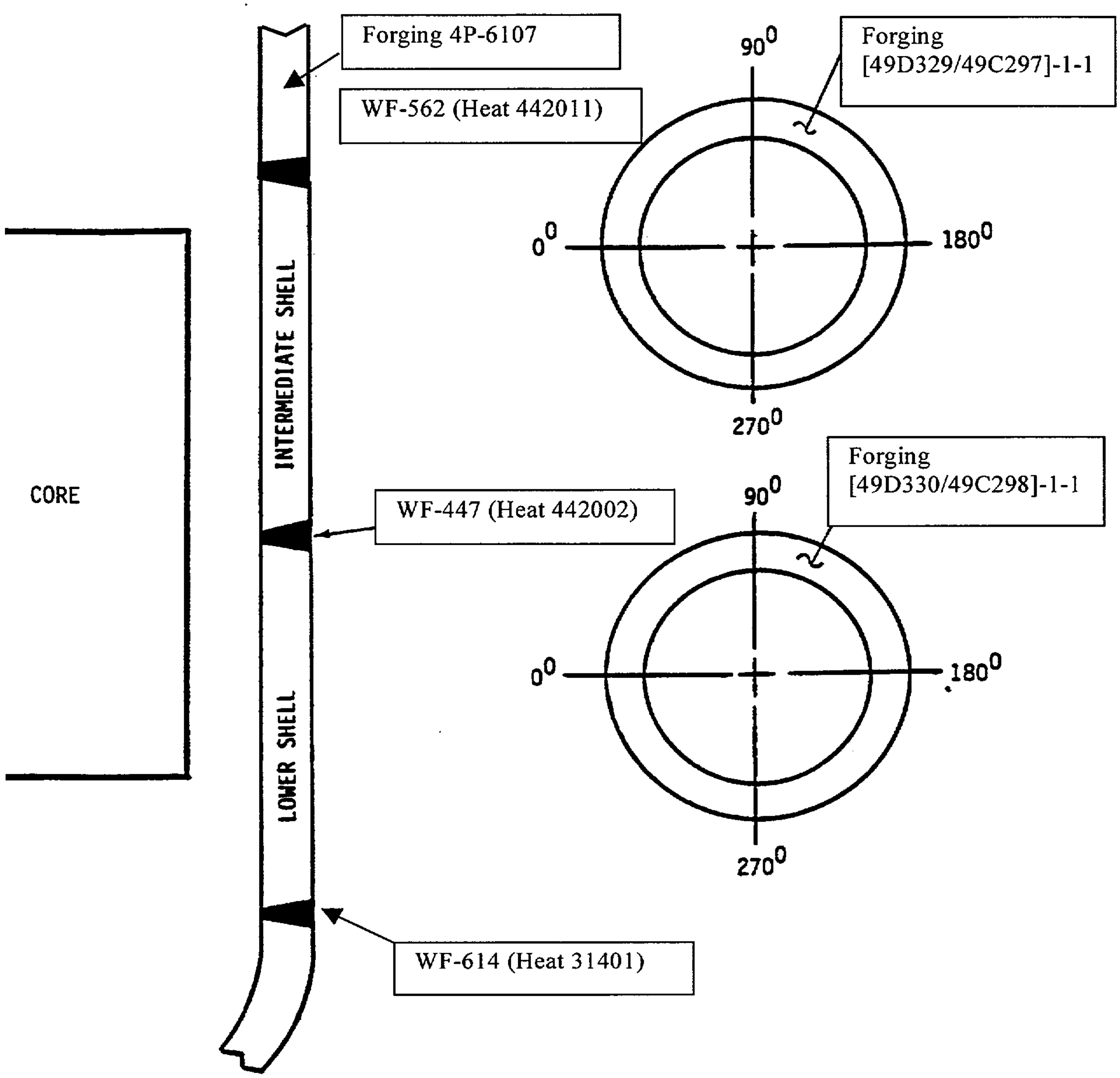

Figure 5.1.2-2

Identification and Location of Beltline Region Material for Byron Unit 2 Reactor Vessel

5.1.3 Reactor Vessel Integrity (Braidwood Units)

5.1.3.1 Introduction

Reactor vessel integrity is impacted by any changes in plant parameters that affect neutron fluence levels or temperature/pressure transients. The changes in neutron fluence resulting from the proposed Commonwealth Edison Company Braidwood Units 1 and 2 Thermal Power Uprate Program have been evaluated to determine the impact on reactor vessel integrity. This assessment included a review of the current material surveillance capsule withdrawal schedules[1, 2], development of new plant heatup and cooldown pressure-temperature (P-T) limit curves, review of the Emergency Response Guideline (ERG)[3] limits, and a revision to the RT_{PTS} values (10 CFR 50.61[4], known as the Pressurized Thermal Shock (PTS) Rule).

The most critical area, in terms of reactor vessel integrity, is the beltline region of the reactor vessel. The beltline region is defined in ASTM E185-82[5] as "the irradiated region of the reactor vessel (shell material including weld regions and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions that are predicted to experience sufficient neutron damage to warrant consideration in the selection of surveillance material". Figures 5.1.3-1 and 5.1.3-2 identify and indicate the location of all beltline region materials of the Braidwood Units 1 and 2 reactor vessels.

5.1.3.2 Input Parameters and Assumptions

Uprated Fluence Projections

Fluence projections (Section 7.5) on the vessel were evaluated for the uprated power level for input to the reactor vessel integrity evaluations. Typically, fluence values are used to evaluate end-of-life (EOL) transition temperature shift (EOL ΔRT_{NDT}) for development of surveillance capsule withdrawal schedules, determining EOL upper shelf energy (USE) values, adjusted reference temperature (ART) values for determining applicability of heatup and cooldown curves, ERG limits, and RT_{PTS} values.

Vessel Inlet Water Temperature

Due to the uprated power, the reactor vessel inlet temperature may also change. This updated inlet temperature (Section 2.0) has been reviewed to verify its compliance with ASTM E900[6], which is the basis for Regulatory Guide 1.99, Revision 2.

5.1.3.3 Description of Analysis/Evaluations

The reactor vessel integrity evaluation for the Braidwood Units 1 and 2 uprating included the following objectives:

1. Review reactor vessel surveillance capsule removal schedules for Braidwood Units 1 and 2 to determine if changes are required as a result of changes in vessel fluence due to the Thermal Power Uprate Program. This evaluation is consistent with the recommended practices of ASTM E185-82 and meets the requirements of Appendix H of 10 CFR Part 50[7].

2. Revised P-T limit curves have been generated for 16, 22 and 32 EFPY at Braidwood Unit 1 and 12, 16, 22 and 32 EFPY at Braidwood Unit 2. These curves account for the effects of uprated fluence projections and the latest Code changes as described later in this report. The methodology of Regulatory Guide 1.99, Revision 2[8], will be used in any required calculations.

3. Revised RT_{PTS} values have been determined accounting for the effects of uprated fluence projections in the Braidwood Units 1 and 2 reactor vessel at EOL (32 EFPY) and license renewal (48 EFPY). The current PTS Rule, 10 CFR Part 50.61[4], will be used to ensure that the screening criteria is met.

4. Review USE values at EOL for all reactor vessel beltline materials in the Braidwood Units 1 and 2 reactor vessels to assess the impact of uprated fluence projections.

5. Review reactor vessel inlet temperature for Braidwood Units 1 and 2 to verify that it maintains an acceptable level after the uprated condition takes affect.

Surveillance Capsule Withdrawal Schedules

A surveillance capsule withdrawal schedule is developed to periodically remove surveillance capsules from the reactor vessel. This procedure is followed to effectively monitor the condition of the reactor vessel materials under actual operating conditions. ASTM E185-82 defines the recommended number of surveillance capsules and the recommended withdrawal schedule, based on the vessel material predicted transition temperature shifts (ΔRT_{NDT}). The surveillance capsule withdrawal schedule is in terms of effective full-power years (EFPY) of plant operation, with a design life of 32 EFPY. Other factors that must be considered in establishing the surveillance capsule withdrawal schedule are maximum fluence values at the vessel inside surface and 1/4-thickness (1/4T) location.

The first surveillance capsule is typically scheduled to be withdrawn early in vessel life to verify initial predictions of surveillance material response to the actual radiation environment. It is generally removed when the predicted shift exceeds expected scatter by sufficient margin to be measurable. Normally, the capsule with the highest lead factor is withdrawn first. Early withdrawal also permits verification of the adequacy and conservatism of reactor vessel pressure-temperature operational limits.

The withdrawal schedule for the maximum number of surveillance capsules to be withdrawn is adjusted by the lead factor so that:

- exposure of the *second* surveillance capsule withdrawn occurs when the accumulated neutron fluence of the capsule corresponds to a value midway between that of the first and third capsules,
- exposure of the *third* surveillance capsule withdrawn does not exceed the peak EOL 1/4T fluence,
- exposure of the *fourth* surveillance capsule withdrawn does not exceed the peak EOL reactor vessel fluence, and
- exposure of the *fifth* surveillance capsule withdrawn does not exceed twice the peak EOL reactor vessel fluence.

Per ASTM E185-82, the four steps used for development of a surveillance capsule withdrawal schedule are as follows:

1. Estimate peak vessel inside surface fluence at EOL and the corresponding transition temperature shift (ΔRT_{NDT}). This identifies the number of capsules required.

2. Obtain the lead factor for each surveillance capsule relative to peak beltline fluence.

3. Calculate the EFPY for the capsule to reach peak vessel EOL fluence at the inside surface and 1/4T locations. These results are used to establish the withdrawal schedule for all but the first surveillance capsule.

4. Schedule surveillance capsule withdrawals at the nearest vessel refueling date.

A current surveillance capsule withdrawal schedule for Braidwood Units 1 and 2 reactor vessels is documented in WCAP-15316 Rev. 1[1] and WCAP-15369[2]. This schedule has been evaluated for the Power Uprate Project due to increased neutron fluence.

<u>Heatup and Cooldown Pressure-Temperature Limit Curves</u>

New heatup and cooldown curves for Braidwood Units 1 and 2 have been generated using uprated fluence projections at 16 / 22 / 32 EFPY (Unit 1) and 12 / 16 / 22 / 32 EFPY (Unit 2). These heatup and cooldown curves and the subsequent data points are documented in WCAP-15364[9] (Unit 1) and WCAP-15373[10] (Unit 2). These draft WCAPs will be finalized prior to the power uprate implementation.

The heatup and cooldown curves documented in these reports were generated using the most limiting ART values and the NRC approved methodology documented in WCAP-14040-NP-A, Revision 2[11], "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" with the following exceptions:

1. The fluence values used in this report are calculated fluence values, not best estimate fluence values.

2. The K_{Ic} critical stress intensities are used in place of K_{Ia} critical stress intensities. This methodology is taken from approved ASME Code Case N-640[12].

3. The reactor vessel flange temperature requirement has been eliminated. Justification has been provided in WCAP-15316[13].

4. The 1996 Version of Appendix G to Section XI[14] will be used rather than the 1989 version.

5. The methodology from approved ASME Code Case N-588[18] was used to consider circumferentially oriented flaws in development of P-T Curves.

Pressurized Thermal Shock (PTS)

Pressurized Thermal Shock (PTS), a limiting condition on reactor vessel integrity, may occur during a severe system transient such as a loss-of-coolant-accident (LOCA) or a steam line break. Such transients may challenge the integrity of a reactor vessel under the following conditions:

- severe overcooling of the inside surface of the vessel wall followed by high repressurization,
- significant degradation of vessel material toughness caused by radiation embrittlement, and
- the presence of a critical-size defect in the vessel wall.

The PTS concern arises if one of these transients acts on the beltline region of a reactor vessel where a reduced fracture resistance exists because of neutron irradiation. Such an event may produce the propagation of flaws postulated to exist near the inner wall surface, thereby potentially affecting the integrity of the vessel.

In 1985, the Nuclear Regulatory Commission (NRC) issued a formal ruling on PTS. It established screening criterion on pressurized water reactor (PWR) vessel embrittlement as measured by nil-ductility reference temperature, termed RT_{PTS}. RT_{PTS} screening criteria values were set (using conservative fracture mechanics analysis techniques) for beltline axial welds, plates, and beltline circumferential weld seams for end-of-life plant operation. All PWR vessels in the United States have been required to evaluate vessel embrittlement in accordance with these criteria through EOL.

The Nuclear Regulatory Commission amended its regulations for light water nuclear power plants to change the procedure for calculating radiation embrittlement. The revised PTS Rule was published in the Federal Register, December 19, 1995 with an effective date of January 18, 1996[4]. This amendment makes the procedure for calculating RT_{PTS} values consistent with the methods given in Regulatory Guide 1.99, Revision 2.

The Rule establishes the following requirements for all domestic, operating PWRs:

- For each pressurized water nuclear power reactor for which an operating license has been issued, the licensee shall have projected values of RT_{PTS}, accepted by the NRC, for each reactor vessel beltline material for the EOL fluence of the material.

