APPENDIX A - APPLICATION OF NRC REGULATORY GUIDES

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DRAWINGS CITED IN THIS APPENDIX*

*The listed drawings are included as "General References" only; i.e., refer to the drawings to obtain additional detail or to obtain background information. These drawings are not part of the UFSAR. They are controlled by the Controlled Documents Program.

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REGULATORY GUIDE 1.1

NET POSITIVE SUCTION HEAD FOR EMERGENCY CORE COOLING AND CONTAINMENT HEAT REMOVAL SYSTEM PUMPS

The Licensee meets all objectives set forth in Revision 0 of Regulatory Guide 1.1 (Safety Guide 1) as presented in Subsection 6.3.2.2.

REGULATORY GUIDE 1.2

THERMAL SHOCK TO REACTOR PRESSURE VESSELS

Westinghouse follows all recommendations of Revision 0 of Regulatory Guide 1.2 (Safety Guide 2). The guide Position C.1 is followed by Westinghouse's own analytical and experimental programs as well as by participation in the Heavy Section Steel Technology (HSST) Program at Oak Ridge National Laboratory.

Analytical techniques have been developed by Westinghouse to perform fracture evaluations of reactor vessels under thermal shock loadings.

Under the heavy section steel technology program, a number of 6-inch thick 39-inch OD steel pressure vessels containing carefully prepared and sharpened surface cracks are being tested. Test conditions include both hydraulic internal pressure loadings and thermal shock loadings. The objective of this program is to validate analytical fracture mechanics techniques and demonstrate quantitatively the margin of safety inherent in reactor pressure vessels.

A number of vessels have been tested under hydraulic pressure loadings, and results have confirmed the validity of fracture analysis techniques. The results and implications of the hydraulic pressure tests are summarized in Oak Ridge National Laboratory report ORNL-TM-T5909.

Six thermal shock experiments have been completed and are now being evaluated. For representative conditions, flaws are shown to initiate and arrest in a predictable manner. These tests have demonstrated the applicability of presently used fracture assessment procedures to both high and low toughness vessel, as shown in reports ORNL/NUREG-40 and ORNL-6187.

Fracture toughness testing of irradiated compact tension fracture toughness specimens has been completed. The complete postirradiation data on 0.394-inch, 2-inch, and 4-inch thick specimens are now available from the HSST program. Both static and dynamic postirradiation fracture toughness data have been obtained. Evaluation of the data obtained to date on material irradiated to fluences between 2.2 and 4.5 x 10^{19} n/cm^2 indicates that the reference toughness curve as contained in the ASME Section III Code remains a conservative lower bound for toughness values for pressure vessel steels.

Details of progress and results obtained in the HSST program are available in the heavy section steel technology program progress reports, issued by Oak Ridge National Laboratory.

Regulatory Position C.2 is followed inasmuch as no significant changes have been made in approved core or reactor designs.

The guide position C.3 is followed since the vessel design does not include the use of an engineering solution to assure adequate recovery of the fracture toughness properties of the vessel material. If additional margin is needed, the reactor vessel can be annealed at any point in its service life. This solution is already feasible, in principle, and could be performed with the vessel in place.

An assessment of pressurized thermal shock events for the Byron and Braidwood Stations has been documented in response to 10 CFR 50.61, which appears in Subsection 5.3.1.5.1.

These requirements remain in effect even though Regulatory Guide 1.2 was withdrawn on June 17, 1991. It has been superseded by 10 CFR 50.61 and Regulatory Guide 1.154.

REGULATORY GUIDE 1.3

ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT FOR BOILING WATER REACTORS

This guide is pertinent to BWRs only.

REGULATORY GUIDE 1.4

ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT FOR PRESSURIZED WATER REACTORS

The requirements in Revision 2 of this guide have been adhered to in all pertinent sections of this application. The meteorology assumptions from the guide are detailed in Subsections 2.3.4 and 2.3.5. The guide assumptions on radioisotope releases are detailed in Section 15.6.5, as are the assumptions on containment spray effectiveness.

This guide, although used in the original plant design, has been superseded by Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors”.

REGULATORY GUIDE 1.5

ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A STEAM LINE BREAK ACCIDENT FOR BOILING WATER REACTORS

Regulatory Guide 1.5 (Safety Guide 5) is pertinent to BWRs only.

REGULATORY GUIDE 1.6

INDEPENDENCE BETWEEN REDUNDANT STANDBY (ONSITE) POWER SOURCES AND BETWEEN THEIR DISTRIBUTION SYSTEMS

The Licensee complies with Revision 0 of this regulatory guide. Refer to Subsections 8.1.1 and 8.1.6 for further information.

REGULATORY GUIDE 1.7

CONTROL OF COMBUSTIBLE GAS CONCENTRATIONS IN CONTAINMENT FOLLOWING A LOSS-OF-COOLANT ACCIDENT

Per 10 CFR 50.44 and Technical Specification Amendments Nos. 143 and 137 for Byron Station, Units 1 and 2 and Braidwood Station, Units 1 and 2, respectively, Regulatory Guide 1.7 Revision 2 is no longer applicable for Byron and Braidwood. See Subsection 6.2.5 for further discussion.

REGULATORY GUIDE 1.8

PERSONNEL SELECTION AND TRAINING

The Licensee complies with Revision 1 of this regulatory guide. Refer to Sections 13.1 and 13.2 for further information. In addition, the STA training program incorporates Revision 2 of this regulatory guide, which has more restrictive requirements than Revision 1.

Training and personnel selection for the radiation protection program is discussed in Section 12.1 and Section 12.5.

REGULATORY GUIDE 1.9

SELECTION, DESIGN, QUALIFICATION AND TESTING OF DIESEL-GENERATOR UNITS USED AS CLASS 1E ONSITE ELECTRIC POWER SYSTEMS AT NUCLEAR POWER PLANTS

Regulatory Guide (RG) 1.9, Revision 3, endorses IEEE Standard 387-1984, "IEEE Standard Criteria for Diesel Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations." In addition to this standard, RG 1.9, Revision 3, provides supplemental regulatory positions. The Licensee complies with IEEE Standard 387-1984 and these supplemental regulatory positions in Revision 3 with the following clarifications regarding:

1. Regulatory Position C.1.4

Due to the high transformer in-rush current, the voltage may dip below the required limit of 75% of nominal upon energizing the 480-Volt substation transformers and their auxiliary loads. However, this dip is of a very short duration (0.2 to 0.5 seconds) and will occur immediately after the diesel generator breaker is closed. Since there is a 1.8-to-2.0 second delay between diesel generator breaker closure and the reset of the bus undervoltage relays, the voltage will have recovered to the required limits prior to the beginning of load sequencing.

Exception is taken to the minimum frequency design requirement during emergency load sequencing of 95 percent of nominal (57 Hz). The design limitations of available emergency diesel generator (EDG) replacement subcomponents, in conjunction with existing EDG performance limitations, may preclude achieving this transient loading performance requirement as a result of maintenance or modification. This situation has manifested itself at Byron Station as a result of the modification of the obsolete EDG electronic governing system with its modernized replacement. Subsequent modification testing and modeling determined that the performance limitations of the new electronic governor, in conjunction with combustion air (turbocharger) limitations, resulted in a momentary frequency drop below 57 Hz during the start of the electric-driven auxiliary feedwater pump. This frequency drop could not be compensated for by the governor dynamic adjustment or equipment maintenance.

Evaluations have been performed that justify short duration frequency drops below 57 Hz during large pump motor starts. Consideration was given to the effects of the transient on ECCS flow requirements, pumps/motors, motor-operated valves, battery chargers, instrument inverters and diesel generator protection. It was determined that the affected systems and components will meet their intended safety-related design basis functions for short periods below 57 Hz during emergency load sequencing.

A clarification is provided for the line item, "Frequency should be restored to within 2 percent of nominal in less than 60 percent of each load-sequence interval for step load increase."

Starting the containment spray (CS) pump in the load sequence is dependent on when the Hi-3 containment pressure signal is reached. As described in Table 8.3-5, for train A, if containment spray actuation is not required at 27 seconds after a safety injection actuation signal, automatic start of the CS pump is blocked until all other loads are sequenced on to the EDG. Train A has an additional auxiliary feedwater (AF) pump load. Therefore, the CS pump starting time in the load sequence may either be at 27 seconds, which is the more severe EDG load case, or after 52 seconds.

At the 27-second sequence case, the frequency is restored as recommended in the Regulatory Guide. In the greater than 52-second sequence case, there is a possibility of a 5-second load-sequence interval between the AF pump start and the CS pump start. EDG performance modeling under full-flow conditions indicates that the AF pump exceeds the 60-percent loading interval frequency recovery guidelines. Even though the frequency undershoot has cleared, a potential frequency overshoot greater than 2 percent of nominal frequency may still be momentarily present beyond the 60-percent loading interval frequency guideline. Although a time interval of greater than 60 percent of the loading interval may be required for the frequency to recover, worst-case analytical analysis and actual EDG testing have demonstrated that the frequency will recover to within 2 percent of nominal prior to the potential loading of the CS pump on to the EDG. Once the load sequencing has been completed, the EDG will operate at nominal frequency ±2 percent, as described in the Regulatory Guide.

2. Regulatory Position C.2.2

Exception is taken to the last sentence in this paragraph, "Jumpers and other nonstandard configurations or arrangements should not be used subsequent to initial equipment startup conditions." In order to successfully accomplish certain diesel tests it is necessary to use jumpers to simulate particular engine signals. The use of jumpers is a normal practice in diesel engine testing, and the safe use of the jumpers is ensured with detailed procedures, which include independent verification of circuit restoration.

3. Regulatory Position C.2.2.1, "Start Test"

Each EDG undergoes a startup test on a monthly basis from "standby conditions." Once every 6 months this test is supplemented by verifying proper start up from "normal standby conditions." This test is covered in paragraph C.2.2.3, and is further discussed in Item 4 below.

4. Regulatory Positions C.2.2.1, C.2.2.4, C.2.2.5, and C.2.2.6

Exception has been taken against use of the term "standby conditions" to denote "normal standby conditions."

The term "standby condition" is interpreted as any conditional state of the EDG in which the EDG is considered operable. More specifically, standby conditions for an EDG refer to a condition whereby the diesel engine lube oil is being continuously circulated and engine jacket water and lube oil temperatures are consistent with manufacturer's recommended operating range (low lube oil and jacket water temperature alarm settings to the high lube oil and jacket water temperature alarm settings).

The term "normal standby condition" defines a conditional state of the EDG in which lube oil and jacket water temperatures are within the prescribed temperature bands of these subsystems when the EDG has been at rest for an extended period of time with the prelube oil and jacket water circulating systems operational. It should be noted that the semiannual fast start test described in paragraph C.2.2.3 is performed from "normal standby conditions."

5. Regulatory Position C.2.2.6, "Combined SIAS and LOOP Tests"

Exception is taken to the statement that the EDG be tested for proper response to a LOOP in conjunction with a safety injection actuation signal (SIAS) in whatever sequence they might occur. The EDGs are tested for response to a loss of offsite power (LOOP), to a SIAS, and to a LOOP and SIAS when they occur concurrently. Performing a LOOP/SIAS EDG test in whatever sequence they might occur is beyond the original licensing basis, provides no additional value, and was not included in Regulatory Guide 1.9, Revision 2 or Regulatory Guide 1.108.

6. Regulatory Position C.2.2.7, "Single-Load Rejection Test"

Exception is taken to the specified power factor requirements of this paragraph. This test is performed with the diesel generator in the isochronous mode of operation. In this mode of operation, the power factor is a function of the generator loads only and cannot be varied by voltage or speed adjustment.

7. Regulatory Position C.2.2.8, "Full-Load Rejection Test"

Exception is taken to the power factor range of between 0.8 and 0.9. In order to ensure the DG is tested under load conditions that bound design conditions and comply with the recommendations of Regulatory Guide 1.9, testing will be performed using an upper limit on power factor of ≤ 0.89. This power factor range bounds the actual design basis inductive loading the DG would experience. Since this testing is performed with the diesel generator synchronized with offsite power, grid conditions may not permit achieving a power factor of ≤ 0.89. In this case, the required power factor limit is not required to be met, however, the power factor shall be maintained as close to the limit as practicable.

8. Regulatory Position C.2.3.2.3, “Refueling Outage Testing”

Exception is taken to the statement that the overall emergency diesel generator unit design capability should be demonstrated at every refueling outage by performing the tests identified in Table 1 of Regulatory Guide 1.9. Refueling Outage Testing as identified in Table 1 of Regulatory Guide 1.9 is performed in accordance with the Technical Specifications, and the test interval may be supplanted with performance-based, risk-informed test intervals. This statement in Regulatory Position C.2.3.2.3 is in accordance with Section 6.5.2 of IEEE Standard 387-1984. By taking exception to Regulatory Position C.2.3.2.3, exception is also being taken to the statement in Section 6.5.2 of IEEE Standard 387-1984 that the diesel generator unit shall be given one cycle of each of the specified tests at least once every 18 months to demonstrate its continued capability of performing its required function.