- The assessment of RT_{PTS} must use the calculation procedures given in the PTS Rule and must specify the bases for the projected value of RT_{PTS} for each beltline material. The report must specify the copper and nickel contents and the fluence values used in the calculation for each beltline material.

- This assessment must be updated whenever there is a significant change in projected values of RT_{PTS} or upon the request for a change in the expiration date for operation of the facility. Changes to RT_{PTS} values are significant if either the previous value or the current value, or both values, exceed the screening criteria prior to expiration of the operating license, including any renewal term (if applicable), for the plant.

- The RT_{PTS} screening criteria values for the beltline region are:

  270°F for plates, forgings and axial weld materials, and

  300°F for circumferential weld materials.

RT_{PTS} must be calculated for each vessel beltline material using a fluence value, f, which is the EOL fluence for the material. Equation 1 must be used to calculate values of RT_{NDT} for each weld, plate or forging in the reactor vessel beltline.

$$\text{Equation 1:}\quad RT_{NDT} = RT_{NDT(U)} + M + \Delta RT_{NDT}$$

Where,

$RT_{NDT(U)}$ = Reference Temperature for a reactor vessel material in pre-service or non-irradiated condition

M = Margin added to account for uncertainties in values of RTNDT(U), copper and nickel contents, fluence, and calculation procedures. M is evaluated from Equation 2.

$$\text{Equation 2:} \quad M = \sqrt{\sigma_U^2 + \sigma_\Delta^2}$$

σ_U is the standard deviation for $RT_{NDT(U)}$.

σ_U = 0°F when $RT_{NDT(U)}$ is a measured value.

σ_U = 17°F when $RT_{NDT(U)}$ is a generic value.

σ_Δ is the standard deviation for RT_{NDT}. σ_Δ is not to exceed one half of ΔRT_{NDT}

For plates and forgings:

σ_Δ = 17°F when surveillance capsule data is not used.

σ_Δ = 8.5°F when surveillance capsule data is used.

For welds:

σ_Δ = 28°F when surveillance capsule data is not used.

σ_Δ = 14°F when surveillance capsule data is used.

ΔRT_{NDT} is the mean value of the transition temperature shift, or change in RT_{NDT}, due to being irradiated and must be calculated using Equation 3.

$$\text{Equation 3:} \quad \Delta RT_{NDT} = (CF) * f^{(0.28-0.10\log f)}$$

CF (°F) is the chemistry factor, which is a function of copper and nickel content. CF is determined from Tables 1 and 2 of the PTS Rule (10 CFR 50.61). Surveillance data deemed credible must be used to determine a material-specific value of CF. A material-specific value of CF is determined in Equation 5.

f is the calculated neutron fluence, in units of 10^{19} n/cm^2 (E > 1.0 MeV), at the clad-base-metal interface on the inside surface of the vessel at the location where the material in question receives the highest fluence. The 32 EFPY EOL fluence and 48 EFPY license renewal fluences are used in calculating RT_{PTS}.

Equation 4 must be used for determining RT_{PTS} using Equation 3 with EOL fluence values for determining RT_{PTS}

$$\text{Equation 4:}\quad RT_{PTS} = RT_{NDT(U)} + M + \Delta RT_{PTS}$$

To verify that RT_{NDT} for each vessel beltline material is a bounding value for the specific reactor vessel, licensees shall consider plant-specific information that could affect the level of embrittlement. This information includes but is not limited to reactor vessel operating temperature and any related surveillance program results. Results from the plant-specific surveillance program must be integrated into the RT_{NDT} estimate if the plant-specific surveillance data has been deemed credible. According to Regulatory Guide 1.99, Revision 2, in order to use surveillance data there has to be "...two or more credible data sets...from the reactor in question." A material-specific value of CF for surveillance materials is determined from Equation 5.

$$\text{Equation 5:}\quad CF = \frac{\sum[A_i * f_i^{(0.28-0.10\log f_i)}]}{\sum[f_i^{(0.56-0.20\log f_i)}]}$$

In Equation 5, "A_i" is the measured value of ΔRT_{NDT} and "f_i" is the fluence for each surveillance data point. If there is clear evidence that the copper and nickel content of the surveillance weld differs from the vessel weld, i.e., differs from the average for the weld wire heat number associated with the vessel weld and the surveillance weld, the measured values of RT_{NDT} must be adjusted for differences in copper and nickel content by multiplying them by the ratio of the chemistry factor for the vessel material to that for the surveillance weld.

Emergency Response Guideline (ERG) Limits

Emergency Response Guideline (ERG) pressure-temperature limits[3] were developed to establish guidance for operator action in the event of an emergency, such as a PTS event. Generic categories of limits were developed for the guidelines based on the limiting inside surface RT_{NDT} at EOL. These generic categories were conservatively generated for the Westinghouse Owners Group (WOG) to be applicable to all Westinghouse plants.

The highest EOL RT_{NDT} for which the generic category ERG pressure-temperature limits were developed is 250°F for a longitudinal flaw and 300°F for a circumferential flaw. Thus, if the limiting vessel material has an EOL RT_{NDT} that exceeds 250°F for a longitudinal flaw or 300°F for a circumferential flaw, plant-specific ERG pressure-temperature limits are required.

A comparison of the current RT_{PTS} calculation (which is the EOL RT_{NDT} value (32 EFPY)) has been made to the uprated RT_{PTS} values for Braidwood Units 1 and 2 to determine if the applicable ERG category would change.

Upper Shelf Energy (USE)

The integrity of the reactor vessel may be affected by changes in system temperatures and pressures resulting from the power uprate. To address this consideration, an evaluation was performed to assess the impact of the power uprate on the USE values for all reactor vessel beltline materials in the Braidwood Units 1 and 2 reactor vessels. The USE assessment used the results of the neutron fluence evaluation for the power uprate and Figure 2 of Regulatory Guide 1.99, Revision 2[8] to determine if a further decrease in USE at EOL would occur due to the effects of the uprate on fluence projections.

Inlet Temperature

The basis of the equations and tables from Regulatory Guide 1.99, Revision 2 and 10 CFR 50.61, which are used in all the analyses described herein, is ASTM E900. Paragraph 1.1.4 of ASTM E900 stipulates that these equations are valid only in the temperature range of 530°F to 590°F. Therefore, the reactor vessel inlet temperature must be maintained within this range to retain validity of all existing analyses.

5.1.3.4 Acceptance Criteria for Analyses/Evaluations

Surveillance Capsule Withdrawal Schedule

The proposed surveillance capsule withdrawal schedules developed for Braidwood Units 1 and 2 following the uprating shall meet the requirements of ASTM-185-82. A satisfactory number of surveillance capsules shall remain in the reactor vessel so that further analysis, such as for life extension, can be completed as necessary.

Heatup and Cooldown Pressure-Temperature Limit Curves

If the fluence projections for Braidwood Units 1 and 2 increase from that used in the current heatup and cooldown curves, then new applicability dates must be calculated or new heatup and cooldown curves must be generated.

Pressurized Thermal Shock (PTS)

The uprated RT_{PTS} values for all beltline materials shall not exceed the screening criteria of the PTS Rule. Specifically, the RT_{PTS} values of the base metal (plates or forgings) shall not exceed 270°F, while the girth weld metal RT_{PTS} values shall not exceed 300°F through the end-of-license (32 EFPY) or license renewal (48 EFPY).

Emergency Response Guideline (ERG) Limits

The ERG limits shall be developed to establish guidelines for operator action in the event of an emergency, such as a PTS event.

Upper Shelf Energy (USE)

At power uprate conditions, the EOL USE values for all reactor beltline materials shall meet the requirements of 10 CFR 50, Appendix G[15].

Inlet Temperature

The reactor vessel inlet temperature must be maintained in the range of 530°F to 590°F for the current analyses described herein to remain valid.

5.1.3.5 Results

An evaluation of the impact of uprating on reactor vessel integrity was performed for Braidwood Units 1 and 2. Per Section 7.5, one can determine, by comparison to previous documentation, that the neutron fluence projections for Braidwood Units 1 and 2 after the 5% power uprating have increased from previous analyses.

Surveillance Capsule Withdrawal Schedule

Surveillance capsule withdrawal schedules have been developed for Braidwood Units 1 and 2 reactor vessels based on the uprated fluence projections and are presented in Tables 5.1.3-1 and 5.1.3-2. The withdrawal schedules are formally documented in WCAP-15316 Rev. 1[1] and WCAP-15369[2].

The surveillance capsule withdrawal schedules for Braidwood Units 1 and 2 are based on ASTM E185-82. Per ASTM E185-82, the withdrawal of a capsule is to be scheduled at the nearest vessel refueling outage to the calculated EFPY established for the particular surveillance capsule withdrawal. The capsules removed from the Braidwood Units 1 and 2 vessel to date meet the intent of ASTM E185-82 for the removal of the first capsule.

Heatup and Cooldown Pressure-Temperature Limit Curves

Since the uprated fluence projections are higher than those used in the current heatup and cooldown curves, new applicability dates must be calculated or new heatup and cooldown curves must be generated. ComEd chose to have new heatup and cooldown curves generated for Braidwood Units 1 and 2 at 16/22/32 EFPY (Unit 1) and 12/16/22/32 EFPY (Unit 2). These heatup and cooldown curves, and the subsequent data points, are documented in WCAP-15364 (Unit 1) and WCAP-15373 (Unit 2). Summaries of the adjusted reference temperature values for Braidwood Units 1 and 2 are presented in Tables 5.1.3-3 and 5.1.3-4.

Pressurized Thermal Shock (PTS)

The 5% power uprating caused calculated neutron fluence values to increase for Braidwood Units 1 and 2. Therefore, the RT_{PTS} values that were generated for all beltline region materials of the Braidwood Units 1 and 2 reactor vessels for EOL (32 EFPY) and license renewal (48 EFPY) were recalculated and are presented in Tables 5.1.3-6 and 5.1.3-7 (also documented in WCAP-15365[16] for Unit 1 and WCAP-15381[17] for Unit 2). These RT_{PTS} values increased primarily due to the 5% power uprating. However, other circumstances, such as updated chemistry factor values and data integration, also had an effect.

All RT_{PTS} values remain below the NRC screening criteria values using the projected fluence values through 32 and 48 EFPY for Braidwood Units 1 and 2.

Emergency Response Guideline (ERG) Limits

Based on the revised fluence projections after the power uprating, new RT_{PTS} values were determined to be 99°F and 101°F for Unit 1 (WCAP-15365[16]), and 98°F and 101°F for Unit 2 (WCAP-15381[17]). This is well below the 200°F maximum for Category I ERG limits (See Table 5.1.3-5). Thus, Braidwood Units 1 and 2 are not required to change ERG Plant Specific Limits for EOL and license renewal due to the 5% power uprating.

Upper Shelf Energy (USE)

The calculated neutron fluence values for the uprated condition at Braidwood Units 1 and 2 have increased. Therefore, the upper shelf energy (USE) values were recalculated and are presented in Tables 5.1.3-8 and 5.1.3-9. These tables demonstrate that all beltline materials still have a USE greater than 50 ft-lb through end of license (EOL, 32 EFPY) as required by 10 CFR 50, Appendix G[18], after the 5% power uprating.

Vessel Inlet Water Temperature

Per Section 2.0, which contains the new PCWG parameters, the inlet temperature is within the range from 542°F to 555.7°F. This inlet temperature is within the range of 530°F to 590°F. Therefore, all current analyses remain valid.

5.1.3.6 Conclusions

The fluence projections under the uprated condition have increased from the previously documented values. Thus, new heatup and cooldown limit curves were developed and are presented in WCAP-15364 (Unit 1) and 15373 (Unit 2). The EOL and License Renewal ΔRT_{PTS} values at both Braidwood Units 1 and 2 were also recalculated but still remain well below the PTS screening criteria. The Braidwood Units 1 and 2 surveillance capsule withdrawal schedules from WCAPs-15316 Rev. 1 (Unit 1) and 15381 (Unit 2) still remain valid despite the increase in fluence projections. This is the same for the ERG Limit, which remains at Category I after the 5% power uprating. Lastly, the Braidwood Units 1 and 2 USE values are still greater than 50 ft-lb through end of license.

It is concluded that the uprating program for Braidwood Units 1 and 2 will not have a significant impact on the reactor vessel integrity.

5.1.3.7 References

1. WCAP-15316, Rev. 1, "Analysis of Capsule W from the Commonwealth Edison Company Braidwood Unit 1 Reactor Vessel Radiation Surveillance Program," Ed Terek, et al., December, 1999.

2. WCAP-15369, "Analysis of Capsule W from the Commonwealth Edison Co. Braidwood Unit 2 Reactor Vessel Radiation Surveillance Program," T. J. Laubham, et al., March 2000.

3. Emergency Response Guidelines - Revision 1B, Westinghouse Owners Group, 2/28/92.

4. 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," Federal Register, Volume 60, No. 243, dated December 19, 1995, effective January 18, 1996.

5. ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," E706 (IF), in ASTM Standards, Section 3, American Society for Testing and Materials, Philadelphia, PA, 1993.