Compliance with the requirements of this guide is described further in Subsections 8.1.2, 8.1.20, 8.3.1.1.2.2 and 8.3.1.2. Therefore, the Licensee meets the objectives set forth in this regulatory guide.

REGULATORY GUIDE 1.10

MECHANICAL (CADWELD) SPLICES IN REINFORCING BARS OF CATEGORY I CONCRETE STRUCTURES

Subsection B.2.3 of Appendix B describes conformance to the regulatory positions in Revision 1 of Regulatory Guide 1.10. The requirements remain in effect even though the regulatory guide was withdrawn on July 8, 1981. The regulatory positions are now covered by one or more national standards.

REGULATORY GUIDE 1.11

INSTRUMENT LINES PENETRATING PRIMARY REACTOR CONTAINMENT

The Licensee complies with the requirements of Regulatory Guide 1.11 (Safety Guide 11) and its supplement as discussed in Subsection 7.1.2.5.

REGULATORY GUIDE 1.12

INSTRUMENTATION FOR EARTHQUAKES

The seismic instrumentation plans for Byron and Braidwood Stations satisfy the Regulatory Guide 1.12, Revision 1, requirements for plants with maximum foundation accelerations of 0.3g or less. Byron and Braidwood Stations have been designed for a foundation acceleration of 0.2g. A comparison between requirements of Regulatory Guide 1.12 and the Byron/Braidwood seismic instrumentation plan is given in the following.

As required by Regulatory Guide 1.12, Part C Regulatory Position 1, three triaxial time-history accelerographs - one in the free field, one on the containment building foundation, and one on the containment shell wall are provided. Additional triaxial time-history accelerograph sensors are provided on the containment refueling floor and at the foundation of the river screen house (Byron).

One triaxial peak accelerograph capable of measuring the absolute peak acceleration in three orthogonal directions, coinciding with the major axes of the analytical building model, is provided at each of the following three locations:

a. on the accumulator tank in the containment building,

b. on the safety injection piping in the containment building, and

c. on the essential service return piping in the auxiliary building.

These locations satisfy the requirements of Regulatory Guide 1.12, Part C, Regulatory Position 1, Sections a.1, a.2, and a.3.

A triaxial response spectrum recorder capable of measuring both horizontal and vertical motion and capable of providing signals for immediate control room indication is provided on the containment building base slab. The location and specification of this recorder is in compliance with Section b, Part C of Regulatory Guide 1.12.

Two additional triaxial response spectrum recorders, with the same specifications as above are provided on the floor of the counting room in the auxiliary building, and on the operating floor of the containment building. These comply with the requirements of Sections c.1 and c.2 of Part C, Regulatory Position 1.

Section c.3 of Part C, Regulatory Position 1 calls for a separate triaxial response spectrum recorder capable of measuring both horizontal motions and the vertical motion to be provided at the foundation of an independent Seismic Category I structure where the response is different from that of the reactor containment structure.

Except for the Byron river screen house, all the structures are founded on rock, and will have the same foundation response as the containment structure. The triaxial time-history accelerograph sensor provided at the foundation level of the Byron river screen house is used to determine the response spectra for this location, using the playback unit provided in the control room.

The specifications of all the response spectrum recorders, such as dynamic range, frequency range, damping, etc., satisfy the requirements of Part C, Regulatory Positions 4 and 5.

In general, the seismic instrumentation plan for Byron and Braidwood Stations complies with Regulatory Guide 1.12.

Additional information on instrumentation for earthquakes is provided in Subsection 3.7.4.1 and 3.7.4.2.

REGULATORY GUIDE 1.13

SPENT FUEL STORAGE FACILITY DESIGN BASIS

The plant design conforms with the requirements in Revision 1 of this guide as presented in Subsections 9.1.2, 9.1.3, 9.1.4, 6.5.1, and 9.4.5.

REGULATORY GUIDE 1.14

REACTOR COOLANT PUMP FLYWHEEL INTEGRITY

The design meets the requirements in Revision 1 of Regulatory Guide 1.14 with the exceptions noted below. The shaft and the bearings supporting the flywheel are capable of withstanding any combination of the normal operating loads, anticipated transients, the design-basis LOCA, and the safe shutdown earthquake loads.

Since the issuance of Regulatory Guide 1.14, Revision 1, the NRC Staff has provided to Westinghouse a copy of Draft 2, Revision 2, of Regulatory Guide 1.14 (via an April 12, 1976, letter from Robert B. Minogue to C. Eicheldinger). This draft was formulated from industry and concerned parties' comments. It is significant that the Draft 2 version incorporates several of the Westinghouse comments on Revision 1. Since Draft 2 has not been formally published as Revision 2 of Regulatory Guide 1.14, the exceptions and clarifications (from the original Westinghouse comments) are provided in the following:

a. Post-Spin Inspection

Westinghouse has shown in WCAP-8163, "Topical Report on Reactor Coolant Pump Integrity in LOCA," that the flywheel would not fail at 290% of normal speed for a flywheel flaw of 1.15 inches or less in length. Results for a double ended guillotine break at the pump discharge with full separation of pipe ends assumed, show the maximum overspeed was calculated in WCAP-8163 to be about 280% of normal speed for the same postulated break, and an assumed instantaneous loss of power to the reactor coolant pump. In comparison with the overspeed presented above, the flywheel could withstand a speed up to 2.3 times greater than the flywheel spin test speed of 125% provided that flaws no greater than 1.15 inches are present. If the maximum speed were 125% of normal speed or less, the critical flaw size for failure would exceed 6 inches in length. Nondestructive tests and critical dimension examinations are all performed before the spin tests. The inspection methods employed (described in WCAP-8163) provide assurance that flaws significantly smaller than the critical flaw size of 1.15 inches for 290% of normal speed would be detected. Flaws in the flywheel will be recorded in the prespin inspection program (see WCAP-8163). Flaw growth attributable to the spin test (i.e., from a single reversal of stress, up to speed and back), under the most adverse conditions, is about three orders of magnitude smaller than that nondestructive inspection techniques are capable of detecting. For these reasons, Westinghouse does not perform postspin inspections and believes the prespin test inspections are adequate.

b. Interference Fit Stresses and Excessive Deformation

Much of Revision 1 deals with stresses in the flywheel resulting from the interference fit between the flywheel and the shaft. Because Westinghouse's design specifies a light interference fit between the flywheel and the shaft; at zero speed, the hoop stresses and radial stresses at the flywheel bore are negligible. Centering of the flywheel relative to the shaft is accomplished by means of keys and/or centering devices attached to the shaft, and at normal speed, the flywheel is not in contact with the shaft in the sense intended by Revision 1. Hence, the definition of "Excessive Deformation," as defined in Revision 1 of Regulatory Guide 1.14, is not applicable to the Westinghouse design since the enlargement of the bore and subsequent partial separation of the flywheel from the shaft does not cause unbalance of the flywheel. Extensive Westinghouse experience with reactor coolant pump flywheels installed in this fashion has verified the adequacy of the design.

Westinghouse's position is that combined primary stress levels, as defined in Revision 0 of Regulatory Guide 14 (C.2), (a) and (c), are both conservative and proven and that no changes to these stress levels are necessary. Westinghouse designs to these stress limits and thus does not have permanent distortion of the flywheel bore at normal or spin test conditions.

c. Section B, Discussion of Cross Rolling Ratio of 1 to 3

Cross Rolling Ratio - Westinghouse's position is that specification of a cross rolling ratio is unnecessary since past evaluations have shown that ASME SA-533-B Class 1 materials produced without this requirement have suitable toughness for typical flywheel applications. Proper material selection and specification of minimum material properties in the transverse direction adequately ensure flywheel integrity. An attempt to gain isotropy in the flywheel material by means of cross rolling is unnecessary since adequate margins of safety are provided by both flywheel material selection (ASME SA-533-B Class 1) and by specifying minimum yield and tensile levels and toughness test values taken in the direction perpendicular to the maximum working direction of the material.

d. Section C, Item 1a: Relative to Vacuum-melting and Degassing Process or the Electroslag Process

Vacuum Treatment - The requirements for vacuum melting and degassing process or the electroslag process are not essential in meeting the balance of the Regulatory Position nor do they, in themselves, ensure compliance with the overall Regulatory Position. The initial Safety Guide 14 stated that the "flywheel material should be produced by a process that minimized flaws in the material and improves its fracture toughness properties." This is accomplished by using SA-533 material including vacuum treatment.

e. Section C, Item 2b: Westinghouse interprets this paragraph to mean:

Design Speed Definition - Design speed should be 125% of normal speed or the speed to which the pump motor might be electrically driven by station turbine generator during anticipated transients, whichever is greater. Normal speed is defined as the synchronous speed of the a-c drive motor at 60 Hz.

f. Section C, Item 4b: Inservice Inspection of Reactor Coolant Pump Flywheel

For reactor coolant pump motor serial numbers 4S88P961 and 1S88P961, in lieu of Regulatory Position c.4.b(1) and c.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheel may be conducted at approximately 10 year intervals coinciding with the Inservice Inspection schedule as required by ASME Section XI.

For all other reactor coolant pump motors, in lieu of Regulatory Position c.4.b(1) and c.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheel may be conducted at an interval not to exceed 20 years.

The requirements for examination procedures and acceptance criteria as described in Regulatory Guide 1.14 will be followed. This inspection program meets the intent of Regulatory Guide 1.14 in assuring the continued integrity of the reactor coolant pump flywheels.

The flywheel integrity is described in Subsections 5.4.1.5.2 and 5.4.1.5.3.

REGULATORY GUIDE 1.15

TESTING OF REINFORCING BARS FOR CATEGORY I CONCRETE STRUCTURES

Subsection 3.8.3.6, Table 3.8-2, and Section B.2 of Appendix B describe conformance to the regulatory positions in Revision 1 of Regulatory Guide 1.15. The requirements remain in effect even though the regulatory guide was withdrawn on July 8, 1981. The regulatory positions are now covered by one or more national standards.

REGULATORY GUIDE 1.16

REPORTING OF OPERATING INFORMATION - APPENDIX A TECHNICAL SPECIFICATIONS

The reporting of specific operating information described in Regulatory Guide 1.16, Revision 4 is no longer applicable to Byron and Braidwood as a result of the issuance of Technical Specification Amendment Nos. 142 and 136 for Byron Station, Units 1 and 2 and Braidwood Station, Units 1 and 2, respectively.

The Licensee complies with the reporting requirements contained in Title 10 of the Code of Federal Regulations and stations' Technical Specifications and Technical Requirements Manual.

REGULATORY GUIDE 1.17

PROTECTION OF NUCLEAR POWER PLANTS AGAINST INDUSTRIAL SABOTAGE

The Industrial Security Plan for the Byron and Braidwood Stations has been provided to the Regulatory Staff on a proprietary basis. No comparison of the plan and this regulatory guide is provided in this response. Compliance with the requirements of Revision 1 of Regulatory Guide 1.17 is presented in Section 13.6 and Subsection 9.5.2.2. The requirements remain in effect even though the regulatory guide was withdrawn on May 21, 1991.

REGULATORY GUIDE 1.18

STRUCTURAL ACCEPTANCE TEST FOR CONCRETE PRIMARY REACTOR CONTAINMENTS

The structural acceptance test conformed to the requirements of the ASME Code Section III, Division 2/ACI 359-80 Article CC-6000, as stated in Subsection 3.8.1.7.2.2. These requirements supersede those endorsed in Regulatory Guide 1.18, which was withdrawn on July 8, 1981.

REGULATORY GUIDE 1.19

NONDESTRUCTIVE EXAMINATION OF PRIMARY CONTAINMENT LINER WELDS

The plant design conforms to the regulatory positions in Revision 1 of Regulatory Guide 1.19 (Safety Guide 19) as described in Sections B.5 and B.6. The requirements remain in effect even though the regulatory guide was withdrawn on July 8, 1981. The regulatory positions are now covered by one or more national standards.

REGULATORY GUIDE 1.20

COMPREHENSIVE VIBRATION ASSESSMENT PROGRAM FOR REACTOR INTERNALS DURING PREOPERATIONAL AND INITIAL STARTUP TESTING

The requirements in Revision 2 of Regulatory Guide 1.20 are met. Refer to Subsection 3.9.2.4 for further discussion.

REGULATORY GUIDE 1.21

MEASURING, EVALUATING, AND REPORTING RADIOACTIVITY IN SOLID WASTES AND RELEASES OF RADIOACTIVE MATERIALS IN LIQUID AND GASEOUS EFFLUENTS FROM LIGHT-WATER-COOLED NUCLEAR POWER PLANTS

The Licensee conforms with Revision 1 of this regulatory guide with the following clarifications keyed to paragraph numbers in the regulatory position. Exception is taken to the biannual reporting requirement specified in Regulatory Guide 1.21, Revision 1 since the reporting requirement in 10 CFR 50.36a was revised from biannual to annual in 1996 (reference 61 FR 39299).

1. Hourly meteorological data are recorded for all periods throughout the year, and quarterly summaries are reported. A separate meteorological data base for periods of batch releases is not provided.

10. If multiple sample points are given for detection of radioiodine and readings are below the threshold of detection, the threshold limits are not summed over the number of sample points to give the total release rate.

13. Radiological impact on man is provided for the maximum exposed individual.

    Appendix B:

    E.1. Total body and significant organ doses to individual from receiving-water-related pathways are provided for the maximum exposed individual.

    E.3. Organ doses to individuals in unrestricted areas from radioactive iodine and radioactive material in particulate form from all exposure pathways of exposure are provided for the maximum exposed individual.

    E.4. Total body doses to individuals and populations from direct radiation from the facility are incorporated for the maximum exposed individual in 10 CFR 20 calculations.

    E.5. Total body doses to the population and average doses to individuals in the population from all receiving-water-related pathways are not included.

    E.6. Total body doses to the population and average doses to individuals in the population from gaseous effluents to a distance of 50 miles from the site are not included.

14. Sensitivities in Appendixes A and B of this guide may not be practicable. These releases are measured to the lowest levels consistent with existing technology.

    Appendix B:

    D.1. The total quantity of solid waste in cubic meters and curies is summed for each quarter.

Assurance of measuring very low levels of radioactivity is subject to interpretation of readout which may be effected by noise level, calibration, radiation, background, etc. In addition, the instrument sensitivity required to assure compliance with the guide may not be available with current technology.

REGULATORY GUIDE 1.22

PERIODIC TESTING OF PROTECTION SYSTEM ACTUATION FUNCTIONS

The Licensee complies with Revision 0 of Regulatory Guide 1.22 (Safety Guide 22). Refer to Subsections 7.1.2.6, 7.2.2, 7.3.2.2, 8.1.3, 8.3.1.2 and 12.3.4.1 for further information.

REGULATORY GUIDE 1.23

ONSITE METEOROLOGICAL PROGRAMS

The Licensee complies with Revision 1 of Regulatory Guide 1.23 with the following clarification:

Based on a commitment made to the NRC during the implementation of Alternative Source Terms, finer wind speed categories provided in the latest appropriate regulatory guidance are to be used the next time the dose consequence calculations associated with the LOCA, MSLB, CREA, LRA, SGTR, and FHA events are revised. Finer wind speed categories from Regulatory Guide 1.23 Rev. 1 have been used in the latest revised analyses.

Refer to Subsection 2.3.3 for further information.

REGULATORY GUIDE 1.24

ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A PRESSURIZED WATER REACTOR RADIOACTIVE GAS STORAGE TANK FAILURE

The Licensee complies with the regulatory position in Revision 0 of Regulatory Guide 1.24 (Safety Guide 24) as presented in Subsection 15.7.1.3.

REGULATORY GUIDE 1.25

ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT IN THE FUEL HANDLING AND STORAGE FACILITY FOR BOILING AND PRESSURIZED WATER REACTORS

The NSSS vendor's practice and recommendations are in agreement with Revision 0 of Regulatory Guide 1.25 (Safety Guide 25), except that footnote C.1.c cannot be met. This footnote states that the average burnup for the peak assembly should be 25,000 MWd/ton or less. The lead rod average burnup for the peak assembly is in the 60,000 MWd/ton range for Westinghouse fuel. (See Subsections 15.7.4.2 and 15.7.4.3.)

This guide, although used in the original plant design, has been superseded by Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors".

REGULATORY GUIDE 1.26

QUALITY GROUP CLASSIFICATIONS AND STANDARDS FOR WATER-, STEAM-, AND RADIOACTIVE-WASTE-CONTAINING COMPONENTS OF NUCLEAR POWER PLANTS

The Licensee complies with the regulatory positions stated in Revision 3 of this regulatory guide. However, exceptions to this regulatory guide may be taken if replacement components, parts, or materials are no longer available as ASME Section III items. These replacements will be purchased according to the Exelon Generation Company Quality Assurance program but may not be certified to the ASME Code. This practice is consistent with the NRC Staff position defined in Generic Letter 89-09. A list of all components, parts, and materials replaced according to this practice are included in Table A1.26-1. Refer to Subsection 3.2.2 and Table 3.2-1 for additional information on the classification of structures, components, and systems.

TABLE A1.26-1

COMPONENT, PART, OR MATERIAL	REFERENCE	SAFETY CAT.	QUALITY GRP	ELECTRICAL
1. Control Room Chiller/ Condenser Tubes	R. A. Flahive to D. Wozniak 9/27/89	I	C	n/a

TABLE A1.26-1

ASME SECTION III SUBSTITUTION COMPONENTS AND PARTS

COMPONENT, PART, OR MATERIAL	REFERENCE	SAFETY CAT.	QUALITY GRP	ELECTRICAL
Emergency Diesel Generator Lube Oil Strainers 1/2DG02MA/B, 1/2DG03MA/B & 1/2DG06MA/B.	Parts Evaluation No. A-1991-122-0	I	C	N/A
Emergency Diesel Generator Lube Oil Thermostatic Controller Valves 1/2DG5003A/B.	Parts Evaluation No. A-1992-159-0	I	C	N/A
Emergency Diesel Generator Engine Crankcase Lube Oil Manual Fill/Drain Isolation Valves 1/2DO072A/B.	Parts Evaluation No. A-1992-227-0	I	C	N/A

REGULATORY GUIDE 1.27

ULTIMATE HEAT SINK FOR NUCLEAR POWER PLANTS

The Licensee meets all objectives set forth in Revision 2 of this regulatory guide as presented in Subsections 2.3.1, 2.4.11, 9.2.5.1, 9.2.5.2, 9.2.5.3, and Technical Specification 3.7.9.

REGULATORY GUIDE 1.28

QUALITY ASSURANCE PROGRAM REQUIREMENTS (DESIGN AND CONSTRUCTION)

The Licensee complies with the positions of the regulatory guide with the following exception:

> Regulatory Guide 1.28, Revision 3 requires the licensee to ensure the requirements of NQA-1 are met by its suppliers. Suppliers are audited to NQA-1 or ANSI/ASME N45.2-series standards. Because of the large quantity of vendors maintained on the approved bidders list, the three year frequency of audits, and some non-ASME vendors who are reluctant to revise their QA program, not all suppliers meet NQA-1.

The Requirements of Regulatory Guide 1.28 will be applied to ANSI/ASME NQA-1-1994 vice the endorsed ANSI/ASME NQA-1-1983.

Also refer to the Quality Assurance Program Topical Report NO-AA-10.

REGULATORY GUIDE 1.29

SEISMIC DESIGN CLASSIFICATION

The Licensee complies with the regulatory positions stated in Revision 3 of this regulatory guide. Refer to Subsections 3.2.1 and 3.10.1.2.1 for further information.

REGULATORY GUIDE 1.30

QUALITY ASSURANCE REQUIREMENTS FOR THE INSTALLATION, INSPECTION, AND TESTING OF INSTRUMENTATION AND ELECTRIC EQUIPMENT

Regulatory Guide 1.30 endorsed ANSI N45.2.4, Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment. NQA-1-1994, Subpart 2.4, Installation, Inspection, and Testing Requirements for Power, Instrumentation, and Control Equipment at Nuclear Facilities supersedes this commitment to Regulatory Guide 1.30 and ANSI N45.2.4 as documented in the Quality Assurance Topical Report (NO-AA-10). Refer to Subsections 3.11.2, 7.1.2.8, and 8.1.5 for further information.

REGULATORY GUIDE 1.31

CONTROL OF FERRITE CONTENT IN STAINLESS STEEL WELD METAL

The Licensee complies with the regulatory position described in Regulatory Guide 1.31, Revision 3.

The position concerning the control of delta ferrite in the stainless steel welding is discussed in Subsections 5.2.3, 5.3.1.4, and 6.1.1.1.

REGULATORY GUIDE 1.32

CRITERIA FOR SAFETY-RELATED ELECTRIC POWER SYSTEMS FOR NUCLEAR POWER PLANTS

The Licensee complies with the regulatory positions in Revision 2 of this guide with the following exceptions/clarifications:

Regulatory Position C.1.a.

See position on Regulatory Guide 1.93.

Regulatory Position C.1.d.

See position on Regulatory Guides 1.6 and 1.75.

Regulatory Position C.1.e.

See position on Regulatory Guide 1.75.

Regulatory Position C.1.f.

See position on Regulatory Guide 1.9.

Regulatory Position C.2.a.

See position on Regulatory Guide 1.81.

Regulatory Position C.2.b.

See position on Regulatory Guide 1.93.

REGULATORY GUIDE 1.33

QUALITY ASSURANCE PROGRAM REQUIREMENTS (OPERATION)

The Licensee complies with Revision 2 of this regulatory guide with the exception of regulatory positions C.2 and C.4. In lieu of specifying individual audit frequencies, audits are conducted on a performance-driven frequency not to exceed 24 months. Refer to Topical Report NO-AA-10 for further information on the Quality Assurance Program at the Byron/Braidwood Stations.

In lieu of the 45.2 daughter documents referenced in ANSI N18.7-1976/ANS 3.2, the following sections of NQA-1-1994, which incorporates the requirements of NQA-1-1989 and NQA-2-1989 into one document, will be utilized per the following matrix:

ANSI N45.2 Daughter Standard	
ANSI N45.2.1	Subpart 2.1
ANSI N45.2.2	Subpart 2.2
ANSI N45.2.3	Subpart 2.3
ANSI N45.2.5	Subpart 2.5
ANSI N45.2.6	Per Regulatory Guide 1.28 Revision 3 regulatory position C.1, Basic Requirement 2, Supplement 2S-1, and Appendix 2A-1 will be utilized; however, NQA-1-1994 will be used instead of NQA-1-1983
ANSI N45.2.8	Subpart 2.8
ANSI N45.2.9	Per Regulatory Guide 1.28 Revision 3 regulatory position C.2, Basic Requirement 17 and Supplement 17S-1 will be utilized; however, NQA-1-1994 will be used instead of NQA-1-1983.
ANSI N45.2.11	NQA-1-1994 Basic Requirement 3 and Supplement 3S-1 will be utilized.
ANSI N45.2.13	NQA-1-1994 Basic Requirement 7 and Supplement 7S-1 will be utilized.

REGULATORY GUIDE 1.34

CONTROL OF ELECTROSLAG WELD PROPERTIES

The Licensee complies with the regulatory position in Revision 0 of the guide whenever the electroslag welding process is used for components made of ferritic or austenitic materials. However, electroslag welding is not used for equipment purchased on the Licensee's specifications. (See Subsections 5.3.1.4 and 5.4.2.1.1 for further information.)

REGULATORY GUIDE 1.35

INSERVICE INSPECTION OF UNGROUTED TENDONS IN PRESTRESSED CONCRETE CONTAINMENT STRUCTURES

The requirements in Regulatory Guide 1.35 are incorporated into Section XI, IWL, 1992 Edition, 1992 Addenda, as modified by 10 CFR 50.55a(b)(2)(viii). The Licensee complies with these requirements. Refer to Subsection 3.8.1.7.3.2 for further information.

REGULATORY GUIDE 1.35.1

DETERMINING PRESTRESSING FORCES FOR INSPECTION OF PRESTRESSED CONCRETE CONTAINMENT STRUCTURES

The Licensee complies with the July 1990 edition of this regulatory guide. Refer to Subsection 3.8.1.7.3.2 for further information.

REGULATORY GUIDE 1.36

NONMETALLIC THERMAL INSULATION FOR AUSTENITIC STAINLESS STEEL

The Licensee complies with Revision 0 of this guide.

The NSSS vendor practice meets recommendations of Regulatory Guide 1.36 but is more stringent in several respects as discussed below. (Also see paragraph 5.2.3.2 for further information.)