6. ASTM E900, "Standard Guide for Predicting Neutron Radiation Damage to Reactor Vessel Materials, E 706 (IIF)," Reapproved 1994.

7. 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," Federal Register, Volume 60, 243, dated December 19, 1995, effective January 18, 1996.

8. Regulatory Guide 1.99, Revision 2, May 1988, "Radiation Embrittlement of Reactor Vessel Materials."

9. WCAP-15364 (draft), "Braidwood Unit 1 Heatup and Cooldown Curves for Normal Operation," 1999.

10. WCAP-15373 (draft), "Braidwood Unit 2 Heatup and Cooldown Curves for Normal Operation," 2000.

11. WCAP-14040-NP-A, Revision 2, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," J. D. Andrachek, et al., January 1996.

12. ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1," February 26, 1999.

13. WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants," W. Bamford, et al., October 1999.

14. ASME Boiler and Pressure Vessel Code, Section XI, "Rule for Inservice Inspection of Nuclear Power Plant Components," Appendix G, "Fracture Toughness Criteria for Protection Against Failure," 1996.

15. 10 CFR 50, Appendix G, "Fracture Toughness Requirements," Federal Register, Volume 60, No. 243, dated December 19, 1995.

16. WCAP-15365, "Evaluation of Pressurized Thermal Shock for Braidwood Unit 1," E. Terek, 1999.

17. WCAP-15381, "Evaluation of Pressurized Thermal Shock for Braidwood Unit 2," T. J. Laubham, 2000.

18. ASME Code Case N-588, "Attenuation to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels," Section XI, Division 1, Approved December 12, 1997.

Table 5.1.3-1

Braidwood Unit 1 Reactor Vessel Surveillance Capsule Withdrawal Schedule

Capsule	Location	Lead Factor[(a)]	Removal Time (EFPY)[(b)]	Fluence (n/cm^2,E>1.0 MeV)[(a)]
U	58.5°	4.37	1.10	3.87×10^{18} (c)
X	238.5°	4.24	4.234	1.24×10^{19} (c)
W	121.5°	4.19	7.61	2.09×10^{19} (c)
Z	301.5°	4.20	Standby	(d)
V	61.0°	3.92	Standby	(e)
Y	241.0°	3.92	Standby	(e)

Notes:

(a) Updated in Capsule W dosimetry analysis, See WCAP-15316 Rev. 1.

(b) Effective Full Power Years (EFPY) from plant startup.

(c) Plant specific evaluation.

(d) This capsule will reach a fluence of approximately 2.94×10^{19} (48 EFPY Peak Fluence) at approximately 12 EFPY.

(e) These capsules will reach a fluence of approximately 2.94×10^{19} (48 EFPY Peak Fluence) at approximately 13 EFPY.

Table 5.1.3-2 Braidwood Unit 2 Reactor Vessel Surveillance Capsule Withdrawal Schedule				
Capsule	**Location**	**Lead Factor[a]**	**Removal Time (EFPY)[b]**	**Fluence (n/cm^2,E>1.0 MeV)[a]**
U	58.5°	4.41	1.15	4.00×10^{18} (c)
X	238.5°	3.85	4.215	1.23×10^{19} (c)
W	121.5°	4.17	8.53	2.25×10^{19} (c)
Z	301.5°	4.17	Standby	(d)
V	61.0°	3.92	Standby	(e)
Y	241.0°	3.92	Standby	(e)

Notes:

(a) Updated in Capsule W dosimetry analysis, See WCAP-15369.

(b) Effective Full Power Years (EFPY) from plant startup.

(c) Plant specific evaluation.

(d) This capsule will reach a fluence of approximately 2.94×10^{19} (48 EFPY Peak Fluence) at approximately 12 EFPY.

(e) These capsules will reach a fluence of approximately 2.94×10^{19} (48 EFPY Peak Fluence) at approximately 13 EFPY.

Table 5.1.3-3
Summary of Adjusted Reference Temperature (ART) Values at 1/4T and 3/4T Locations for Braidwood Unit 1

Material	16 EFPY		22 EFPY		32 EFPY	
	1/4T ART	3/4T ART	1/4T ART	3/4T ART	1/4T ART	3/4T ART
Intermediate Shell Forging 49D383-1/49C344-1	24	7	29	12	36	18
Lower Shell Forging 49D867-1/49C813-1	34	17	39	22	46	28
- Using Surveillance Data	21	8	26	12	31	17
Circumferential Weld WF-562	109	88	117	95	126	103
- Using Surveillance Data	84[(a)]	70[(a)]	89[(a)]	74[(a)]	94[(a)]	79[(a)]
Nozzle Shell Forging 5P-7016	39[(a)]	28[(a)]	43[(a)]	31[(a)]	48[(a)]	35[(a)]
Circumferential Weld WF-645	34	12	43	18	53	26

Notes:
(a) These ART values were used to generate the Braidwood Unit 1 heatup and cooldown curves

Table 5.1.3-4 Summary of Adjusted Reference Temperature (ART) Values at 1/4T and 3/4T Locations for Braidwood Unit 2								
	12 EFPY		16 EFPY		22 EFPY		32 EFPY	
Material	**1/4T ART**	**3/4T ART**	**1/4T ART**	**3/4T ART**	**1/4T ART**	**3/4T ART**	**1/4T ART**	**3/4T ART**
Intermediate Shell Forging 49D963-1/49C904-1	1	-9	4	-7	8	-3	12	0
Lower Shell Forging 50D102-1/50C97-1	27	8	33	13	39	19	43	26
- Using Surveillance Data	14	11	15	12	16	13	18	14
Circumferential Weld WF-562	103	82	109	87	117	94	125	102
- Using Surveillance Data	79[(a)]	66[(a)]	83[(a)]	70[(a)]	88[(a)]	74[(a)]	93[(a)]	79[(a)]
Nozzle Shell Forging 5P-7056	54[(a)]	45[(a)]	58[(a)]	47[(a)]	62[(a)]	50[(a)]	67[(a)]	54[(a)]
Circumferential Weld WF-645	26	5	33	10	41	16	51	25

Notes:
(a) These ART values were used to generate the Braidwood Unit 2 heatup and cooldown curves.

Table 5.1.3.5 ERG Pressure-Temperature Limits[2]	
Applicable RT_{NDT} (ART) Value[(a)]	**ERG Pressure-Temperature Limit Category**
RT_{NDT} < 200°F	Category I
200°F < RT_{NDT} < 250°F	Category II
250°F < RT_{NDT} < 300°F	Category IIIb

Notes:
(a) Longitudinally oriented flaws are applicable only up to 250°F, the circumferentially oriented flaws are applicable up to 300°F

Table 5.1.3-6

RT_{PTS} Calculation for Braidwood Unit 1 Beltline Region Materials at EOL (32 EFPY) and License Renewal (48 EFPY)

Material	32 EFPY RT_{PTS} (°F)	48 EFPY RT_{PTS} (°F)
Intermediate Shell Forging 49D383-1/49C344-1	41	44
Lower Shell Forging 49D867-1/49C813-1	51	54
- Using Surveillance Data	26	42
Circumferential Weld WF-562	138	146
- Using Surveillance Data	99	101
Nozzle Shell Forging 5P-7016	55	60
Circumferential Weld WF-645	54	64

Table 5.1.3-7

RT_{PTS} Calculation for Braidwood Unit 2 Beltline Region Materials at EOL (32 EFPY) and License Renewal (48 EFPY)

Material	32 EFPY RT_{PTS} (°F)	48 EFPY RT_{PTS} (°F)
Intermediate Shell Forging 49D963-1/49C904-1	17	22
Lower Shell Forging 50D102-1/50C97-1	48	52
- Using Surveillance Data	19	21
Circumferential Weld WF-562	136	145
- Using Surveillance Data	98	101
Nozzle Shell Forging 5P-7056	74	80
Circumferential Weld WF-645	66	78

Table 5.1.3-8 Braidwood Unit 1 Predicted End-of-License (32 EFPY) USE Calculations for all Beltline Region Materials	
Material	**Projected EOL USE (ft-lb)**
Intermediate Shell Forging 49D383-1/49C344-1	98
Lower Shell Forging 49D867-1/49C813-1 (Using S/C Data)	117
Nozzle Shell Forging 5P-7016	124
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011) Using S/C Data	75
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-645 (Heat H4498)	72

Table 5.1.3-9 Braidwood Unit 2 Predicted End-of-License (32 EFPY) USE Calculations for all Beltline Region Materials	
Material	**Projected EOL USE (ft-lb)**
Intermediate Shell Forging 49D963-1/49C904-1	95
Lower Shell Forging 50D102-1/50C97-1 Using S/C Data	125
Nozzle Shell Forging 5P-7056	92
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011) Using S/C Data	67
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-645 (Heat H4498)	72

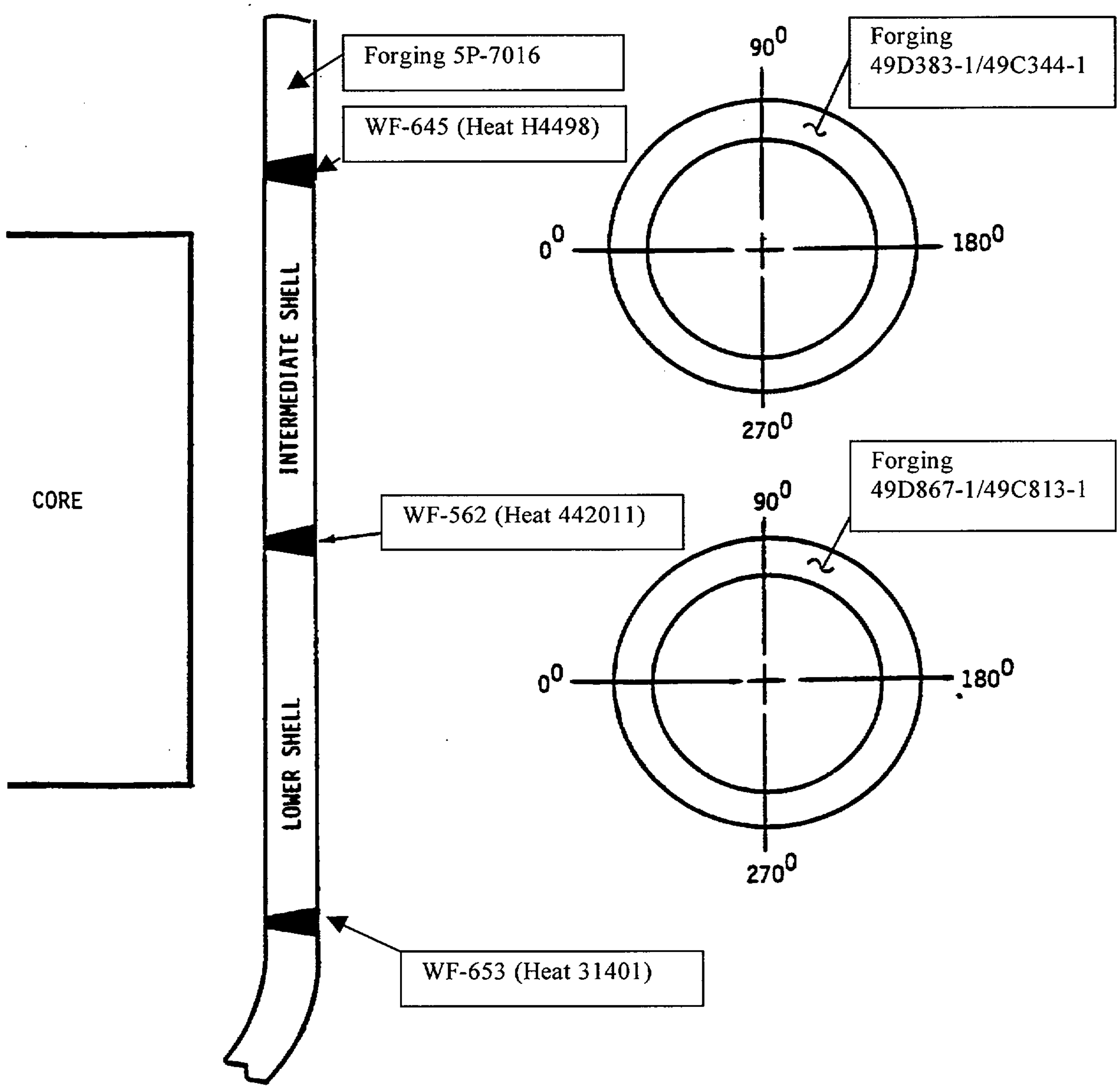

Figure 5.1.3-1
Identification and Location of Beltline Region Material for the Braidwood Unit 1 Reactor Vessel