The nonmetallic thermal insulation used on the reactor coolant pressure boundary is specified to be made of compounded materials which yield low leachable chloride and/or fluoride concentrations. The compounded materials in the form of blocks, boards, cloths, tapes, adhesives, cements, etc., are silicated to provide protection of austenitic stainless steels against stress corrosion which may result from accidental wetting of the insulation by spillage, minor leakage, or other contamination from the environmental atmosphere. Each lot of insulation material is qualified and analyzed to assure that all of the materials provide a compatible combination for the reactor coolant pressure boundary.

The tests for qualification specified by the guide (ASTM C692-71 or RDT M12-IT) allow use of the tested insulation materials if no more than one of the metallic test samples cracks. Westinghouse rejects the tested insulation material if any of the test samples cracks.

The vendor procedure is more specific than the procedures suggested by the guide, in that the Westinghouse specification requires determination of leachable chloride and fluoride ions from a sample of the insulating materials. The procedures in the guide (ASTM D512 and ASTM D1179) do not differentiate between leachable and unleachable halogen ions.

In addition vendor experience indicates that only one of the three methods allowed under ASTM D512 and ASTM D1179 for chloride and fluoride analysis is sufficiently accurate for reactor applications. This is the "referee" method, which is used by Westinghouse. These requirements are defined in Westinghouse Process Specification PS-83336KA.

REGULATORY GUIDE 1.37

QUALITY ASSURANCE REQUIREMENTS FOR CLEANING OF FLUID SYSTEMS AND ASSOCIATED COMPONENTS OF WATER-COOLED NUCLEAR POWER PLANTS

Regulatory Guide 1.37 endorsed ANSI N45.2.1, Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants. NQA-1-1994, Subpart 2.1, Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components for Nuclear Power Plants supersedes this commitment to Regulatory Guide 1.37 and ANSI N45.2.1 as documented in the Quality Assurance Topical Report (NO-AA-10). Refer to Subsections 5.2.3, 5.4.2.1.1, and 6.1.1.1 for further information.

REGULATORY GUIDE 1.38

QUALITY ASSURANCE REQUIREMENTS FOR PACKAGING, SHIPPING, RECEIVING, STORAGE, AND HANDLING OF ITEMS FOR WATER-COOLED NUCLEAR POWER PLANTS

Regulatory Guide 1.38 endorsed ANSI N45.2.2, Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of items for Water Cooled Nuclear Power Plants. NQA-1-1994, Subpart 2.2, Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of items for Nuclear Power Plants supersedes this commitment to Regulatory Guide 1.38 and ANSI N45.2.2 as documented in the Quality Assurance Topical Report (NO-AA-10). These practices are audited for compliance in accordance with the Quality Assurance Program as described in Topical Report NO-AA-10.

REGULATORY GUIDE 1.39

HOUSEKEEPING REQUIREMENTS FOR WATER-COOLED NUCLEAR POWER PLANTS

Regulatory Guide 1.39 endorsed ANSI N45.2.3, Housekeeping Requirements for Water Cooled Nuclear Power Plants. NQA-1-1994, Subpart 2.3, Quality Assurance Requirements for Housekeeping for Nuclear Power Plants supersedes this commitment to Regulatory Guide 1.39 and ANSI N45.2.3 as documented in the Quality Assurance Topical Report (NO-AA-10).

The Licensee complies with Subpart 2.3 of ANSI/ASME NQA-1-1994. Facility cleanness, care of material and equipment, fire prevention and protection, disposal of debris, protection of material and control of access are controlled by the station. Normal management attention and periodic audits under the Quality Assurance Program provide the desired result at the Byron and Braidwood Stations. Independent audits by NRC Region III personnel also contribute to the effectiveness of good housekeeping practices. Refer to Topical Report NO-AA-10 for further information on the Quality Assurance Program.

Alternative Clarification

Byron/Braidwood shall comply with Subpart 2.3 of ANSI/ASME NQA-1-1994 except for the following alternative clarification:

Section 2.2, Classification of Cleanness Zones. Byron/Braidwood does not have any areas meeting the description of zone 1 per table listed. For the purpose of Foreign Material Exclusion, zone designations will be determined based on Work Control and Foreign Material Exclusion program requirements and controls implemented consistent with best industry practices.

Section 3.1, In lieu of a written record of the entry and exit of all personnel, Personnel Accountability for Zones 1, 2 and 3 will be controlled as determined by administrative programmatic controls for the stations Site Access, Locked Doors, Radiation Work Permits, Work Control and Foreign Material Exclusion program.

REGULATORY GUIDE 1.40

QUALIFICATION TESTS OF CONTINUOUS-DUTY MOTORS INSTALLED INSIDE THE CONTAINMENT OF WATER-COOLED NUCLEAR POWER PLANTS

NSSS Scope

It is the Westinghouse position that motors inside containment comply with the qualification control requirements of Criterion III to Appendix B to 10 CFR 50. These requirements are satisfied by qualification as described in WCAP-8587 and its supplement which contains appropriate EQDPs (equipment qualification data packages) for Westinghouse supplied continuous duty motors within the containment. The Licensee is in compliance with the objectives of Regulatory Guide 1.40, Revision 0.

Non-NSSS Scope

The Licensee complies with the requirements of Regulatory Guide 1.40, Revision 0, with the clarification to the regulatory position identified and justified below:

Regulatory Position Cl

To the extent practicable, auxiliary equipment that is part of the installed motor assembly should also be qualified in accordance with IEEE 334-1971.

Licensee's Position

Comply with regulatory position, in that to the extent practicable, auxiliary equipment essential to the safety function of the installed motor assembly will be qualified in accordance with IEEE 334-1971.

Justification of Licensee's Position

Nonessential auxiliaries have no safety function and should be excluded from the requirements.

REGULATORY GUIDE 1.41

PREOPERATIONAL TESTING OF REDUNDANT ONSITE ELECTRIC POWER SYSTEMS TO VERIFY PROPER LOAD GROUP ASSIGNMENTS

The Licensee complies with Revision 0 of this regulatory guide. Refer to Subsection 8.1.8 for further information.

REGULATORY GUIDE 1.42

INTERIM LICENSING POLICY ON AS LOW AS PRACTICABLE FOR GASEOUS RADIOIODINE RELEASES FROM LIGHT WATER-COOLED NUCLEAR POWER REACTORS

This guide was withdrawn on March 18, 1976. Based on the date that the operating license applications were docketed, the following documents were used in lieu of Regulatory Guide 1.42: Appendix I of 10 CFR 50 and Regulatory Guides 1.109, 1.111, and 1.112.

REGULATORY GUIDE 1.43

CONTROL OF STAINLESS STEEL WELD CLADDING OF LOW-ALLOY STEEL COMPONENTS

The Licensee complies with the requirements in Revision 0 of this guide. Refer to Subsection 5.3.1.4 for further information.

Westinghouse meets the intent of Regulatory Guide 1.43 by requiring qualification of any high heat input process, such as the submerged-arc wide-strip welding process and the submerged-arc-6-wire process used on SA-508 Class 2 material, with a performance test as described in Regulatory Position 2 of the guide. No qualifications are required by the regulatory guide for SA-533 material and equivalent chemistry for forging grade SA-506 Class 3 material.

REGULATORY GUIDE 1.44

CONTROL OF THE USE OF SENSITIZED STAINLESS STEEL

The Licensee complies with Revision 0 of this guide with the following clarifications keyed to paragraph numbers in the Regulatory Position.

1. The Licensee complies in general with the intent of the requirements of this guide. With regard to fabrication, shipment, storage, and construction, the Applicant requests that contaminants be avoided and cleaning solutions be halide free.

2. The Licensee complies with the requirement in that it specified ASME material specifications which require material to be supplied in the solution annealed condition.

3. The Licensee does not agree with this requirement. Specification of solution annealed material is sufficient.

4. The Licensee's specifications prohibit the use of materials that have been exposed to sensitizing temperatures in the range of 800° to 1500°F.

5. Same as Item 4.

6. The Licensee does not agree with the requirement to perform intergranular corrosion tests for each welding procedure. Control of the PWR reactor coolant within the limits of the Technical Requirements Manual (TRM) ensures a benign environment (i.e., low oxygen, low fluoride, and chroride content), and the use of welding filler materials with a minimum ferrite number (FN) of 7.5 FN (approximately 7.5% ferrite content) mitigates the concerns for intergranular stress corrosion cracking (IGSCC).

The position on Regulatory Guide 1.44 is discussed in part in Subsection 5.2.3.4 (Fabrication and Processing of Austenitic Stainless Steel) and in Subsection 6.1.1.1.

REGULATORY GUIDE 1.45

REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION SYSTEMS

The Licensee complies with Revision 0 of this guide with the following clarifications/exceptions keyed to paragraph numbers in the regulatory position.

1. Identified leak sources are piped to either the RC drain tank or a miscellaneous drain tank to be utilized for this purpose only. The temperature of selected drain lines is monitored to identify leaks. Tank inventories are monitored. Temperature monitoring is more sensitive to small leaks than flow rate monitoring specified in the Position.

2. Unidentified leak sources are monitored to as accurate an equivalent flow rate as is practicable.

   Containment floor drain and reactor cavity flow monitors for unidentified leakage may not always be accurate to within 1 gpm; however, sump level monitoring and pump run time monitoring are used as alternate means of monitoring floor drain flow.

3. The following leak detection systems are provided:

Identified Sources

a. RC drain tank level indication and temperature indication of selected inlet lines, or

b. pressurizer relief tank level indication and temperature indication of selected inlet lines.

Unidentified Sources

a. containment floor drain and reactor cavity flow, and sump level,

b. containment atmosphere particulate radioactivity monitoring, and

c. containment gaseous radioactivity monitoring.

d. VCT level and net charging/letdown flow indication provide quantitative indication of RCS leakage.

e. Containment dry-bulb temperatures and pressure provide indirect indication of leakage to the containment.

4. Intersystem leakage between primary and secondary plant is monitored via air ejector off-gas radiation monitors. Also, pressurizer and makeup tank levels are monitored to yield total reactor coolant leakage. Refer to subsection 5.2.5.5 for additional leakage detection methods.

5. Leak detector sensitivity is as low as practicable. Refer to Subsection 5.2.5.2 for additional information on containment radiation monitor sensitivity.

6. The containment floor drain and reactor cavity flow indications are designed to remain functional after an SSE and are powered by non-ESF buses. The sump level indications are seismically qualified and powered by ESF buses. Although the containment radiation monitors are not seismically qualified, they are seismically mounted and can be powered from safety-related buses, via the non-safety to safety related cross-tie breakers, if necessary.

7. Conversions to common leakage equivalent are supplied to operators wherever possible. Conversions to a common leakage equivalent are not possible in all cases. In these cases, the system is intended primarily for localization or identification of a leak with no quantitative implications.

Further information on reactor coolant pressure boundary leak detection can be found in Subsection 5.2.5.

REGULATORY GUIDE 1.46

PROTECTION AGAINST PIPE WHIP INSIDE CONTAINMENT

This regulatory guide was withdrawn on March 1, 1985 because the July 1981 revision of Standard Review Plan 3.6.2 provides more current information. The NRC review is included in the Safety Evaluation Reports, NUREG-0876, dated February 1982, for Byron and NUREG-1002, dated November 1983, for Braidwood.

REGULATORY GUIDE 1.47

BYPASSED AND INOPERABLE STATUS INDICATION FOR NUCLEAR POWER PLANT SAFETY SYSTEMS

The Licensee complies with the regulatory position in Revision 0 of this guide as discussed in Section 7.5 and Subsections 7.1.2.10 and 8.1.9.

REGULATORY GUIDE 1.48

DESIGN LIMITS AND LOADING COMBINATIONS FOR SEISMIC CATEGORY I FLUID SYSTEM COMPONENTS

Regulatory Guide 1.48 was withdrawn on March 1, 1985 because the July 1981 revision of the Standard Review Plan 3.9.3 provides more current information. The NRC review is included in the Safety Evaluation Reports, NUREG-0876, dated February 1982, for Byron and NUREG-1002, dated November 1983, for Braidwood.

REGULATORY GUIDE 1.49

POWER LEVELS OF NUCLEAR POWER PLANTS

The Byron/Braidwood design meets the recommendations of Regulatory Guide 1.49, Revision 1, since the initial power level is less than 3800 MWt and analyses and evaluation are made at assumed core power levels less than the levels in the guide.

REGULATORY GUIDE 1.50

CONTROL OF PREHEAT TEMPERATURE FOR WELDING OF LOW-ALLOY STEEL

The Licensee complies with Revision 0 of this guide with the following comments and exceptions keyed to paragraph numbers in the regulatory position.

1.a. Licensee requires preheat temperatures referenced in applicable codes but does not require a maximum interpass temperature. To date, it has not been found necessary to specify a maximum interpass temperature.