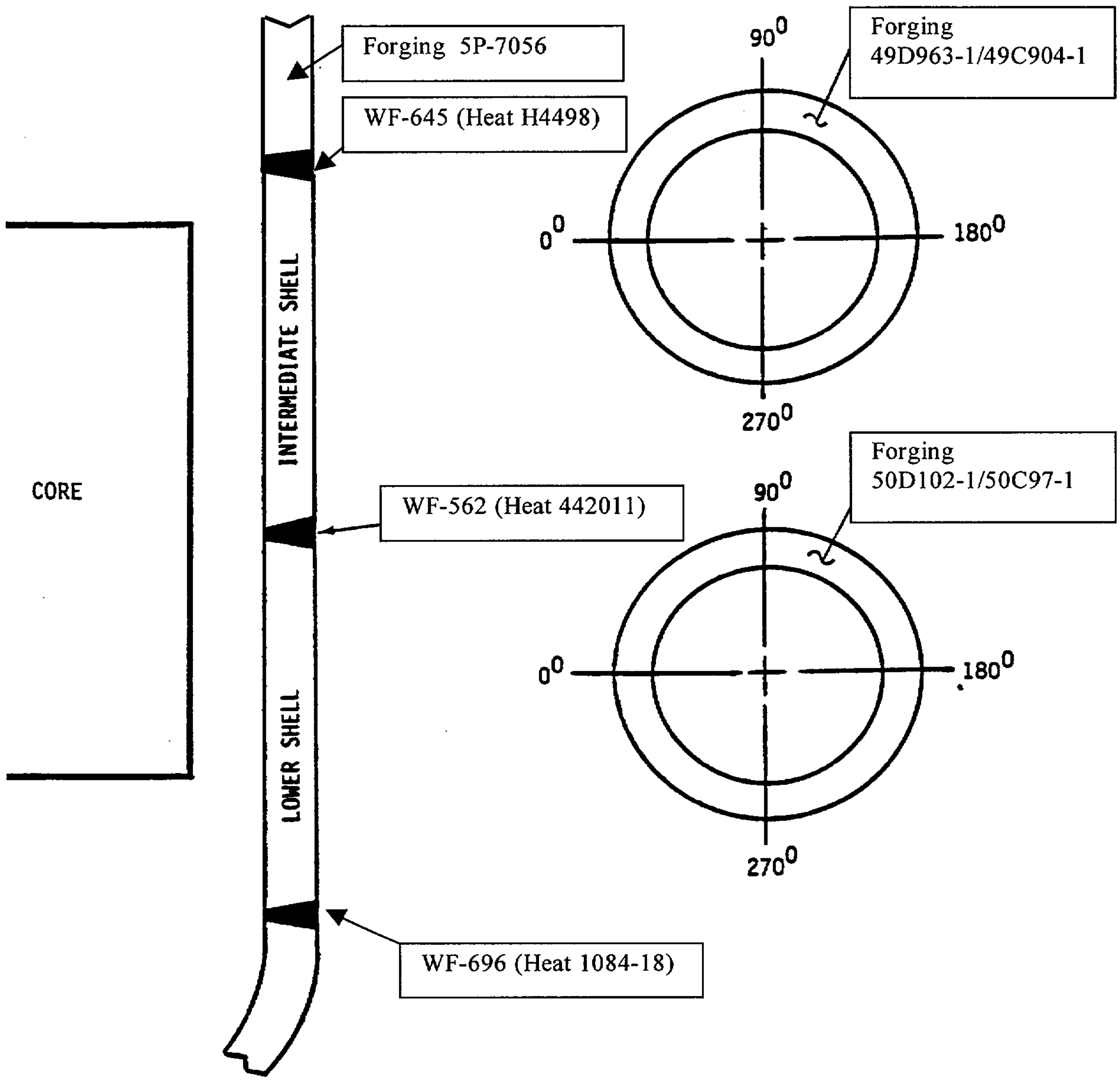

Figure 5.1.3-2
Identification and Location of Beltline Region Material for the Braidwood Unit 2 Reactor Vessel

5.2 Reactor Pressure Vessel System

The Reactor Pressure Vessel (RPV) System consists of the reactor vessel, reactor internals, fuel, and control rod drive mechanisms. The reactor internals function to support and orient the reactor core fuel assemblies and control rod assemblies, absorb control rod assembly dynamic loads, and transmit these and other loads to the reactor vessel. The reactor vessel internal components also function to direct coolant flow through the fuel assemblies, provide adequate cooling flow to the various internal structures, and support in-core instrumentation. They are designed to withstand forces due to structural deadweight, preload of fuel assemblies, control rod assembly dynamic loads, vibratory loads, earthquake accelerations, and Loss-of-Coolant Accident (LOCA) loads.

Evaluation of the uprated conditions requires that the reactor vessel/internals/fuel system interface be assessed to assure compatibility and that the structural integrity of the reactor vessel/internals/fuel system is not adversely affected. In addition, thermal-hydraulic analyses are required to determine plant-specific core bypass flows, pressure drops, and upper head temperatures for input to LOCA and non-LOCA safety analyses and NSSS performance evaluations.

Generally, the areas of concern most affected by changes in system operating conditions are:

- Reactor internals system thermal/hydraulic performance
- Rod control cluster assembly (RCCA) scram performance
- Reactor internals system structural response and integrity

5.2.1 Thermal/Hydraulic System Evaluations

5.2.1.1 System Pressure Losses

The principal Reactor Coolant System (RCS) flow route through the reactor pressure vessel system at the Byron and Braidwood units begins at the four inlet nozzles. At this point, flow turns downward through the reactor vessel core barrel annulus. After passing through this downcomer region, flow enters the lower reactor vessel dome region. This region is occupied by the internals energy absorber structure, lower support columns, bottom-mounted instrumentation columns, and supporting tie plates. From this region, flow passes upward through the lower core plate and into the core region. After passing up through the core, the coolant flows into the upper plenum, turns, and exits the reactor vessel through the four outlet nozzles. Note that support columns and RCCA guide columns occupy the upper plenum region.

A key area in evaluation of core performance is the determination of hydraulic behavior of coolant flow within the reactor internals system, i.e., vessel pressure drops, core bypass flows, RPV fluid temperatures and hydraulic lift forces. The analyses determined the distribution of pressure and flow within the reactor vessel, internals and the reactor core for the uprated conditions.

5.2.1.2 Bypass Flow Analysis

Bypass flow is the total amount of reactor coolant flow bypassing the core region and is not considered effective in the core heat transfer process. Since variations in the size of the bypass flow paths, such as gaps at the outlet nozzles and core barrel, occur during manufacturing, or changes due to different fuel assembly designs or changes in RCS conditions, plant-specific as-built dimensions are used to demonstrate that bypass flow limits are not violated. Analyses are performed to determine core bypass flow values. This ensures that either the design bypass flow limit for the plant will not be exceeded or a revised design core bypass flow is required. If required, a revised design core bypass flow is developed, and core bypass flow values are re-analyzed. Note that since the as-built information is different between Byron and Braidwood Unit 1 and Unit 2, each unit has a different best estimate core bypass flow value.

The present design core bypass flow limit is 8.3% (with thimble plugs removed) of the total reactor vessel flow for each Byron and Braidwood unit. The purpose of this evaluation is to ensure that the design value of 8.3% can be maintained at the uprated RCS conditions. The principal core bypass flow paths are the:

Baffle/Barrel Region

Vessel Head Cooling Spray Nozzles

Core Barrel – Reactor Vessel Outlet Nozzle Gap

Fuel Assembly – Baffle Plate Cavity Gap

Fuel Assembly Thimble Tubes

Fuel assembly hydraulic characteristics and system parameters, such as reactor coolant inlet temperature, pressure, and flows were used in conjunction with the THRIVE computer code to determine the impact of uprated conditions on total core bypass flow. THRIVE solves the mass and energy balances for the reactor internals fluid system and has been used in the original design basis and for other nuclear plant licensing analysis (Reference 1). The total best estimate core bypass flow values (including uncertainties) were determined to be 6.83% and 6.6% for Byron Units 1 and 2 and 7.39% and 6.84% for Braidwood Units 1 and 2, respectively.

5.2.1.3 Hydraulic Lift Forces

The reactor internals hold-down spring is essentially a large diameter belleville-type spring of rectangular cross section. The purpose of this spring is to maintain a net clamping force between 1) the reactor vessel head and upper internals flanges, and 2) the reactor vessel shell flange and core barrel flange of the internals.

An evaluation was performed to determine hydraulic lift forces on the various reactor internal components. The results of the calculations show that, with the uprated RCS conditions for Byron and Braidwood Units 1 and 2, the reactor internals will remain seated and stable for all conditions.

5.2.1.4 RCCA Scram Performance Evaluation

The RCCAs represent the interface between the fuel assemblies and other internal components. An evaluation was performed to determine the potential impact, due to power uprating at Byron and Braidwood Units 1 and 2, on RCCA scram characteristics used in the FSAR accident analyses. This analysis is based on 17x17 VANTAGE 5 (or VANTAGE+) Fuel Assemblies presently used in the Byron and Braidwood units.

Calculations were performed, which indicated that, for even the most severe case, the current maximum drop time-to-dashpot entry of 2.7 seconds for the Byron and Braidwood Units 1 and 2 remains conservatively applicable for accident analyses.

5.2.1.5 Momentum Flux and Fuel Rod Stability

Baffle jetting is a hydraulically induced instability or vibration of fuel rods caused by a high velocity jet of water. This jet is created by high-pressure water being forced through gaps between the baffle plates surrounding the core. To guard against fuel rod failures from flow induced vibration, the crossflow from baffle joint gaps must be limited to a specific momentum flux, V^2h, that is, the product of the gap width, h, and the square of the baffle joint jet velocity, V^2.

To assess the impact of the uprated RCS conditions on baffle jetting margins of safety at the Byron and Braidwood units, the ratio of the margins of safety between the present plant configuration and the uprated configuration has been determined. In this evaluation, it was assumed that there were no degraded bolts in the baffle/barrel region. The results show that, based on mechanical design flow, the margins of safety for momentum flux at uprated conditions do not change significantly from those at the present conditions.

5.2.2 Mechanical System Evaluations

The RCS mechanical response, subjected to auxiliary line breaks of a LOCA transient, is performed in three steps. First, the RCS is analyzed for the effects of loads induced by normal operation, which includes thermal, pressure, and deadweight effects. From this analysis, the mechanical forces acting on the RPV, which would result from release of equilibrium forces at the break locations, are obtained. In the second step, the loop mechanical loads and reactor internals hydraulic forces are simultaneously applied, and the RPV displacements due to the LOCA are calculated. Finally, the structural integrity of the reactor coolant loop and component supports, to deal with the LOCA, are evaluated by applying the calculated reactor vessel displacements to a mathematical model of the reactor coolant loop. Thus, the effects of vessel displacements upon the loop and reactor vessel/internals are evaluated.

5.2.2.1 Loss-of-Coolant Accident (LOCA) Loads

LOCA loads applied to the Byron and Braidwood Units 1 and 2 RPV system consist of (a) reactor internal hydraulic loads (vertical and horizontal), (b) reactor coolant loop mechanical loads, and (c) pressure loads acting on the baffle plates. All loads are calculated individually and combined in a time-history manner.

The severity of a postulated break in a reactor vessel is related to two factors: the distance from the reactor vessel to the break location, and the break opening area. The nature of the reactor vessel decompression following a LOCA, as controlled by the internals structural configuration previously discussed, results in larger reactor internal hydraulic forces for pipe breaks in the cold leg than in the hot leg (for breaks of similar area and distance from the RPV). Pipe breaks farther away from the reactor vessel are less severe because the pressure wave attenuates as it propagates toward the reactor vessel.

Since Byron and Braidwood Units 1 and 2 take credit for Leak-Before-Break (LBB) applied to the primary loop, LOCA analyses of the reactor pressure vessel system for postulated ruptures of primary loop piping are not required. The next limiting breaks to be considered are branch-line breaks, such as in the accumulator line, pressurizer surge line, and Residual Heat Removal (RHR) line. With consideration of LBB, such auxiliary line breaks are not as severe as the main line breaks (e.g., RPV inlet nozzle or RCP outlet nozzle break).

Since LOCA forces generated for Byron and Braidwood, using the 144 sq. in. Reactor Pressure Vessel Inlet Nozzle (RPVIN) and Reactor Coolant Pump Outlet Nozzle (RCPON) breaks, remain bounded for the uprating with the application of Leak-Before-Break (LBB), the current analysis of the Byron and Braidwood Units remains bounded.

5.2.2.2 Flow Induced Vibrations

Flow-Induced Vibrations (FIV) of pressurized water reactor internals have been studied at Westinghouse for a number of years. The objective of these studies was to assure the structural integrity and reliability of reactor internal components.

Results from scale model and in-plant tests indicate that the primary cause of lower internals excitation is flow turbulence generated by expansion and turning of flow at the transition from the inlet nozzle to the barrel-vessel annulus, and wall turbulence generated in the downcomer.

The PCWG parameters, which could potentially influence FIV response of the reactor internals, include inlet nozzle flow velocities, vessel/core inlet temperatures, and vessel outlet temperatures. Generally, the inlet nozzle velocity for FIV response during hot functional testing is calculated using mechanical design flows, which are approximately 15% higher than thermal design flows.

The other parameter, which would influence the FIV response, is core inlet temperature. For the most limiting case at uprated conditions, the vessel/core inlet temperature is 542°F. The current analysis covers a temperature range of 538.2°F to 557.0°F.

Since the changes evaluated in this program did not include any changes to the mechanical characteristics of the fuel assembly design, there is no change in the fuel-related core barrel vibration response.

For the uprated conditions, it was determined that FIV loads on the guide tubes and upper support columns increase by approximately 2.15%, and in the lower internals by about 5%. Previous FIV analyses on the guide tubes and upper support columns show that there is sufficient margin to accommodate this increase in FIV loads. Consequently, the structural integrity of the Byron and Braidwood reactor internals remains acceptable with regard to flow-induced vibrations.

5.2.3 Structural Evaluation of Reactor Internal Components

In addition to supporting the core, a secondary function of the reactor vessel internals assembly is to direct coolant flows within the vessel. While directing primary flow through the core, the internals assembly also establishes secondary flow paths for cooling the upper regions of the reactor vessel and the internals structural components. Some of the parameters influencing the mechanical design of the internals lower assembly are the pressure and temperature differentials across its component parts and the flow rate required to remove heat generated within the structural components due to radiation (e.g., gamma heating). The configuration of the internals provides adequate cooling capability. Also, the thermal gradients resulting from gamma heating and core coolant temperature changes are maintained below acceptable limits within and between the various structural components.

Structural evaluations are required to demonstrate that the structural integrity of reactor internal components is not adversely affected either directly by the uprated RCS conditions and transients, or by secondary effects on reactor thermal/hydraulic or structural performance. Heat generated in reactor internal components, along with the various fluid temperature changes, result in thermal gradients within and between components. These thermal gradients result in thermal stresses and thermal growth, which must be considered in the design and analysis of the various components.

Since the Byron and Braidwood reactor internals were designed prior to introduction of Subsection NG of the ASME Boiler and Pressure Code Section III, a plant-specific stress report on the reactor internals was not required. However, the design of the Byron and Braidwood reactor internals was evaluated according to Westinghouse internal criteria, which is similar to the ASME Code. Moreover, the structural integrity of the Byron and Braidwood reactor internals design has been assured by analyses performed on both generic and plant-specific bases. These analyses were used as the basis for evaluation of critical Byron and Braidwood reactor internal components for uprating and revised design transients.

5.2.3.1 Lower Core Plate

Structural evaluations were performed to demonstrate that the structural integrity of the lower core plate is not adversely affected either by the uprated RCS conditions or by secondary effects on reactor thermal/hydraulic or structural performance. For this lower core plate evaluation, the criteria described in Section III, Subsection NG of the ASME Code were utilized.

The conclusion of these evaluations is that structural integrity of the lower core plate is maintained. The uprated RCS conditions produced acceptable margins of safety and fatigue utilization factors for all ligaments under all loading conditions. Table 5.2.3-1 lists the evaluation results.

5.2.3.2 Baffle/Barrel Region Components

The Byron and Braidwood Units 1 and 2 lower internals assembly consists of a core barrel, into which baffle plates are installed, supported by interconnecting former plates. A lower core support structure is provided at the bottom of the core barrel and a neutron panel surrounds the core barrel. The components comprising the lower internals assembly are precision machined. The baffle and former plates are bolted into the core barrel. The reactor vessel internals configuration for Byron and Braidwood Units 1 and 2 incorporates upflow in the barrel-baffle region.

5.2.3.2.1 Core Barrel Evaluation

The thermal stresses in the core active region of the core barrel shell are primarily due to temperature gradients through the thickness of the core barrel shell. Calculations were performed to determine the thermal bending and skin stresses in the core barrel for the uprated RCS conditions. Calculations were also performed for normal and upset conditions. The maximum and minimum thermal bending and skin stresses were then used to determine cyclic stresses; these in turn were used to determine the allowable number of fatigue cycles based on

ASME code allowable values. These calculations indicated that the actual number of fatigue cycles, based on all normal/upset conditions, was well below the allowable value. From these conservative results, it has been concluded that the core barrel is structurally adequate for the uprated RCS conditions at the Byron and Braidwood units.

5.2.3.2.2 Baffle Plate Evaluation

The thermal stresses in the baffle plate are caused primarily by the temperature gradient across the baffle thickness. The temperature difference between baffle and barrel produces the dominant loads on the baffle-former bolts. Calculations were performed to determine the thermal moments in the baffle plates for the uprated RCS conditions. Calculations were also performed for normal and upset conditions. The maximum and minimum thermal bending and skin stresses were then used to determine cyclic stresses; these in turn were used to determine the allowable number of fatigue cycles based on ASME code allowables. These calculations indicated that the actual number of design fatigue cycles, based on all normal/upset conditions, was well below the allowable. From these very conservative results, it can be concluded that the baffle plates are structurally adequate for the uprated RCS conditions at the Byron and Braidwood units.

5.2.3.2.3 Baffle/Barrel Bolt Evaluation

The bolts are evaluated for loads resulting from hydraulic pressure, seismic loads, preload, and thermal conditions. The temperature difference between baffle and barrel produces the dominant loads on the baffle-former bolts. Hydraulic pressure and seismic loads produce the primary stresses, whereas bolt preloading and thermal conditions produce the secondary stresses. The uprated RCS conditions do not affect deadweight or preload forces.

Since these bolts are qualified by test, the evaluation of the revised loads consisted of comparing the existing operating loads to those developed with the uprated RCS conditions. The results indicate that the thermal baffle/former and barrel/former bolt loads, with the currently analyzed condition, bound those developed with the uprated RCS conditions. Therefore, it is concluded that the baffle/former and barrel/former bolts are structurally adequate for the uprated RCS conditions at the Byron and Braidwood units.

5.2.3.2.4 Upper Core Plate Evaluations

The upper core plate positions the upper ends of the fuel assemblies and the lower ends of the control rod guide tubes, thus serving as the transitioning member for the control rods in entry and retraction from the fuel assemblies. It also controls coolant flow exiting the fuel assemblies and serves as a boundary between the core and exit plenum. The upper core plate is restrained from vertical movement by the upper support columns, which are attached to the upper support plate assembly. Four equally-spaced core plate alignment pins restrain lateral movement.

The stresses in the upper core plate are mainly due to hydraulic, seismic, and thermal loads. The total thermal stresses are due to thermal bending moments through the thickness and surface peak stresses. Evaluations were performed to determine the impact of uprating on the structural integrity of the upper core plate. As a result of this evaluation, it is concluded that the upper core plate is structurally adequate for the uprated RCS conditions at the Byron and Braidwood units.

5.2.3.4 Additional Components

A series of assessments were performed on reactor internal components, which were not significantly impacted by the power uprating (and the resulting internal heat generation rates) but are affected by the uprated RCS conditions due to primary loop design transients. These components are:

a. Core barrel plug
b. Lower core support plate
c. Lower support columns
d. Core barrel outlet nozzle
e. Core barrel flange
f. Lower radial restraints (clevis inserts)
g. Upper core plate alignment pin
h. Upper support columns
i. Upper support plate
j. Guide tubes and support pins
k. Neutron pads

The results of these assessments demonstrate that the above listed components are structurally adequate for the uprated RCS conditions at the Byron and Braidwood units.

5.2.4 Results/Conclusions

Analyses have been performed to assess the effect of changes due to power uprate. The results of these analyses are as follows:

1. The total core bypass flow values (with uncertainties) were determined to be 6.83% and 6.6% for Byron Units 1 and 2, and 7.39% and 6.84% for Braidwood Units 1 and 2, respectively. Therefore, the design core bypass flow value of 8.3% of the total vessel flow is maintained for the uprating.

2. Hydraulic forces were calculated to assess the structural integrity of the reactor internals. It was determined that the Byron and Braidwood reactor internals assemblies will remain seated and stable at the uprated conditions.

3. An RCCA performance evaluation was completed and indicated that the current 2.7 second RCCA drop time to dashpot entry limit (from gripper release of the drive rod) is satisfied at power uprate conditions.

4. Baffle plate momentum flux margins of safety due to power uprate conditions are relatively unchanged from present conditions for mechanical design flow, and remain acceptable.

5. Evaluations were completed and indicated that the uprated RCS conditions will not adversely impact the response of reactor internals systems and components due to seismic/LOCA excitations and flow induced vibrations.

6. Evaluations of the critical reactor internal components were performed, which indicated that the structural integrity of the reactor internals is maintained at the uprated RCS conditions.

5.2.5 References

1. Safety evaluation by the office of Nuclear Reactor Regulation related to Amendment No. 137 to Facility Operating License No. NPF-2 and Amendment No. 129 to Facility Operating License No. NPF-8 Southern Nuclear Operating company Inc., et al. Joseph M. Farley Nuclear Plant, Units 1 and 2 Docket Nos. 50-348 and 50-364.

Table 5.2.3-1 Maximum Calculated Stress, Allowable, and CUF at the Most Critical Reactor Internal Component							
Reactor Internal Component	**Max Stress Pm**	**Code Limit Sm**	**Max Stress Pm+Pb**	**Code Limit 1.5Sm**	**Max Stress Pm+Pb+Q**	**Code Limit 3Sm**	**CUF**
Lower Support Column (5.2.3.4)	3415	16,100	22,502	24,150	35,140	48,300	0.267

5.3 Fuel Assemblies

The combined effects of the design basis loads are considered in evaluating the capability of fuel assemblies and their components to maintain structural integrity. This is necessary so that fuel assembly functional requirements are met while maintaining the core coolable geometry and the ability of reactor core safe shutdown.

Increasing core power will not change the core plate motions. The core power uprating does not increase operating or transient loads such that they will adversely affect fuel assembly functional requirements. Fuel assembly structural integrity is not affected and the core coolable geometry is maintained for the 17x17 VANTAGE+ (Zirlo™ with 0.360 rod and debris mitigating features) Fuel Assembly Design for Byron and Braidwood Units 1 and 2.

Following the core power uprate, the flow per assembly in the Byron and Braidwood units will be slightly higher than the flow per assembly in previous Byron and Braidwood analyses. The lift forces derived for the core power uprating are slightly higher than the lift forces from previous Byron and Braidwood analyses. The fuel assembly holddown spring capability was verified to still be acceptable. Thus, fuel assembly structural integrity is not affected by the core power uprating.

Other areas, which were considered for potential impact from the core, power uprate, such as fuel rod fretting, oxidation and hydriding of thimbles and grids, fuel rod growth gap, and guide thimble wear, were determined to have a negligible effect. It is concluded that the fuel assemblies are not adversely impacted by the core power uprate.

5.4 Control Rod Drive Mechanisms

5.4.1 Introduction

This section addresses the ASME Code of record structural considerations for the pressure boundary components of the Westinghouse full length L-106A Control Rod Drive Mechanisms (CRDMs) and seismic sleeves. The CRDMs are evaluated for the Byron and Braidwood Uprating Program PCWG parameters (Chapter 2.0) and the NSSS design transients (Chapter 3.0).

5.4.2 Input Parameters and Assumptions

The Model L-106A CRDMs were originally designed and analyzed to meet the Byron and Braidwood equipment specification (References 1 and 2) and the ASME Code (Reference 3). Sub-component evaluations have been performed for these Byron and Braidwood CRDMs.

The Byron and Braidwood Uprating Program modifies the original basis PCWG parameters and the NSSS design transients. None of the other input parameters are changed or they remain bounded by the current evaluations.

The Byron and Braidwood CRDMs are of the cold head type, defined by the vessel/core inlet temperature on the PCWG Parameters, and must satisfy the NSSS design transients defined by the cold leg per the site-specific equipment specification (References 1 and 2).

The current Byron and Braidwood LOCA vessel and loop piping hydraulic forcing functions remain bounding for the Uprating conditions per Section 6.6. The Byron and Braidwood Uprating seismic loads are unchanged or remain bounded by the current operating conditions.

All maintenance and repair activities on the CRDMs have been done in accordance with the original design requirements. Therefore, the CRDMs continue to meet their original design requirements.

The impacts of the uprate on the CRDMs are discussed in Section 5.4.3. The largest temperature and pressure changes from either the high temperature case or the low temperature case are evaluated in Section 5.4.3.

5.4.3 Description of Analysis and Evaluations

5.4.3.1 Transient Discussion

From the PCWG parameters shown in Section 2.0, there are no changes from the current reactor coolant pressure of 2250 psia for any of the Uprating cases. The cold leg temperature defined by the vessel/core inlet temperature on the PCWG Parameters for the Byron and Braidwood Uprating is a maximum of 555.7°F, which is less than the original basis temperature of 558.4°F. Since none of the temperatures exceeds the original basis temperature and the pressure does not change, the Uprating PCWG Parameters are bounded by the original site-specific analyses.

From Table 5.4-1, the present conditions provide a cold leg temperature range of 538.2°F to 558.4°F compared to 538.2°F to 555.7°F for the Uprating. The Uprating T_{COLD}, Low T_{avg} value is based on the footnotes to the tables of Section 2.0, limiting the minimum T_{COLD} value to 538.2°F. Therefore, the current operation range bounds the range of cold leg temperatures for the Uprating.