1.b. Welding procedures are qualified in the preheat temperature range. It is not possible to consistently maintain preheat at the minimum temperature during welding procedure qualification.

2. The Licensee does not agree with this position. It is impossible to maintain preheat temperature during fabrication of spool pieces when four or five welds have been made prior to a complete post-weld heat treatment of the spool piece. The only way this could be accomplished would be to have intermittent post-weld heat treatment which in the case of the higher alloy steel, such as 2-1/4 chrome, may be detrimental.

3. Preheat temperature limit is monitored, but not interpass temperature.

Westinghouse considers that this guide applies to ASME Section III Class 1 Components.

The NSSS vendors' practice for Class 1 components is in agreement with the requirements of Regulatory Guide 1.50 except for regulatory positions 1 (b) and 2. For class 2 and 3 components, Westinghouse does not apply any of Regulatory Guide 1.50 recommendations.

In the case of regulatory position 1 (b), the welding procedures are qualified within the preheat temperature ranges required by Section IX of the ASME Code. Westinghouse qualification procedures.

In the case of regulatory Position 2, the vendor's position described in WCAP-8577, "The Application of Preheat Temperature After Welding of Pressure Vessel Steels," has been found acceptable by the NRC. This WCAP establishes the guidelines which permit the component manufacturer to either maintain the preheat until a post-weld heat treatment or allow the preheat to drop to ambient temperature.

In the case of reactor vessel main structural welds, the practice of maintaining preheat until the intermediate or final post-weld heat treatment has been followed by the vendor. In either case, the welds have shown high integrity.

The NSSS vendor meets Regulatory Position 4 in that, for their components, the examination procedures required by Section III and the inservice inspection requirements of Section XI are met. (For further information, see Paragraph 5.3.1.4.)

REGULATORY GUIDE 1.51

INSERVICE INSPECTION OF ASME CODE CLASS 2 AND 3 NUCLEAR POWER PLANT COMPONENTS

The guidance in Regulatory Guide 1.51 has been incorporated in the 1974 edition of Section XI of the ASME Boiler and Pressure Vessel Code. Later editions of the code were used for preservice and inservice inspections of ASME Code Class 2 and 3 components. This regulatory guide was withdrawn on July 5, 1975 because it was no longer needed.

REGULATORY GUIDE 1.52

DESIGN, TESTING AND MAINTENANCE CRITERIA FOR ENGINEERED-SAFETY-FEATURE ATMOSPHERE CLEANUP SYSTEM AIR FILTRATION AND ADSORPTION UNITS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS

The Licensee complies with Revision 2 of the regulatory position with the following comments and exceptions keyed to paragraph numbers in Section C of the position:

1.e (Deleted) (Note 1)

2.a Entrained water droplets are not considered credible due to significant quantities of ductwork with elbows. Water droplets, if present, will impinge on ducts and drop out of vertical duct risers as the air enters exhaust plenums. However, the auxiliary building exhaust system does contain prefilters which can serve as demisters.

2.d (Deleted) (Note 1)

2.f The auxiliary building nonaccessible area exhaust filter system consists of three built-up filter trains (one standby) with a rated capacity of 62,000 cfm each. The system design flow is 62,730 cfm, which is within 110% of installed filter rated capacity. For maintenance purposes each train is divided into three banks with each bank sized for seven filters wide and three high. Each train has a total of 62 HEPA filter elements.

The auxiliary building accessible area exhaust filter system has a rated capacity of 186,000 cfm. The system design flow is 125,490 cfm. This system consists of four built-up filter trains (one standby), and each train is divided into three banks. Each bank is sized for seven filters wide and three filters high. Each train has a total of 62 HEPA filter elements.

2.g ALL ESF filter systems have local control panel airflow indication. In addition, the flow rate of each of the stacks is recorded at the local control panel. The airflow rate through the control room emergency makeup air filter units and the auxiliary building and fuel handling building exhaust charcoal booster fans is continuously sensed.

The differential pressure across all of the ESF filter unit fans is indicated on local control panels. High and low differential pressure, and fan trip annunciation is provided on the main control panel. The setpoints for low and high differential pressure alarms will be such that flows at ±10% of design flow will be alarmed at the main control panel in the form of low or high differential pressure.

The differential pressures across all HEPA filters are indicated on local control panels. High differential pressure across all HEPA filters is annunciated on the main control panel and on local control panels. Upset conditions will therefore be identified to the control room operator allowing him to take appropriate action.

The auxiliary building exhaust flow rate and fuel handling building exhaust flow rate is continuously sensed by the exhaust stack airflow measuring equipment and recorded in a local control panel and by the plant computer.

2.j Filter trains are not designed to be removable from the building as an intact unit. The size of the train precludes shipment off-site and there are no facilities for onsite disposal of the intact unit. The filter elements are removable and can be disposed of through the solid radwaste system.

2.1 Filter Housings

All of the auxiliary building and fuel handling building exhaust system filter housings are designed in accordance with ANSI N509-76. The housings are at negative pressure with respect to their surroundings and are located in auxiliary building general area which is a low airborne radiation environment. Any in-leakage from the general area will not adversely affect Appendix I releases. Hence, the housings were not leak tested to the ANSI N509 requirements. However, filter mounting leak tests were performed in accordance with ANSI N510-80.

The control room emergency makeup air system filter housings are designed in accordance with ANSI N509-76. The filter housings are at negative pressure with respect to their surroundings, and are located within the control room boundary which is a habitable environment the same as the control room. Any in-leakage will be from the control room environment and, therefore, will not adversely affect the quality of that environment; hence, the housings were not leak tested to ANSI N509 requirements. However, filter mounting frame tests were performed in accordance with ANSI N510-80. Some field welds may have been painted prior to performing the mounting frame leakage test.

Ductwork

All auxiliary building and fuel handling building exhaust system ductwork upstream of the filter units is under negative pressure with respect to its surroundings and is located in the same areas of the buildings served by the exhaust systems. Any in-leakage will be filtered prior to discharge to the atmosphere, hence, this ductwork was not be tested to ANSI N509 requirements.

However, prior to the final system turnover, the nonaccessible exhaust system of the auxiliary building HVAC system and the fuel handling building exhaust system was operated and the ductwork was visually and audibly checked for leaks. Leaks were sealed.

All control room emergency makeup air system ductwork is located within the control room boundary which is a habitable environment. Any ductwork leakage will not adversely affect the habitability of the environment, hence, this ductwork was not tested to ANSI N509 requirements. However, prior to final system turnover, the control room emergency makeup air system was operated and visual and audible leaks in ductwork were sealed.

The design airflow quantities for each system were verified during testing, adjusting and balancing of the systems. Deviations of more than ± 10% of the design flow quantities were evaluated and any disposition was documented. For the auxiliary building exhaust system, minimum airflow quantities are limited to system requirements while maximum airflow quantities are limited to 110% of the installed filter capacity. A calibrated orifice was used in lieu of a gas flow totalizer for determining leakage.

2.m (Deleted) (Note 1)

3.b (Deleted) (Note 1)

3.d Replacement HEPA filters shall meet the requirements of ANSI N509-1989 in lieu of N509-1976.

3.e (Deleted) (Note 1)

3.h (Deleted) (Note 1)

3.i Replacement activated carbon shall meet the requirements of Table 5.1 of ANSI N509-1980 in lieu of ANSI N509-1976, which was replaced by ANSI N509-1980.

3.n Ductwork is designed, constructed, and tested in accordance with intent of Section 5.10 of ANSI N509-1976. The longitudinal seams, however, are either seal welded or mechanical lock type (Pittsburgh lock with sealant). Silicone sealant is used as a permanent sealant for HVAC ductwork.

Fan peak pressure tests were not performed. For systems that have isolation devices, the fans are provided with high differential pressure trips or high/low flow trips.

3.p Bubble tight isolation and shutoff dampers are provided only for the control room intakes. Two parallel blade isolation dampers in series are provided in the VA system nonaccessible system charcoal filter bypass.

4.b The space provided between components is 3 feet from the front (or rear) of the components to the nearest obstacle (filter frame or other filter component). This allows 3 feet of access between components.

4.c (Deleted) (Note 1)

4.d The periodicity of Emergency Makeup Unit testing is set in accordance with the Surveillance Frequency Control Program.

5.b Airflow distribution tests were performed to ensure that the airflow through any individual filter element does not exceed 120% of the element's rated capacity.

The VA system accessible, nonaccessible and fuel handling building filters air capacity tests were performed to verify that maximum flow is not greater than 110% of filter rated capacity. Airflow capacity tests were performed to ensure that the plenum flow rate does not exceed 110% of filter rated capacity. The minimum flow rate is based on the exhaust flow rate needed to meet both ALARA and equipment qualification requirements.

Filtration unit airflow capacity tests were performed at the system design pressure range corresponding to clean and dirty filter losses. Tests were performed at 1.25 times dirty filter conditions to verify system stability only. Filter pressure losses for airflow capacity tests were simulated without filters in place.

5.c Silicone sealant was used as a permanent sealant for HVAC ductwork.

5.d The acceptance criteria for bypass leakage through the control room HVAC make-up charcoal adsorber is less than 1.0%.

6. Replacement activated carbon shall meet the requirements of Table 5.1 of ANSI N509-1980 in lieu of ANSI N509-1976, which was replaced by ANSI N509-1980.

6.b The acceptance criteria for control room HVAC make-up charcoal adsorber lab analysis methyl iodide penetration is less than 2.0%.

The control room HVAC recirculation charcoal is tested to 48 fpm.

Further discussions on this subject can be found in Section 6.5 and Subsections 9.4.1.2 and 12.3.1.7.

Note 1: Exception to this section is no longer required because the Regulatory Guide has been revised to eliminate the criteria to which exception was originally taken.

REGULATORY GUIDE 1.52

DESIGN, TESTING AND MAINTENANCE CRITERIA FOR ENGINEERED-SAFETY-FEATURE ATMOSPHERE CLEANUP SYSTEM AIR FILTRATION AND ADSORPTION UNITS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS

The Licensee complies with Revision 2 of the regulatory position with the following comments and exceptions keyed to paragraph numbers in Section C of the position:

1.e (Deleted) (Note 1)

2.a Entrained water droplets are not considered credible due to significant quantities of ductwork with elbows. Water droplets, if present, will impinge on ducts and drop out of vertical duct risers as the air enters exhaust plenums. However, the auxiliary building exhaust system does contain prefilters which can serve as demisters.

2.d (Deleted) (Note 1)

2.f The auxiliary building nonaccessible area exhaust filter system consists of three built-up filter trains (one standby) with a rated capacity of 63,000 cfm each. The system design flow is 62,730 cfm. For maintenance purposes each train is divided into three banks with each bank sized for seven filters wide and three high. Each train has a total of 63 HEPA filter elements.

The auxiliary building accessible area exhaust filter system has a rated capacity of 189,000 cfm. The system design exhaust flow is 125,490 cfm. This system consists of four built-up filter trains (one standby), and each train is divided into three banks. Each bank is sized for seven filters wide and three filters high. Each train has a total of 63 HEPA filter elements.

2.g ALL ESF filter systems have local control panel airflow indication. In addition, the flow rate of each of the stacks is recorded at the local control panel. The airflow rate through the control room emergency makeup air filter units and the auxiliary building and fuel handling building exhaust charcoal booster fans is continuously sensed.

The differential pressure across all of the ESF filter unit fans is indicated on local control panels. High and low differential pressure, and fan trip annunciation is provided on the main control panel. The setpoints for low and high differential pressure alarms will be such that flows at ±10% of design flow will be alarmed at the main control panel in the form of low or high differential pressure.

The differential pressures across all HEPA filters are indicated on local control panels. High differential pressure across all HEPA filters is annunciated on the main control panel and on local control panels. Upset conditions will therefore be identified to the control room operator allowing him to take appropriate action.

The auxiliary building exhaust flow rate and fuel handling building exhaust flow rate is continuously sensed by the exhaust stack airflow measuring equipment and recorded in a local control panel and by the plant computer.

2.j Filter trains are not designed to be removable from the building as an intact unit. The size of the train precludes shipment off-site and there are no facilities for onsite disposal of the intact unit. The filter elements are removable and can be disposed of through the solid radwaste system.

2.1 Filter Housings

All of the auxiliary building and fuel handling building exhaust system filter housings are designed in accordance with ANSI N509-76. The housings are at negative pressure with respect to their surroundings and are located in auxiliary building general area which is a low airborne radiation environment. Any in-leakage from the general area will not adversely affect Appendix I releases. Hence, the housings were not leak tested to the ANSI N509 requirements. However, filter mounting leak tests were performed in accordance with ANSI N510-80.