The only transient modified in Section 3.0 is the Loss of Load transient. The temperature component for the Low T_{avg} parameters is the only relevant change to the transient. The Uprate program Low T_{avg} peak value provides for a slightly greater ΔT (47.6°F) than the ΔT (43.5°F) previously evaluated.

Although the RCS Cold Overpressurization transient was not added due to the uprating, it must be addressed for completeness. The transient is defined to occur 10 times. Each occurrence entails a thermal cycle and an associated series of 600 pressure cycles. The transient is thus represented by two distinct transient components. The first component is for the ten combined thermal and pressure cycles. The second component is for the additional 5990 pressure-only cycles.

The ten combined thermal and pressure cycles are composed of a temperature change of +38°F and a pressure range of 195 to 800 psi. The 5990 pressure-only cycles are for a pressure change of 605 psi. The RCS Cold Overpressurization transient will be addressed for the Uprating.

5.4.3.2 Component Effects

The only difference from previous evaluations and the current Uprating Program, as discussed in the Section 5.4.2, is the modification of the Loss of Load transient and the addition of the RCS Cold Overpressurization transient.

The site-specific reports evaluate the imposed transients in groups. The method groups similar transients, then imposes a worst-case (bounding) transient for the number of cycles of the combined group of transients on the CRDM components. This reduces the number of different transients requiring analysis.

Both the Uprating Loss of Load transient and the RCS Cold Overpresurization transient are less severe than the upset grouping transient previously evaluated (ΔT = 48°F at a pressure of P = 2643 psig). Therefore, the maximum stress intensities for all of the analyses, with the possible exception of the fatigue analyses, remain bounding for the Uprating. Additional cycle implications were evaluated for fatigue.

The upper joint canopy of the CRDMs has been previously shown to be the area of highest fatigue usage. Since none of the conditions associated with the Uprating would cause a change in the location of the area of highest fatigue usage, the upper joint canopy is again investigated for fatigue.

The Uprating-modified Loss of Load transient (ΔT = 47.6°F at a pressure of P = 2643 psig) is slightly more severe than the previous Loss of Load transient (ΔT = 43.5°F at a pressure of P = 2643 psig), but is still less severe than the transient (ΔT = 48°F at a pressure of P = 2643 psig) used to evaluate the transient group containing the Loss of Load transient. There were no changes in number of Loss of Load transient cycles. Therefore, the modification to the Loss of Load transient has no effect on the CRDM fatigue analysis.

The RCS Cold Overpressurization transient was also evaluated for the CRDMs. As discussed in Section 5.4.3.1, the transient is composed of two separate components. The first component is the more severe in that it contains both a thermal and a pressure transient (ΔT = 38°F with a pressure range between 195 and 800 psi for ten cycles). The second transient component is merely a relatively low pressure, pressure transient (cycles from 195 psi to 800 psi and back to 195 psi 5990 times) at a low temperature.

The pressure and temperature requirements of the first component of the RCS Cold Overpressurization transient are bounded by the same group of transients bounding the Loss of Load transient. Therefore, the only change in the current analysis for the first component of the RCS Cold Overpressurization transient is the increase in fatigue usage by the 10 added cycles of the transient. This results in an increase of fatigue usage of 0.007.

The second component of the RCS Cold Overpressurization transient is composed of a pressure change of 605 psi for a total of 5990 cycles. This component has been shown to not increase the fatigue usage.

5.4.4 Results and Acceptance Criteria

The total change in fatigue usage for the Uprating Program is +0.007. The previous analysis worst-case fatigue usage for the upper joint canopy was 0.934. The worst-case fatigue usage for the Uprating Program is 0.941 for this same upper joint canopy. This is still less than the Code allowable of 1.0. Therefore, the CRDMs continue to meet the E-specifications and the ASME Code of record for the Uprating Program.

Review of the effects of the Uprating PCWG parameters and NSSS design transients on the CRDM site-specific reports, as described in the previous section, shows that the ASME Code of record is still met. The plant-specific E-specification criteria for the CRDM remains satisfied.

5.4.5 Conclusions

The Uprating Program PCWG parameters and NSSS design transients are acceptable to the CRDMs from a structural standpoint. The CRDM pressure boundary parts are considered to still satisfy the CRDM E-specifications and the ASME Code of record. Therefore, the results for the Uprating are consistent with and continue to comply with the current licensing basis/acceptance requirements for Byron Units 1 and 2, and Braidwood Units 1 and 2.

5.4.6 References

1. Plant Design Specification 953516, Revision 2, Project: Byron Project Units 1 and 2, Braidwood Project Units 1 and 2, Equipment: Control Rod Drive Mechanism Model L-106A, Westinghouse Nuclear Energy System, Pittsburgh, PA, April 17, 1984.

2. Plant Design Specification 953516, Revision 1 (for EM 5371 and EM 5961 only), Project: Byron Project Units 1 and 2, Braidwood Project Units 1 and 2, Equipment: Control Rod Drive Mechanism Model L-106A, Westinghouse Nuclear Energy System, Pittsburgh, PA, May 31, 1983.

3. "ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components," American Society of Mechanical Engineers, NY, 1974 Edition with Addenda through Summer 1974.

Table 5.4-1 PCWG Conditions Used to Bracket All Operating Conditions for Byron and Braidwood Uprating				
	Present		Uprating	
	High T_{avg}	Low T_{avg}	High T_{avg}	Low T_{avg}
T_{cold}	558.4°F	538.2°F	555.7°F	538.2°F

Per Section 2.0:

P = 2250 psia for all cases

T_{cold} = 558.4°F Original Basis

T_{cold} = 542.0°F(1) Cases 1 and 2 for Byron and Braidwood Units 1 & 2[(2)]

T_{cold} = 555.7°F Cases 3 and 4 for Byron and Braidwood Units 1 & 2[(2)]

Normal Operating Conditions:

P = 2250 psia

Tcold = 550°F

[(1)] Plant operation limited to a minimum T_{cold} of 538.2°F.

[(2)] Tables 2.1-1 and 2.1-2

Table 5.4-2 Applicable Design Transients and Cycle Count Comparison		
Plant Condition	**Current Evaluation Cold Leg**	**Uprating Cold Leg**
Normal Conditions		
1. Plant Heatup/Cooldown	200	200
2. Unit Loading at 15%/min	13,200	13,200
3. Unit Unloading at 15%/min	13,200	13,200
4. 10% Step Load Increase	2000	2000
5. 10% Step Load Decrease	2000	2000
6. Large Step Load Decrease	200	200
7. Steady State Fluctuations	1.5×10^5 3.0×10^6	1.5×10^5 3.0×10^6
8. Feedwater Cycling	2000	2000
9. 0-15% Unit Loading	500	500
10. 15-0% Unit Unloading	500	500
11. Boron Concentration Equal.	26,400	26,400
12. Refueling	80	80
Upset Conditions		
1. Loss of Load	80	80
2. Loss of Power	40	40
3. Partial Loss of Flow	80	80
4. Reactor Trip		
Case A	230	230
Case B	160	160
Case C	10	10
5. Inadvertent RCS Depressurization	20	20
6. Control Rod Drop	80	80
7. Inadvertent Safety Injection	60	60
8. RCS Cold Overpressurization	N/A	10

Table 5.4-2 (Continued)
Applicable Design Transients and Cycle Count Comparison

Plant Condition	Current Evaluation Cold Leg	Uprating Cold Leg
Emergency Conditions		
1. Small LOCA	5	5
2. Small Steam Line Break	5	5
3. Complete Loss of Flow	5	5
Faulted Conditions		
1. Large LOCA	1	1
2. Large Steam Line Break	1	1
3. Feedwater Line Break	1	1
4. RCP Locked Rotor	1	1
5. Control Rod Ejection	1	1
Test Conditions		
1. Turbine Roll Test	20	20
2. Primary Side Hydrostatic Test	10	10
3. Primary Side Leak Test	200	200
4. Secondary Side Leak Test	80	80

5.5 RCL Piping and Supports

The power uprate program for Byron and Braidwood Units 1 and 2 was reviewed for impact on the existing design basis.

5.5.1 Introduction

The power uprate program and its associated parameters were reviewed for impact on the existing basis analysis for the following components: the reactor coolant primary loop piping, primary equipment supports, primary equipment nozzles, reactor coolant loop branch nozzles, and the pressurizer surge piping. The temperature changes associated with the power uprate cause potential load changes in the components to be reconciled. The changes in the thermal design are also factored into the fatigue analysis of the piping components above. There are no pressure changes since the above piping components are evaluated for the design pressure which bounds the power uprate conditions.

The Byron and Braidwood Units 1 and 2 reactor coolant primary loop piping analyses were performed in 1996 for the application of Leak-Before-Break (LBB) to demonstrate the acceptability of LBB. The evaluation was documented in Westinghouse's WCAP-14559 Revision 1, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Byron and Braidwood Units 1 and 2 Nuclear Power Plants." This analysis was approved by the NRC in a letter dated October 25, 1996. Currently, an LBB evaluation is performed for the Byron and Braidwood Units 1 and 2 reactor coolant primary loop piping due to the power uprate program to show that the conclusion derived in WCAP-14559 Revision 1 is still valid.

5.5.2 Input Parameters and Assumptions

5.5.2.1 Input Parameters

The following items were considered in the evaluation of the above piping and primary equipment supports:

- Performance Capability Working Group (PCWG) Parameters (provided in Section 2.0 of this report)

- Analysis for NSSS design transients (provided in Section 3.1 of this report)
- Loss of Coolant Accident (LOCA) forces analysis (provided in Section 6.6 of this report)
- Results/data of deadweight, thermal, seismic and LOCA from Framatome Technologies, Incorporated (FTI) for Byron Unit 1 and Braidwood Unit 1

The PCWG parameters define the various temperature conditions associated with the potential full power operating conditions of the plant and are used to address the impact of power uprate conditions on the stated piping and primary equipment supports.

The system thermal design transients are used in the evaluation of the piping fatigue. The impact of changes in the system thermal transients were factored into the ASME code stress and fatigue usage factor determination.

The LOCA analysis required for the piping defined in this report has the following inputs to consider: the loop LOCA forces associated with the postulated breaks (as defined in the plant licensing report except those eliminated because of LBB), and reactor vessel dynamic LOCA displacements associated with the postulated break cases. It was determined as part of the power uprate analysis effort that the loop LOCA forces and the reactor vessel dynamic LOCA displacements associated with the postulated break cases do not change as a result of the power uprate conditions and therefore, are not needed for input.

The parameters that are important in the LBB evaluation are the piping forces, moments, normal operating temperature, normal operating pressure, and the material properties.

The current analysis for reactor coolant primary loop piping performed by FTI, due to steam generator replacement at Units 1, is also used to show primary equipment supports in Units 1 are not significantly affected at the power uprate conditions and therefore, remain adequate.

5.5.2.2 Assumptions

The premise for the evaluation for this uprating program is that all analyses, methods and criteria used in the existing design basis for Byron and Braidwood Units 1 and 2 will continue to be used for this evaluation.

5.5.3 Description of Analysis/Evaluation

The results of the existing thermal analysis were reviewed in the determination of the impact of power uprate on current loads and stresses on the reactor coolant primary loop piping, primary equipment supports, primary equipment nozzles, reactor coolant loop branch nozzles, and the pressurizer surge piping. The OBE and SSE results are not affected by the power uprate. Per Section 6.6 of this report, the existing design basis LOCA forces continue to envelop the power uprate condition LOCA loadings. The impact of the power uprate transients on the existing fatigue analysis was evaluated for the critical components of the piping defined in this report. The thermal loading is the only loading that is affected by the power uprate program.

The evaluation performed to address the effects of the power uprate on the pressurizer surge line stratification analysis included a review of the fatigue analysis. The temperature differences between the hot leg and pressurizer were used as a basis for the assessment. The increases in the differential temperature for the thermal design transients affecting stratification were determined and the impacts on the fatigue usage factor and ASME code stress requirements were assessed.

An LBB evaluation was performed for the Byron and Braidwood Units 1 and 2 reactor coolant primary loop piping using the operating parameters and loads applicable to the power uprate program. Information contained in WCAP-14559 Revision 1 was utilized in this evaluation.

5.5.4 Acceptance Criteria

The base acceptance criteria are the requirements of:

The ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Power Plant Components, 1977 Edition and addenda up to and including Summer 1979. This code is used for stress analysis purposes for reactor coolant primary loop piping and branch nozzles.

The ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Power Plant Components, 1983 Edition. This code is applicable for stress analysis purposes for the pressurizer surge line, hot and cold leg fast response RTD thermowells, and crossover leg nozzle caps.

The primary equipment supports are considered insignificantly affected and therefore, adequate when the thermal movements and piping leads at power uprate conditions remain unchanged or insignificantly changed.

5.5.5 Results

The acceptance criteria were met. The results of detailed evaluation for the reactor coolant loop piping show that the ASME code stress requirements of References 1 and 2 are met.