The control room emergency makeup air system filter housings are designed in accordance with ANSI N509-76. The filter housings are at negative pressure with respect to their surroundings, and are located within the control room boundary which is a habitable environment the same as the control room. Any in-leakage will be from the control room environment and, therefore, will not adversely affect the quality of that environment; hence, the housings were not leak tested to ANSI N509 requirements. However, filter mounting frame tests were performed in accordance with ANSI N510-80.

Some field welds may have been painted prior to performing the filter mounting frame leak tests.

Ductwork

All auxiliary building and fuel handling building exhaust system ductwork upstream of the filter units is under negative pressure with respect to its surroundings and is located in the same areas of the buildings served by the exhaust systems. Any in-leakage will be filtered prior to discharge to the atmosphere, hence, this ductwork was not tested to ANSI N509 requirements.

However, prior to the final system turnover, the nonaccessible exhaust system of the auxiliary building HVAC system and the fuel handling building exhaust system will be operated and the ductwork was visually and audibly checked for leaks. Leaks were sealed.

All control room emergency makeup air system ductwork is located within the control room boundary which is a habitable environment. Any ductwork leakage will not adversely affect the habitability of the environment, hence, this ductwork was not tested to ANSI N509 requirements. However, prior to final system turnover, the control room emergency makeup air system was operated and visual and audible leaks in the ductwork were sealed.

The design airflow quantities for each system was verified during testing, adjusting and balancing of the systems. Deviations of more than ± 10% of the design flow quantities were evaluated and any disposition was documented. For the auxiliary building exhaust system, minimum airflow quantities are limited to system requirements while maximum airflow quantities are limited to 110% of the installed filter capacity. A calibrated orifice was used in lieu of a gas flow totalizer for determining leakage.

2.m (Deleted) (Note 1)

3.b Initial duct heater performance testing per ANSI N510-80 was not performed for the Control Room Make-up Air Filter Unit heating coils. Installed heater capacity was verified using control room HVAC system pre-operational test data. The review of the test data demonstrated that the heaters can perform at their design capacity.

3.d Replacement HEPA filters shall meet the requirements of ANSI N509-1989 in lieu of N509-1976.

3.e (Deleted) (Note 1)

3.h (Deleted) (Note 1)

3.i Replacement activated carbon shall meet the requirements of Table 5.1 of ANSI N509-1980 in lieu of ANSI N509-1976, which was replaced by ANSI N509-1980.

3.n Ductwork is designed, constructed, and tested in accordance with the intent of Section 5.10 of ANSI N509-1976. The longitudinal seams, however, are either seal welded or mechanical lock type (Pittsburgh lock with sealant). Silicone sealant is used as a permanent sealant for HVAC ductwork.

Fan peak pressure tests were not performed. For systems that have isolation devices, the fans are provided with high differential pressure trips or high/low flow trips.

3.p Bubble tight isolation and shutoff dampers are provided only for the control room intake. Two parallel blade isolation dampers in series are provided in the VA system nonaccessible system charcoal filter bypass.

4.b The space provided between components is 3 feet from the front (or rear) of the components to the nearest obstacle (filter frame or other filter component). This allows 3 feet of access between components.

4.c (Deleted) (Note 1)

4.d The periodicity of Emergency Makeup Unit testing is set in accordance with the Surveillance Frequency Control Program.

5.b Airflow distribution tests were performed to ensure that the airflow through any individual filter element does not exceed 120% of the element's rated capacity.

The VA system accessible, nonaccessible and fuel handling building filters air capacity tests were performed to verify that maximum flow is not greater than 110% of the filter rated capacity. The minimum flow rate is based on the exhaust flow rate needed to meet both ALARA and equipment qualification requirements.

Filtration unit airflow capacity tests were performed at the system design pressure range corresponding to clean and dirty filter issues. The midpoint filter drop test was not performed. Tests were performed at 1.25 times dirty filter conditions to verify system stability only. Filter pressures losses for airflow capacity tests were simulated without filters in place.

5.c Silicone sealant or other temporary patching material was not used in the ESF filter housings. Silicone sealant is used, however, as a permanent sealant for HVAC ductwork.

A sampling rate of less than 1 cfm was employed for testing filter systems larger than 1000 cfm.

The acceptance criteria for the VA Nonaccessible (NAC) and Fuel Handling Building (FHB) Systems is per the VFTP.

The amount of leakage bypassing the NAC HEPA filters on the standby train when determining NAC Total System Bypass leakage will be the amount measured when the train is on-line.

5.d The acceptance criteria for bypass leakage through the control room HVAC make-up charcoal adsorber is less than 1.0%.

The acceptance criteria for the NAC and FHB Systems is per the VFTP.

The amount of leakage bypassing the NAC charcoal adsorbers on the standby train when determining NAC Total System Bypass leakage will be the amount measured when the train is on-line.

6.a(2) All carbon furnished prior to 1985 as part of the original specification for atmospheric clean-up filtration units was tested to the requirements of Table 5-1 of ANSI N509-1976. All replacement carbon or original carbon furnished in 1985 or later will be tested to the requirements of Table 5-1 of ANSI N509-1980 with the exception that the laboratory test for methyl iodine penetration at 30° C, 95% relative humidity is less than 1%.

6.a(3) Laboratory tests will be performed per the Ventilation Filter Testing Program.

6.b The acceptance criteria for control room HVAC make-up charcoal adsorber lab analysis methyl iodide penetration is less than 2.0%.

The control room HVAC recirculation charcoal is tested to 48 fpm.

The acceptance criteria for the VA NAC and FHB System charcoal penetration of the standby train (service condition when VA System is in the emergency mode) is encompassed by the testing performed at the rated flow with the train on-line. The acceptance criteria for the NAC and FHB Systems is per the VFTP.

Further discussions on this subject can be found in Section 6.5 and Subsections 9.4.1.2 and 12.3.1.7.

Note 1: Exception to this section is no longer required because the Regulatory Guide has been revised to eliminate the criteria to which exception was originally taken.

REGULATORY GUIDE 1.53

APPLICATION OF THE SINGLE-FAILURE CRITERION TO NUCLEAR POWER PLANT PROTECTION SYSTEMS

The Licensee complies with Revision 0 of the guidelines for application of the single failure criteria to nuclear power plant protection systems as discussed in Subsections 7.1.2.11 and 8.1.10.

REGULATORY GUIDE 1.54

QUALITY ASSURANCE REQUIREMENTS FOR PROTECTIVE COATINGS APPLIED TO WATER-COOLED NUCLEAR POWER PLANTS

The Licensee complies with Revision 0 of the regulatory guide position with the exception of some undocumented or unqualified coatings. (See Subsection 6.1.2 for further information.)

REGULATORY GUIDE 1.55

CONCRETE PLACEMENT IN CATEGORY I STRUCTURES

The plant design conforms to the regulatory position in Revision 0 with the following exceptions:

1. ACI 301-72 specifies that the frequency for cylinder testing shall be two cylinders per 100 yards of concrete, tested at 28 days with a minimum of one set per day for each class of concrete.

   The Licensee's position is to use six cylinders per 150 yards of concrete, tested at 7, 28, and 91 days, with a minimum of one set per day for each class of concrete. This exceeds the requirements of both ACI 318-77 and ACI 349-76.

   The compliance with the requirements of this regulatory guide is discussed in detail in Section B.1 of Appendix B.

2. ACI 301-72, Subsection 8.5.3, requires that grouting be applied on the vertical surfaces of construction joints. This requirement has been removed from ACI 381-80.

The requirements remain in effect even though the regulatory guide was withdrawn on July 8, 1981. The regulatory position is now covered by one or more national standards.

REGULATORY GUIDE 1.56

MAINTENANCE OF WATER PURITY IN BOILING WATER REACTORS

This regulatory guide is pertinent to BWRs only.

REGULATORY GUIDE 1.57

DESIGN LIMITS AND LOADING COMBINATIONS FOR METAL PRIMARY REACTOR CONTAINMENT SYSTEM COMPONENTS

The licensee complies with the regulatory position in Revision 0 of this guide with the following clarifications.

Piping penetration assemblies are designed by the following guidelines:

a. The portion of the primary containment penetration assembly which is part of the containment boundary, i.e., the penetration sleeve in its entire length (including the sleeve projection that forms an extension to the wall), is designed in accordance with Subsection NE, Section III of the ASME Code, augmented by the applicable provisions of Regulatory Guide 1.57.

b. The portion of the primary containment penetration assembly which consists of the head fitting (flued head) and part of the process pipeline, is designed in accordance with Subsection NB of the Code so as to satisfy stress requirements for design conditions (NB-3112, NB-3221), normal and upset conditions (NB-3113.1, NB-3112.2, NB-3222, NB-3223), emergency conditions (NB-3224), faulted conditions (NB-3113.4, F-1324.1, F-1324.6, Table F-1322), and testing conditions (NB-3226, NB-6222, NB-6322).

   Part b of the Licensee's position, which refers to the NB classification of the flued head and process pipe, is supported by NA-2134 of the Code and note 3 of Regulatory Guide 1.57.

   Part b discussion is applicable to Type 1 penetration type as described in UFSAR sections 3.8.2.1.3.1.1 and 3.8.2.1.3.2. All other head fittings are MC component and designed to applicable NE requirements of ASME Section III as discussed in Part a and UFSAR Section 3.8.2.1.3.2.

REGULATORY GUIDE 1.58

QUALIFICATION OF NUCLEAR POWER PLANT INSPECTION, EXAMINATION, AND TESTING PERSONNEL

The requirements of Regulatory Guide 1.58 have been incorporated in Regulatory Guide 1.28, Revision 3. Regulatory Guide 1.58 was withdrawn on June 17, 1991.

The Licensee complies with the intent of Regulatory Guide 1.28 Revision 3, but applies it to ASME/ANSI NQA-1-1994.

REGULATORY GUIDE 1.59

DESIGN BASIS FLOODS FOR NUCLEAR POWER PLANTS

The plant design conforms to the regulatory positions in Revision 2 as described in Section 2.4.

REGULATORY GUIDE 1.60

DESIGN RESPONSE SPECTRA FOR SEISMIC DESIGN OF NUCLEAR POWER PLANTS

The plant design conforms to the regulatory positions in Revision 1 as described in Subsection 2.5.2.

REGULATORY GUIDE 1.61

DAMPING VALUES FOR SEISMIC DESIGN OF NUCLEAR POWER PLANTS

The plant design conforms to the regulatory positions in Revision 0 as described in Subsections 3.7.1.3 and 3.7.3.14, and 3.7.3.15 with the single exception of the large piping systems (diameter greater than 12 inches) SSE condition value of 3% critical. A conservative value of 4% critical for the Westinghouse reactor coolant loop configuration has been justified by testing and has been approved by the NRC staff. The test results are given in WCAP-7921-AR, "Damping Values of Nuclear Power Plant Components." The use of higher damping values, when justified by documented test data, have been provided for in Regulatory Position C2.

REGULATORY GUIDE 1.62

MANUAL INITIATION OF PROTECTIVE ACTIONS

The Licensee complies with Revision 0 of this regulatory guide. Refer to Section 7.3 and Subsections 8.1.11, 7.1.2.1.2, 7.1.2.12.1, and 7.2.1.1.2 for further information.

REGULATORY GUIDE 1.63

ELECTRIC PENETRATION ASSEMBLIES IN CONTAINMENT STRUCTURES FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS

The plant design specification requires the penetration vendors to meet the requirements of Regulatory Guide 1.63, Revision 0, which was in effect at the time the construction permit application was docketed. Regulatory Guide 1.63 supplements IEEE 317-1972, the IEEE Standard for "Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations," and contains no specific testing recommendations. Regulatory Position C-4 does however add the quality assurance requirements of ANSI N45.2-1971 and ANSI N45.2.4-1972 to Section 8 (Required Data and Quality Control and Quality Assurance Procedures) of IEEE 317-1972.

The design, construction, installation, and testing of the electrical penetration assemblies will be in accordance with the quality assurance requirements of Regulatory Position C-4. See Subsections 3.11.2, 7.1.2.13, and 8.1.12 for further information.

REGULATORY GUIDE 1.64

QUALITY ASSURANCE REQUIREMENTS FOR THE DESIGN OF NUCLEAR POWER PLANTS

The requirements in Regulatory Guide 1.64 have been incorporated in Regulatory Guide 1.28, Revision 3. Regulatory Guide 1.64 was withdrawn on June 17, 1991.

The Licensee complies with the intent of Regulatory Guide 1.28 Revision 3, but applies it to ANSI/ASME NQA-1-1994.

REGULATORY GUIDE 1.65

MATERIALS AND INSPECTIONS FOR REACTOR VESSEL CLOSURE STUDS

The Licensee complies with Revision 0 of Regulatory Guide 1.65, except for material and tensile strength guidelines and supplemental inservice inspection (ISI) examinations.