The LBB evaluation demonstrated the acceptability of LBB for the reactor coolant primary loop piping using the loads and operating parameters from the power uprate program.

The branch piping movements and primary equipment support loads at the power uprate were shown to be either unchanged or varied by very small amounts. Therefore, the primary equipment supports are considered to be insignificantly affected by these increases.

5.5.6 Conclusions

The parameters associated with the power uprate program for Byron and Braidwood Units 1 and 2 have been evaluated for the following components:

- reactor coolant primary loop piping,
- primary equipment supports,
- reactor coolant loop branch nozzles,
- primary equipment nozzles,
- pressurizer surge piping.

The evaluation indicates that all components meet the required design basis criteria. The evaluation for the stated components concluded that the power uprate had no adverse effect on the ability of these components to operate until the scheduled end of plant operation.

The effects of the power uprate program on the continued applicability of LBB for the reactor coolant primary loop piping at Byron and Braidwood Units 1 and 2 have been evaluated. It is determined that the previous LBB analysis conclusion remains valid, and the dynamic effects of the pipe rupture resulting from postulated breaks in the reactor coolant primary loop piping need not be considered in the structural design basis of the Byron and Braidwood Units 1 and 2 for the power uprate program.

5.5.7 References

1. ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Power Plant Components, 1983 Edition.

2. ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Power Plant Components, 1977 Edition and addenda up to and including Summer 1979.

3. WCAP-14559 Revision 1, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Byron and Braidwood Units 1 and 2 Nuclear Power Plants," April 1996.

5.6 Reactor Coolant Pumps

The Reactor Coolant Pumps (RCPs) at Byron and Braidwood were evaluated for the power uprating in two separate areas: the structural adequacy of the pumps (Section 5.6.1), and the acceptability of the RCP motors (Section 5.6.2).

5.6.1 Reactor Coolant Pumps (Structural)

5.6.1.1 Introduction

This section addresses the ASME Code of record structural considerations for the pressure boundary components of the Westinghouse Model 93A RCPs. The RCPs were evaluated for the Byron and Braidwood Power Uprating Program PCWG parameters (Chapter 2.0) and the Nuclear Steam Supply System (NSSS) design transients (Chapter 3.0).

5.6.1.2 Input Parameters and Assumptions

The Model 93A RCPs were originally designed and analyzed to meet the Byron and Braidwood equipment specification (Reference 1), the RCP generic specification (Reference 2), and the ASME Code (Reference 3). In 1986, a T_{HOT}-Reduction Program evaluation was performed for these Byron and Braidwood RCPs.

The current Byron and Braidwood Power Uprating Program modifies the original basis PCWG parameters and the NSSS design transients based on the T_{HOT}-Reduction Program. The other input parameters are either unchanged or remain bounded by the original or the T_{HOT}-Reduction Program basis.

The Byron and Braidwood RCPs are installed in the Reactor Coolant System (RCS) cold leg, as defined by the steam generator outlet temperature on the PCWG Parameters. They must satisfy the NSSS design transients, as defined by the RCS cold leg per the site-specific and general equipment specifications (References 1 and 2) and updated by the T_{HOT}-Reduction Program and this power uprating.

The current Byron and Braidwood LOCA vessel and loop piping hydraulic forcing functions remain bounding for the power uprating conditions per Section 6.6 of this Report. The Byron and Braidwood uprating power uprating seismic loads are either unchanged or remain bounded by the current operating conditions.

All maintenance and repair activities on the RCPs have been performed in accordance with the original design requirements. Therefore, the RCPs continue to meet their original design requirements.

The original site-specific Pressure Boundary Summary Report (PBSR) is Reference 4. The PBSR reconciles any differences between the site-specific requirements and the generic report requirements, including equipment specification changes, drawing changes, material changes, ASME Code edition changes, and fabrication deviations.

The PBSR was reviewed for compliance with the Byron and Braidwood T_{HOT}-Reduction Program and steam generator tube plugging for both 15% and 24%. The reviews showed that, for the Byron and Braidwood T_{HOT}-Reduction Program, all of the equipment specifications and code requirements were met and the 15% and 24% SG tube plugging requirements were bounded by the T_{HOT}-Reduction Program.

The differences associated with the uprating requirements are discussed in Section 5.6.1.3. The largest temperature and pressure changes from either the high or the low RCS temperature cases are evaluated in Section 5.6.1.3.

A generic evaluation of the Westinghouse Model 93A RCP, including the RCS Cold Overpressurization transient, was performed. This evaluation was also used as the basis for the Power Uprating Program review. This evaluation shows that the RCS Cold Overpressurization local temperature transient associated with the RCP is the more severe temperature transient for most components. The RCS Cold Overpressurization transient impact is more severe since the RCP sees the additional effect of the accumulation, then rapid flushing, of cold injection water in the pump hydraulic cavity.

A thermal evaluation of the RCS Cold Overpressurization transient effects on the Model 93A RCP indicate that the discharge nozzle, casing at the discharge nozzle and weir plate are not affected by the local temperature transient, but should be evaluated for the bulk temperature transient effects. Both the local and bulk temperature transients are evaluated in Section 5.6.1.3.

5.6.1.3 Description of Evaluations and Acceptance Criteria

5.6.1.3.1 Transient Discussion

From the Power Uprating program PCWG parameters for Byron and Braidwood Units 1 and 2, there are no changes from the current reactor coolant pressure of 2250 psia for any of the uprating cases. The RCS cold leg temperature, defined by the SG outlet (RCP inlet) temperature on the PCWG Parameters in Section 2.0 for the Byron and Braidwood uprating is a maximum of 555.4°F for Cases 3 and 4. The maximum uprating RCS temperature is less than the original 4-Loop temperature of 558.1°F. Since none of the temperatures exceed the original basis temperature and the pressure does not change, the uprating PCWG Parameters are bounded by the original site-specific analyses.

Table 5.6-1 summarizes the cold leg PCWG parameters and the bracketing conditions, as well as input data for the referenced reports. From Table 5.6-1, the present conditions provide an RCS cold leg T_{avg} range of 538.2°F to 558.4°F compared to 538.2°F to 555.7°F for the uprating. The uprating T_{COLD}, Low T_{avg} value is based on the footnotes to the tables of Section 2.0, which limit the minimum T_{COLD} value to 538.2°F. Therefore, the current operation range bounds the RCS cold leg temperature ranges.

Section 3.0 provides a list of the required transient changes for the uprating. The transient revisions are based on the T_{HOT}-Reduction program transients. None of the primary side transient descriptions or number of occurrences change. However, the RCS Cold Overpressurization transient cycles has been added to current requirements. Table 5.6-2 compares the original equipment specification required (References 1 and 2) transient cycle count to the T_{HOT}-Reduction program transient cycle count and the Power Uprating Program transient cycle count.

The only transient modified by the Chapter 3.0 requirements from the T_{HOT}-Reduction program is the Loss of Load transient. The temperature component for the Low T_{avg} parameters is the only relevant change to the transient. The Power Uprate program Low T_{avg} peak value requires a slightly greater ΔT (47.6°F) than the ΔT (43.5°F) evaluated in the T_{HOT}-Reduction program. The change is addressed in Section 5.6.1.3.2.

The additional transient is the RCS Cold Overpressurization (also referred to as the Cold Overpressure Mitigation System (COMS) transient). The transient is defined to occur 10 times. Each occurrence entails a thermal cycle and an associated series of 600 pressure cycles. The thermal portion of the transient, as applied to the RCP, is composed of two possible independent events. The first event is the bulk temperature change represented by a change in temperature of +38°F applicable (in general) to all RCS components. The second event applies specifically to the RCP. It assumes that the pump is out of operation for one hour and then restarted. During the hour out of operation, 300 gallons of RCS cold leg seal water flows through the pump and cools the internal components and casing at the suction nozzle to a minimum of 60°F. The discharge nozzle and weir plate remain outside of the cool water flow path. Once the pump is started, the cold water is almost instantaneously replaced with the hot RCS leg water at a maximum temperature of 350°F. This produces a maximum change in temperature of +290°F. This is the more severe condition applicable to the pump (except the discharge nozzle, casing at the discharge nozzle and weir plate) and is investigated in this evaluation. The discharge nozzle, casing at the discharge nozzle and weir plate are investigated for the bulk temperature change of +38°F, since they are removed from the most severe effects of the localized temperature transient.

The COMS transient is thus represented by two distinct transient components. The first component is for ten combined thermal and pressure cycles. The ten combined thermal and pressure cycles are composed of a maximum temperature change of +290°F (+38°F for the discharge nozzle, casing at the discharge nozzle and weir plate) and a pressure range of 195 to 800 psi. The second component is for the additional 5990 pressure-only cycles. The 5990 pressure-only cycles are for a pressure change of 605 psi. The RCS Cold Overpressurization transient is addressed for the uprating in Section 5.6.1.3.2.

5.6.1.3.2 Effect on Components

Reference 4 is the site-specific PBSR for the Byron and Braidwood RCPs. The PBSR was developed to the requirements of the site-specific plant design specifications (Reference 1) and was based on the analyses from the generic reports. The site-specific plant design equipment specifications required the RCPs to be designed for the transient criteria of References 1 and 2.

The PBSR (Reference 4) updated the results of the generic report analyses to comply with the Byron and Braidwood site-specific plant design equipment specifications and the ASME Code

(Reference 3). Therefore, the PBSR includes evaluations for fabrication variations of the RCPs, component drawing changes, component material changes, ASME Code Addenda variation, and design specification changes. The PBSR shows that the RCPs comply with all of the plant design equipment specification and ASME Code requirements.

In 1986, the Byron and Braidwood RCP evaluations were updated to comply with the requirements of the T_{HOT}-Reduction Program. The evaluations show that since none of the changes were severe enough to cause the original worst-case transient to be exceeded, the stress evaluation of the PBSR remained valid. The evaluations also show that the modified ΔPs and ΔTs remain less than the fatigue waiver limits, indicating that the PBSR fatigue waiver remains valid. Therefore, the original RCP PBSR bounds T_{-Hot}-Reduction Program transients.

The only differences between the T_{HOT}-Reduction Program and the current Power Uprating Program, as discussed in Section 5.6.1.3.1, are the modification of the T_{HOT}-Reduction Program Loss of Load transient and the addition of the RCS Cold Overpressurization transient.

The uprating Loss of Load transient (ΔT = 47.6°F at a pressure of P = 2725 psia) is less severe than the transient evaluated in the generic reports and updated in the PBSR (ΔT = 52.5°F at a pressure of P = 2725 psia). There were no changes in the number of cycles for the Loss of Load transient. Therefore, the maximum stress intensities and fatigue evaluation for all the Reference 4 analyses remain bounding for the uprating Loss of Load transient.

The requirements of the RCS Cold Overpressurization transient were added to the RCPs. As discussed in Section 5.6.1.3.1, the transient is composed of two separate components. The effect of the RCS Cold Overpressurization transient was evaluated for the individual RCP components.

Casing, Main Flange, Main Flange Bolts, and Thermal Barrier

Table 5.6-3 lists the conservative results that conclude that these components comply with the ASME Code of Record requirements for the power uprating specifications.

Casing Foot

The Reference 4 casing peak stress intensity range magnitude (based on an elastic-plastic analysis) bounds the power uprating value of 16,530 psig.

The COMS transient increased the fatigue usage from 0.38 to 0.424, which is less than the ASME Code allowable of 1.0.

Casing Suction and Discharge Nozzle

Since the power uprating does not change the peak pressure loads applied to the RCP, the primary stress intensity evaluation of Reference 4 is unchanged for both the suction and discharge nozzles.

The effects of the localized RCP COMS transient conditions on the suction nozzle required that a simplified elastic-plastic analysis, per ASME Code section NB-3228.3, be performed.

The maximum primary plus secondary stress intensity range magnitude (excluding thermal bending stress) for the suction nozzle is 39,490 psi. This is less than the ASME Code allowable of $3S_m$ = 47,850 psi. Therefore, the ASME Code Section NB-3228.3(a) requirement is met.

The fatigue analysis per ASME Code Section NB-3222.4 resulted in a fatigue usage factor of 0.581, which is less than 1.0, the ASME Code allowable. Therefore, ASME Code Sections NB 3228.3(b) and (c) are met.

The ASME Code Section NB-3228.3(d) allowable thermal stress for the suction nozzle is 76,860 psi. The maximum thermal stress range magnitude for the suction nozzle is 52,760 psi. Therefore, ASME Code Section NB-3228.3(d) is met.

ASME Code Section NB-3228.3(e) is met since no temperature exceeds 800°F.

ASME Code Section NB-3228.3(f) is met since the ratio of material minimum specified yield strength (30 ksi) to specified ultimate strength (70 ksi) = 0.43, which is less than 0.80.

Since the six elastic-plastic analysis requirements for the power uprating are met, the suction nozzle remains in compliance with the ASME Code of record.

The discharge nozzle and casing at the discharge nozzle stress analysis is unaffected by the COMS bulk temperature and pressure transients.

The fatigue usage increased due to the 10 additional COMS transient cycles. The usage increased from 0.209 to 0.300, which is still less than 1.0, the ASME Code allowable.