Westinghouse has specified both 45 ft-lb and 25 mils lateral expansion for control of fracture toughness determined by Charpy-V testing, required by the ASME Boiler and Pressure Vessel Code, Section III, Summer 1973 Addenda and 10 CFR 50, Appendix G (July 1973, Paragraph IV.A.4). These toughness requirements assure optimization of the stud bolt material tempering operation with the accompanying reduction of the tensile strength level when compared with previous ASME Boiler and Pressure Vessel Code requirements.

The specification of both impact and maximum tensile strength as stated in the guide results in unnecessary hardship in procurement of material without any additional improvement in quality. The closure stud bolting material is procured to a minimum yield strength of 130,000 psi and a minimum tensile strength of 145,000 psi. This strength level is compatible with the fracture toughness requirements of 10 CFR 50, Appendix G (July 1973, Paragraph 1.C), although higher strength level bolting materials are permitted by the code. Stress corrosion has not been observed in reactor vessel closure stud bolting manufactured from material of this strength level. Accelerated stress corrosion test data do exist for materials of 170,000 psi minimum yield strength exposed to marine water environments stressed to 75% of the yield strength (given in Reference 2 of the guide). These data are not considered applicable to Westinghouse reactor vessel closure stud bolting because of the specified yield strength differences and a less severe environment; this has been demonstrated by years of satisfactory service experience.

The ASME Boiler and Pressure Vessel Code requirement for toughness for reactor vessel bolting has precluded the guide's additional recommendation for tensile strength limitation, since to obtain the required toughness levels, the tensile strength levels are reduced. Prior to 1972, the Code required to 35 ft-lb toughness level which provided maximum tensile strength levels ranging from approximately 155 to 178 ksi (Westinghouse review of limited data - 25 heats). After publication of the Summer 1973 Addenda to the Code and 10 CFR 50, Appendix G, wherein the toughness requirements were modified to 45 ft-lb with 24 mils lateral expansion, all bolt material data reviewed on Westinghouse plants showed tensile strengths of less than 170 ksi.

Additional protection against the possibility of incurring corrosion effects is assured by:

1. Decrease in level of tensile strength comparable with the requirement of fracture toughness as described above.

2. Design of the reactor vessel studs, nuts, and washers, allowing them to be completely removed during each refueling permitting visual and/or nondestructive inspection in parallel with refueling operations to assess protection against corrosion, as part of the inservice inspection program.

3. Design of the reactor vessel studs, nuts, and washers, providing protection against corrosion by allowing them to be completely removed during each refueling and placed in storage racks on the containment operating deck, as required by Westinghouse refueling procedures. The stud holes in the reactor flange are sealed with special plugs before removing the reactor closure. Thus, the bolting materials and stud holes are never exposed to the borated refueling cavity water.

4. Use of a manganese phosphate surface treatment.

Use of Code Case 1605 does not constitute an issue between the NRC and Westinghouse inasmuch as (a) no questions have been raised on this point in vendor's approved standard reference document discussions of this guide and (b) use of this code case has been approved by the NRC via the guideline of Regulatory Guide 1.85.

Inservice inspection examinations of reactor pressure vessel bolting (closure head studs, nuts, washers, etc.) are performed in accordance with the methods specified in the station Ten Year Inservice Inspection (ISI) Plan which is mandated through 10 CFR 50.55a. Volumetric examination of bolting is performed in accordance with the procedures and qualifications of approved versions of ASME Section XI, Division 1, Appendix VIII, “Performance Demonstration for Ultrasonic Examination Systems” with Supplement 8 “Bolts and Studs” mandated through 10 CFR 50.55a.

Further discussion of reactor coolant pressure boundary materials, inspection, and testing is in Subsections 5.2.3 and 5.2.4.

REGULATORY GUIDE 1.66

NONDESTRUCTIVE EXAMINATION OF TUBULAR PRODUCTS

The regulatory positions of the guide have been incorporated in Section III of the ASME code for tubular products intended for use in safety-related systems. This code is used to perform nondestructive examinations. Regulatory Guide 1.66 was withdrawn on September 28, 1977 because it was no longer needed.

REGULATORY GUIDE 1.67

INSTALLATION OF OVERPRESSURE PROTECTION DEVICES

The recommendations of this guide are included in the ASME Boiler and Pressure Vessel Code, which is incorporated by reference in 10 CFR 50.55a. The regulatory guide was withdrawn on April 15, 1983 because it was no longer needed.

REGULATORY GUIDE 1.68

INITIAL TEST PROGRAMS FOR WATER-COOLED REACTOR POWER PLANTS

The Licensee complies with Revision 2 of this regulatory guide, as described in Chapter 14.0, with the following exceptions:

Appendix A.2.b states "To the extent practical, testing should demonstrate control rod scram times at both hot zero power and cold temperature conditions, with flow and no-flow conditions in the reactor coolant system as required to bound conditions under which scram might be required."

A full spectrum of rod drop measurements was made for Byron Unit 1 at cold no-flow, hot no-flow, cold full-flow, and hot full-flow conditions. Byron Unit 2 and Braidwood Units 1 and 2 are identical to Byron Unit 1 with respect to the rod control system. Because of this, no additional design information would be obtained by repeating the entire spectrum of rod drop measurements that was originally done for Byron Unit 1. Consequently, the Licensee intends to perform only the hot full-flow measurements for the remaining three units.

Appendix A.4.c states "Following initial criticality, licensee should conduct pseudo-rod-ejection test to verify calculational models and accident analysis assumptions."

The results of the Byron Unit 1 pseudo-rod-ejection test confirmed the design predictions made for the event within the accuracy of the testing procedure. Verification of core design parameters for the remaining three units can be achieved through control rod worth measurements and flux mapping at zero power and during the power ascension phase. Consequently, the Licensee does not intend to perform the pseudo-rod-ejection test for Byron Unit 2 and Braidwood Units 1 and 2 because no additional information will be provided with regard to core performance because of the design similarity.

Appendix A.4.t states "Performance of natural circulation tests of the reactor coolant system to confirm that the design heat removal capability exists or to verify that flow (without pumps) or temperature data are compatible to prototype designs for which equivalent tests have been successfully completed (PWR).

"As described in the Byron SER Section 5.4.3 the Licensee has referenced the natural circulation testing which was performed at Diablo Canyon. The NRC staff and

Brookhaven National Laboratory have reviewed the Diablo Canyon test results and found them acceptable. A preliminary assessment of differences between Byron and Diablo Canyon that may affect boron mixing under natural circulation has been provided and indicates that the Diablo Canyon test results and supporting analysis satisfy the necessary requirements for Byron. Byron/Braidwood Stations and Diablo Canyon Unit 1 have subsequently been compared in detail to ascertain any differences between the plants that could potentially affect natural circulation flow and attendant boron mixing. Because of the similarity between the plants, the Licensee concluded that the natural circulation capabilities would be similar, and therefore, the results of prototypical natural circulation cooldown tests conducted at Diablo Canyon would be representative of the capability at Byron/Braidwood. The plant comparison is further discussed in subsection 5.4.7. Based on the review of the similarities between Byron/Braidwood and Diablo Canyon, the NRC has concluded that Byron and Braidwood have demonstrated that the Diablo Canyon natural circulation tests are applicable to Byron/Braidwood and that they comply with the requirements of BTP RSB 5-1 (Reference 1). Additionally, simulator training for Byron reactor operators includes natural circulation procedures training.

Appendix A.5.a states "Determine that power reactivity coefficients (PWR) or power vs. flow characteristics (BWR) are in accordance with design values (25%, 50%, 75%, 100%)."

Per recommendations of Westinghouse, Byron NSSS vendor, the Licensee intends to perform this testing at the 30%, 50%, 75%, and 90% power ascension testing plateaus. These testing plateaus correspond to those previously listed in Table 14.2-82 for the Power Reactivity Coefficient Measurement Startup Test.

Appendix A.5.e states "Pseudo-rod-ejection test to validate the rod ejection accident analysis."

The results of the Byron Unit 1 pseudo-rod-ejection test confirmed the design predictions made for the event within the accuracy of the testing procedure. Verification of core design parameters for the remaining three units can be achieved through control rod worth measurements and flux mapping at zero power and during the power ascension phase. Consequently, the Licensee does not intend to perform the pseudo-rod-ejection test for Byron Unit 2 and Braidwood Units 1 and 2 because no additional information will be provided with regard to core performance because of the design similarity.

Appendix A.5.h states "Check rod scram times from data recorded during scrams that occur during the startup test phase to determine that the scram times remain within allowable limits."

During power ascension testing, the Licensee does not intend to formally instrument the rod position indication system for reactor trip review purposes because this would require removal of the rod position indication from service. This would be a violation of Technical Specifications. For this reason the Licensee believes that this requirement applies to BWRs only.

Appendix A.5.i states "Demonstrate capability and/or sensitivity, as appropriate for the facility design of incore and excore neutron flux instrumentation, to detect a control rod misalignment equal to or less than the technical specification limits (50%, 100%) (PWR)."

An evaluation of instrumentation response to a misaligned control rod will be performed during the Byron Unit 1 flux asymmetry startup test conducted at 50% power. The Licensee does not intend to perform this test on Byron Unit 2 or Braidwood Units 1 and 2 because no additional design confirmation will be obtained due to identical core configurations and system designs. Also, in accordance with a recommendation from the NSSS vendor, the Licensee does not intend to perform this testing at 100% power. Although the technical specifications provide relief from the requirements of certain technical specifications when performing physics tests below 85% power, creating a control rod misalignment at 100% power does not fall within this special test exclusion. As a result, technical specification limits regarding rod insertion and/or peaking factors may be exceeded.

Appendix A.5.j states "Verify that plant performance is as expected for rod runback and partial scram."

The Licensee asserts that these particular events are applicable to BWRs only, and therefore will not be performed.

Appendix A.5.ff states, "Demonstrate or verify that important ventilation and air-conditioning systems, including those for the primary containment and steamline tunnel, continue to maintain their service areas within the design limits (50%, 100%)."

The Licensee asserts that the reference to steamline tunnel ventilation applies only to boiling water reactors, in which case this system would be safety-related up to the turbine stop valves. For Byron/Braidwood, this steamline tunnel ventilation system is non-safety-related and, therefore, no such demonstration will be performed.

Appendix A.5.gg states "If appropriate for the facility design, conduct tests to determine operability of equipment provided for anticipated transient without scram (ATWS), if not previously done (25%)."

The initial test program did not include this testing because, at that time, the facility design did not include an ATWS mitigation system (AMS). Subsequently, the facility design was changed to include an AMS, as discussed in subsection 7.7.1.21. AMS was tested during implemention of the design change.

Appendix A.5.kk states "Demonstrate that the dynamic response of the plant is in accordance with design for the loss of or bypassing of the feedwater heater(s) from a credible single failure or operator error that would result in the most severe case of feedwater temperature reduction (50%, 90%)."

As described in Subsection 15.1.1.2, the transient resulting from the simultaneous isolation and bypass of a high-pressure feedwater heater is the most severe heater isolation/bypass event. This accident yields a reduction in feedwater temperature of 18°F and a new specific enthalpy of 399.83 Btu/lb_m. This event is less severe than the transient resulting from a 10% step load increase provided the steam generator inlet temperature does not decrease by more than 55°F and the specific feedwater enthalpy is greater than 342.06 Btu/lb_m. Since the heater isolation/bypass accident satisfies these criteria, the simultaneous isolation and bypass of a heater event is bounded by a 10% step increase in load. The 10% load increase is tested as described in Table 14.2-88. Therefore, a feedwater heater bypass test as described by Appendix A.5.kk will not be performed.

Appendix A.5.mm states "Demonstrate that the dynamic response of the plant is in accordance with design for the case of automatic closure of all main steam line isolation valves. For PWRs, justification for conducting the test at a lower power level, while still demonstrating proper plant response to this transient, may be submitted for NRC staff review (100%)."

The Licensee does not intend to perform this test because the closure of all main steam isolation valves will result in a turbine trip per Byron Subsection 15.2.4. The generator/turbine trip test will be performed at 100% power and is a more severe transient. The Licensee will further perform a turbine trip at about 25% power on Byron Unit 1 with the turbine bypass valves closed and disabled. This will further verify plant dynamic response. The combination of these two trip tests will verify the transient response of the plant and the capability of the secondary side decay heat removal systems to cope with these transients. The disabling of the turbine bypass system will restrict that capability to the steam generator PORVs and auxiliary feedwater systems (i.e., the safety-related systems) and will demonstrate their dynamic capability. The performance of the MSIV test would be redundant and would provide no additional information regarding plant response or capability. In effect, performance of this test would only result in unnecessary cycling of this equipment.