Therefore, the discharge nozzle and casing at the discharge nozzle remain in compliance with the ASME Code of record requirements for the power uprating.

Casing Weir Plate

Since the uprating does not change peak pressure loads, the primary stress intensity evaluation of Reference 4 is unchanged.

The COMS (bulk temperature) transient is not as severe as those previously evaluated for the weir plate. Thus, the ASME Code of record stress requirements for the power uprating COMS transient are met. The weir plate fatigue waiver is not affected by the COMS transient. The ASME Code of record fatigue requirements remain satisfied for the power uprating.

Thermal Barrier Heat Exchanger

The thermal components of the COMS transients and the loss of load transients do not affect the thermal barrier heat exchanger components. Therefore, no changes to the PBSR are required.

Seal Housings and Seal Housing Bolts

There is no change in stress intensity for the seal housings and bolts due to the power uprating parameters.

The fatigue waiver for the seal housing has been evaluated for the power uprating condition, and continues to be met.

The fatigue waiver for the main flange has also been evaluated for the power uprating and continues to be met. Therefore, the seal housing bolts continue to meet the ASME Code Section NB-3232.3 fatigue waiver.

Auxiliary Nozzles

The ASME Code calculations for the auxiliary nozzles were reviewed for the effects of the power uprating conditions. The review indicated that none of the calculations were effected by the power uprating. The ASME Code of record requirements remain satisfied for the power uprating.

Additional Components

Several non-pressure boundary components have been evaluated to the more stringent requirements of ASME Section III, Class 1 although they are Class 2 components.

The stresses remain bounded by the current analyses.

The three components evaluated for fatigue usage are the heat exchanger end plate, thermal sleeve, and shaft at the end plate.

The following are the results of those evaluations:

End Plate	Usage increased from 0.335 to 0.352 which is less than ASME Code allowable of 1.0
Thermal Sleeve	Usage increased from 0.640 to 0.667 which is less than ASME Code allowable of 1.0
Shaft at the End Plate	Usage increased from 0.160 to 0.167 which is less than ASME Code allowable of 1.0

5.6.1.4 Results

The review of the effects of the uprating PCWG parameters and NSSS design transients on the RCP site-specific and generic reports, as described in the previous section, shows that the ASME Code of record (Reference 3) is still met. The general and plant-specific equipment specification criteria for the RCPs (References 1 and 2) remain satisfied.

5.6.1.5 Conclusions

The Power Uprating Program PCWG parameters and NSSS design transients are acceptable to the RCPs from a structural standpoint. The RCP pressure boundary parts will continue to satisfy the RCP equipment specifications and the ASME Code of record. Therefore, the results for the uprating are consistent with, and will continue to comply with, the current licensing basis/acceptance requirements for Byron and Braidwood Units 1 and 2.

5.6.1.6 References

1. Reactor Coolant Pump Equipment Specification 952139, Revision 11, Westinghouse Pressurized Water Reactor Systems Division, Monroeville, PA.

2. General Equipment Specification G-677188, Revision 4, "Reactor Coolant Pump," Westinghouse Nuclear Energy System, Pittsburgh, PA, By J. Green, Revised by A. A. Anderson, January 15, 1976.

3. "ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components," American Society of Mechanical Engineers, NY, 1971 Edition with Addenda through Winter 1972.

4. Cuerden, B., "Pressure Boundary Component Summary Report for the Model 93A Reactor Coolant Pumps of the Commonwealth Edison Byron and Braidwood Nuclear Plants," Engineering Memorandum 5111, Revision 3, Westinghouse Electro-Mechanical Division, Cheswick, PA, July 15, 1983.

Table 5.6-1
PCWG Conditions Used to Bracket All Operating Conditions for Byron and Braidwood Power Uprating

	Present		Uprating	
	High T_{avg}	Low T_{avg}	High T_{avg}	Low T_{avg}
T-cold	558.4°F	538.2°F	555.7°F	538.2°F

OTHER REFERENCE NORMAL OPERATING CONDITIONS

P = 2250 psia for all cases

T-cold = 558.1°F Original Basis

T-cold = 541.7°F Cases 1 and 2 for Byron and Braidwood Units 1 & 2 Section 2

T-cold = 555.4°F Cases 3 and 4 for Byron and Braidwood Units 1 & 2 Section 2

Per Reference 2:

P = 2332 psia at RCP discharge nozzle for a pressurizer pressure of 2250 psia.

Per Reference 4 (Site-specific PBSR):

P = 2250 psia (2332 psi value is combined with positive pressure transients to determine maximum pressures)

T-cold = 556.7°F (Normal Operating Temperature)

Table 5.6-2 Applicable Design Transients and Cycle Count Comparison			
Plant Condition	Equipment Specs. (References 1 & 2), Cold Leg	T_{HOT} Reduction Cold Leg	Uprating Cold Leg
Normal Conditions			
1. Plant Heatup/Cooldown	200	200	200
2. Unit Loading at 15%/min	13,200	13,200	13,200
3. Unit Unloading at 15%/min	13,200	13,200	13,200
4. 10% Step Load Increase	2000	2000	2000
5. 10% Step Load Decrease	2000	2000	2000
6. Large Step Load Decrease	200	200	200
7. Steady State Fluctuations	1.5×10^5 3.0×10^6	1.5×10^5 3.0×10^6	1.5×10^5 3.0×10^6
8. Feedwater Cycling	2000	2000	2000
9. Loop Out of Service	160	N/A	N/A
10. 0-15% Unit Loading	500	500	500
11. 15-0% Unit Unloading	500	500	500
12. Boron Concentration Equal.	26,400	26,400	26,400
13. Refueling	80	80	80
Upset Conditions			
1. Loss of Load	80	80	80
2. Loss of Power	40	40	40
3. Partial Loss of Flow	80	80	80
4. Reactor Trip			
Case A	230	230	230
Case B	160	160	160
Case C	10	10	10
5. Inadvertent RCS Depressurization	20	20	20
6. Inadvertent Startup of an Inactive Loop	10	N/A	N/A
7. Control Rod Drop	80	80	80
8. Inadvertent Safety Injection	60	60	60
9. Excessive Feedwater	30	30[1]	30[1]
10. RCS Cold Overpressurization	N/A	N/A	10

[1] Cycles specified by equipment specification (Reference 1).

Table 5.6-2 (cont.) Applicable Design Transients and Cycle Count Comparison			
Plant Condition	**Equipment Specs. (Reference 1 & 2), Cold Leg**	**T_{HOT} Reduction Cold Leg**	**Uprating Cold Leg**
<u>Emergency Conditions</u>			
1. Small LOCA	5	5	5
2. Small Steam Line Break	5	5	5
3. Complete Loss of Flow	5	5	5
<u>Faulted Conditions</u>			
1. Large LOCA	1	1	1
2. Large Steam Line Break	1	1	1
3. Feedwater Line Break	1	1	1
4. RCP Locked Rotor	1	1	1
5. Control Rod Ejection	1	1	1
6. Steam Generator Tube Rupture	1	N/A	N/A
<u>Test Conditions</u>			
1. Turbine Roll Test	20	20	20
2. Primary Side Hydrostatic Test	10	10	10
3. Primary Side Leak Test	200	200	200
4. Secondary Side Leak Test	80	80	80

Table 5.6-3 Normal/Upset Condition Stress Summary						
Non-Bolting Material Analyzed to Code Section NB-3200						
Component	**$(P_L$ or $P_m)+P_b+Q$ Range[1]**		**Fatigue Usage**		**Thermal Stress Ratchet**	
	Calc. (psi)	**Allow. (psi)**	**Calc.**	**Allow.**	**Calc. (psi)**	**Allow. (psi)**
Casing	89,600[2] (35,560)	$3S_m$ = 58,570 (48,630)	0.3069	<1.0	54,380	76,860
Main Flange	44,250	$3S_m$ = 60,000	0.1496	<1.0	11,890	67,480
Thermal Barrier Flange	55,900	$3S_m$ = 60,000	0.4462	<1.0	43,310	78,370

Bolting Material Analyzed to Code Section NB-3200						
	P_m		**$(P_L$ or $P_m)+P_b$**		**Fatigue Usage**	
	Calc. (psi)	**Allow. (psi)**	**Calc. (psi)**	**Allow. (psi)**	**Calc.**	**Allow.**
Main Flange Bolt	44,180	$2S_m$ = 73,600	110,900	$3S_m$ = 113,400	0.9811	<1.0

(1) Value in parentheses is for thermal bending removed.

(2) Normal and Upset conditions were substantiated by simplified elastic-plastic analysis per ASME Code Section NB-3228.

5.6.2 RCP Motor

The worst case loads for the RCP motors were calculated for the Byron and Braidwood power uprated conditions. Using the revised loads, all of the RCP motors were evaluated in the four areas where parameter changes affect performance. These areas are continuous operation at the revised hot loop rating, continuous operation at revised cold loop rating, starting, and the loads on thrust bearings. The results of the evaluation follow.

5.6.2.1 Continuous Operation at Revised Hot Loop Rating

The motor is required to drive the pump continuously under hot loop conditions without exceeding a stator winding temperature rise of 75°C (corresponding to the NEMA Class B temperature rise limit in a 50°C ambient).

The worst case hot loop load under the revised operating conditions is 6981 HP. This revised load does not exceed the nameplate rating of the motor (7000 HP). The motors have been shown by test to operate within the specification limits at the hot loop nameplate rating. Therefore, continuous operation at the revised load is acceptable.

5.6.2.2 Continuous Operation at Revised Cold Loop Rating

The new cold loop rating of 8940 HP exceeds the nameplate cold loop rating of the motor by 2.17%. Analysis indicates that this increase could cause the cold loop temperature rise to exceed the NEMA guaranteed limit for a Class F winding (100°C in a 50°C Ambient) by about 3°C. Therefore, operation at full cold loop conditions will result in accelerated thermal aging of the insulation. By design, the motors are intended to operate under cold loop conditions for a maximum of 3000 hours. Exceeding the Class F limit by 3°C during 3000 hours of operation can be expected to accelerate the aging and reduce life by approximately 700 hours (~1 month) from the 40 year design life. The analysis and the operation time at the cold shutdown condition is conservative, ComEd will continue to monitor motor operation at cold loop conditions and perform routine stator testing, with visual inspection of the stator, to detect any possible stator insulation degradation.

5.6.2.3 Starting

The motor is required to start across the line with a minimum 80% starting voltage, against the reverse flow of the other pumps running at full speed, under cold loop conditions. The limiting component for this type of starting duty is the rotor cage winding. A conservative analysis was used to determine if the cage winding temperature will exceed the design limits (300°C on the bars and 50°C on the resistance rings).

Using the new load torque curve, the starting temperature rise for the rotor bars and resistance rings has been calculated. The results show bar temperature of 228.0°C and ring temperature of 24.90°C. These temperatures do not exceed design limits. Therefore, the motor can safely accelerate the load under worst case conditions.

5.6.2.4 Loads on Thrust Bearings

Excessive or inadequate loading can adversely affect performance of the thrust bearings in an RCP motor. The axial impeller down thrust for the revised parameters decreased from 55,000 lbs. to 53,420 lbs. for hot loop operation and from 75,000 lbs. to 72,434 lbs. for cold loop operation. The decrease in the impeller down thrust loads results in an increase in the overall up thrust load in the bearing from 101,200 lbs. to 102,780 for the hot operation, and a decrease in the overall down thrust load for the cold operation from 96,500 lbs. to 93,934 lbs. for the cold operation. These changes are not significant and thus the thrust bearings are considered acceptable for the revised loads.

5.7 Steam Generators

Byron and Braidwood Units 1 have replacement Babcock & Wilcox, Incorporated (BWI) steam generators installed. Byron and Braidwood Units 2 have the original Model D5 steam generators installed. Design differences between the two steam generator models required separate evaluations at uprated conditions.

BWI steam generator power uprate evaluations are addressed in Section 5.7.1. D5 steam generator power uprate evaluations are addressed in Section 5.7.2.

5.7.1 BWI Steam Generators

The Byron and Braidwood Unit 1 Replacement Steam Generators (RSG) were analyzed at the uprated power conditions in the areas of structural acceptability, thermal-hydraulics, U-bend fatigue, tube degradation and tube plugging and repair.

5.7.1.1 Structural Evaluation

5.7.1.1.1 Introduction

Uprating of Byron Unit 1/Braidwood Unit 1 to 3600.6 MWt NSSS power will incorporate steam generator tube plugging (SGTP) for 0% to 5% maximum in any steam generator. The uprating will also include operation at high and low temperature conditions.

5.7.1.1.2 Input Parameters and Assumptions

The input parameters used for the steam generator structural evaluation are given in Table 5.7.1.1-1. NSSS design transients considered for the RSG structural qualification for 5% power uprate conditions are the same as for the previous analysis at 3425 MWt except for primary and secondary side fluid variations. Primary and secondary side fluid variations for the individual design transients at 5% power uprate were developed and used in the RSG structural qualification evaluation.