A test program has been established to ensure that all structures, systems, and components will satisfactorily perform their safety-related functions. This test program provides additional assurance that the plant has been properly designed and constructed and is ready to operate in a manner that will not endanger the health and safety of the public, that the procedures for operating the plant safely have been evaluated and have been demonstrated, and that the plant and procedures are fully prepared to operate the facility in a safe manner.

The test program includes simulation of equipment failures and control system malfunctions that could reasonably be expected to occur during the plant lifetime. The test program also includes testing for interactions such as the performance of interlock circuits in the reactor protection system. It also determines that proper permissive and prohibit functions are performed.

Care is taken to ensure that redundant channels of the equipment are tested independently.

The initial startup testing, conducted after the fuel loading and before commercial operation, will confirm the design bases and demonstrate, where practical, that the plant is capable of withstanding the anticipated transient and postulated accidents.

A detailed description of the test program is provided in Chapter 14.0.

References

1. NRC Letter, "Byron Station Units 1 and 2 and Braidwood Station Units 1 and 2, Natural Circulation Cooldown," dated November 4, 1988.

REGULATORY GUIDE 1.68.1

PREOPERATIONAL AND INITIAL STARTUP TESTING OF FEEDWATER AND CONDENSATE SYSTEMS FOR BOILING WATER REACTOR POWER PLANTS

This guide is pertinent to BWRs only.

REGULATORY GUIDE 1.68.2

INITIAL STARTUP TEST PROGRAM TO DEMONSTRATE REMOTE SHUTDOWN CAPABILITY FOR WATER-COOLED NUCLEAR POWER PLANTS

The Licensee complies with the position in Revision 1 of this regulatory guide. Refer to Table 14.2-86 for additional information.

REGULATORY GUIDE 1.68.3

PREOPERATIONAL TESTING OF INSTRUMENT AND CONTROL AIR SYSTEMS

This guide, which replaces Regulatory Guide 1.80, is not applicable to Byron and Braidwood. Refer to the discussion of Regulatory Guide 1.80.

REGULATORY GUIDE 1.69

CONCRETE RADIATION SHIELDS FOR NUCLEAR POWER PLANTS

The Licensee complies with the regulatory position in Revision 0 of this guide. Concrete radiation shielding is discussed in Subsection 12.3.2.

REGULATORY GUIDE 1.70

STANDARD FORMAT AND CONTENT OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS

The FSAR is written in accordance with the content and format set forth by Regulatory Guide 1.70, Revision 2, which was the current revision. The content and format have been maintained in the UFSAR and its updates.

However, as part of the ongoing effort to improve the quality of the UFSAR, the guidelines provided in Nuclear Energy Institute (NEI) 98-03, "Guidelines for Updating Final Safety Analysis Reports," Revision 1, June 1999, as endorsed by NRC Regulatory Guide 1.181, "Content of the Updated Final Safety Analysis Report in Accordance with 10CFR50.71(e)," Revision 0, September 1999, are used to further improve the content of the UFSAR. While the UFSAR will continue to follow the general organizational recommendations, i.e., format, specified in this Regulatory Guide, the reorganization options described in NEI 98-03 will be used to simplify information contained in the UFSAR to improve its focus, clarity, and maintainability.

REGULATORY GUIDE 1.71

WELDER QUALIFICATION FOR AREAS OF LIMITED ACCESSIBILITY

The Licensee maintains that limited-accessibility qualification or requalification described in Revision 0 of Regulatory Guide 1.71 exceeds ASME Section III and IX requirements and is an unduly restrictive and unnecessary requirement. Acceptability of welds will be determined by required examinations. Multiple production welds of similar components in the shop will be subjected to close control and supervision achieving the same purpose as the guide. See Subsections 5.3.1.4 and 5.2.3 for further information.

REGULATORY GUIDE 1.72

SPRAY POND PLASTIC PIPING

This regulatory guide does not apply to this application, since the Byron/Braidwood design does not utilize spray ponds.

REGULATORY GUIDE 1.73

QUALIFICATION TESTS OF ELECTRIC VALVE OPERATORS INSTALLED INSIDE THE CONTAINMENT OF NUCLEAR POWER PLANTS

This regulatory guide indicates the NRC acceptance (with certain qualifications) of the requirements of IEEE 382-1972, "IEEE Trial-Use Guide for Type Test of Class I Electric Valve Operators for Nuclear Power Generating Stations."

The Licensee complies with the objectives set forth in Revision 0 of this regulatory guide as indicated in Subsections 6.2.4.2 and 8.1.13.

REGULATORY GUIDE 1.74

QUALITY ASSURANCE TERMS AND DEFINITIONS

The requirements in Regulatory Guide 1.74 have been incorporated in Regulatory Guide 1.28, Revision 3. Regulatory Guide 1.74 was withdrawn on September 1, 1989.

The Licensee complies with the intent of Regulatory Guide 1.28 Revision 3, but applies it to ANSI/ASME NQA-1-1994.

REGULATORY GUIDE 1.75

PHYSICAL INDEPENDENCE OF ELECTRIC SYSTEMS

NSSS Scope

The commitment to comply with the intent of the requirements in Revision 2 of this guide is presented in Subsections 7.1.2.2.1 and 7.1.2.2.2.

Non-NSSS Scope

Physical independence of redundant electric systems is discussed in Subsections 8.1.14 and 8.3.1.4, respectively.

The Licensee complies with the requirements of this guide with the exceptions and/or clarifications to the regulatory positions identified and justified below:

Regulatory Position C1

Section 3, "Isolation Device" should be supplemented as follows: "(Interrupting devices actuated only by fault current are not considered to be isolation devices within the context of this document.)"

Licensee's Position

Interrupting devices actuated only by fault current may be used as isolation devices provided that the requirements of IEEE 384-1977, including the change (fuses are acceptable isolation devices in dc power circuits) as proposed by the IEEE-PES-NPEC-SC6.5 Working Group in their September 7, 1988 letter, are met.

Justification of Licensee's Position

There is no technical justification for precluding the use of Class 1E circuit breakers or Class 1E fuses actuated only by fault or overload current as circuit interrupting or isolation devices. (For further discussion of this subject, see S&L letter to the NRC dated December 21, 1978 and NRC's response dated March 28, 1979.)

Byron/Braidwood Design

Although the Licensee believes that a single circuit breaker or fuse (actuated by fault current only) provides adequate isolation, the Byron/Braidwood design will incorporate the following additional features to further ensure isolation and thus satisfy NRC concerns.

The Licensee (where practical) will provide two interrupting devices (in series) actuated only by fault current. These two interrupting devices will be: 1) Class lE qualified and 2) coordinated with their upstream interrupting device; breakers will be periodically tested to verify coordination. Periodic testing of fuses, in dc power circuits, to verify coordination is not required, provided that each fuse is tested (for example, resistance measurement) to verify overcurrent protection as designed. In lieu of periodic testing, a documented Periodic Inspection and Maintenance procedure shall be implemented which will ensure:

- that the proper size and type of fuse is installed,
- that the fuse shows no physical sign of deterioration, and
- that the fuse connections are tight and clean.

Any remaining non-Class lE loads (not utilizing two interrupting devices) will be tripped from the Class lE buses with a Safety Injection coincident with loss of offsite power signal. The cables which supply non-Class lE loads from redundant Class lE buses are routed through separate raceways.

Regulatory Position C2

Section 3, "Raceway". Interlocked armor enclosing cable should not be construed as a "raceway."

Licensee's Position

Although not a "raceway" in the same sense as a conduit or cable tray, recognition of and design credit for the additional protection provided by the metallic jacket of interlocked armored cable should be included in the regulatory guide. Use of armored cable, in lieu of the separation distances stated in the regulatory guide, should be permitted when justified by specific testing and/or analysis, as providing the required degree of protection for Class lE circuits against specific credible hazards.

Justification of Licensee's Position

There is no technical justification for precluding the use of armored cable, in lieu of separation distances, to provide adequate isolation between Class 1E and non-Class 1E circuits and between redundant Class 1E circuits, when shown to be adequate by specific testing and/or analysis.

Regulatory Position C6

Analyses performed in accordance with Sections 4.5(3), 4.6.2, and 5.1.1.2 should be submitted as part of the Safety Analysis Report and should identify those circuits installed in accordance with these sections.

Licensee's Position

The referenced analysis, when performed to justify deviation from specific requirements of standard IEEE 384-1974, shall be prepared on a case-by-case basis, shall be documented and be on permanent file, available for NRC review, but will not be an integral part of the safety analysis report.

Justification of Licensee's Position

The Licensee's position is consistent with that taken for other plant design records; e.g., routine design calculations, design document revisions, etc.

Regulatory Position C7

Non-Class lE instrumentation and control circuits should not be exempted from the provisions of Section 4.6.2.

Licensee's Position

Low energy non-Class lE instrumentation and control circuits are not required to be physically separated or electrically isolated from "associated" circuits provided (a) the non-Class lE circuits are not routed with "associated" circuits of a redundant division, and (b) they are analyzed to demonstrate that Class lE circuits are not degraded below an acceptable level. As part of the analysis, consideration shall be given to the potential energy and identification of the circuits involved.

Justification of Licensee's Position

The Licensee's position is consistent with the industry consensus position regarding required separation between non-Class lE circuits and "associated" circuits, taken in the 1977 and 1981 revisions to IEEE-384, Section 4.6.1(4).

Regulatory Position C8

Section 5.1.1.1 should not be construed to imply that adequate separation of redundant circuits can be achieved within a confined space such as a cable tunnel that is effectively unventilated.

Licensee's Position

Adequate separation of redundant Class lE circuits can be achieved in areas of the plant that are effectively unventilated.

Justification of Licensee's Position

There is no technical justification for precluding the routing of redundant Class lE circuits through areas of the plant that may be "effectively unventilated" provided that adequate physical separation is provided between redundant circuits and appropriate thermal derating factors for such circuits have been incorporated into the plant design.

Regulatory Position C9

Section 5.1.1.3 should be supplemented as follows:
"(4) Cable splices, in raceways, should be prohibited."

Licensee's Position

Cable splices, either within raceways or at the interface of raceways and equipment, etc., are permitted provided they are intended by the plant design as indicated on the design documents.

Justification of Licensee's Position

There is no technical justification for precluding the use of cable splices within raceways or at their interfaces with equipment, etc., provided that they are an integral part of the plant design as indicated on the design documents.

Regulatory Position C10

Section 5.1.2, the phrase "at a sufficient number of points" should be understood to mean at intervals not to exceed 5 feet throughout the entire cable length. Also, the preferred method of marking cable is color coding.

Licensee's Position

Cable installed in exposed Class 1E and "associated" circuit raceways shall be identified in a manner of sufficient durability and at sufficient intervals to facilitate initial verification that the installation is in conformance with the separation criteria. Methods of providing the justification, other than color coding of the cable jacket, are acceptable.

Justification of Licensee's Position

There is no technical justification for requiring the intervals of identification of such cables to not exceed every 5 feet throughout the entire cable length. Cable system designs employing less frequent identification intervals and that provide for verification that the installation is in conformance with the separation criteria are acceptable.

Use of cable jacket color coding alone, as a method of providing cable identification, may not be as effective as alternative methods. Other methods, e.g., using unique cable identification number with a segregation code, could be a more positive method of facilitating verification that the cable installation is in accordance with the design separation requirements.

Regulatory Position C11

Section 5.1.2 should be supplemented as follows: "The method of identification used should be simple and should preclude the need to consult any reference material to distinguish between Class 1E and non-Class 1E circuits, between non-Class 1E circuits 'associated' with different redundant Class 1E systems, and between redundant Class 1E systems."

Licensee's Position

The method of initial installation verification need not preclude consultation of reference documents.

Justification of Licensee's Position

There is no technical justification for precluding use of reference documents during the installation verification process (e.g., use of design documents and installation records). A system based upon the use of such reference documents could be a most effective check that a cable is installed in accordance with the design documents and in a raceway of compatible segregation assignment.

Regulatory Position C12

Pending issuance of other acceptable criteria, those portions of Section 5.1.3 (exclusive of the Note following the second paragraph) that permit the routing of cables through the cable spreading area(s) and, by implication, the control room, should not be construed as acceptable. Also, Section 5.1.3 should be supplemented as follows: "Where feasible, redundant cable spreading areas should be utilized."

Licensee's Position

Power cables installed in dedicated solidly enclosed metallic raceways in air (e.g., rigid steel conduit or solid cable trays with solid flush covers), may be routed through those areas designated as "cable spreading areas," where justified by analysis or other suitable means.

Justification of Licensee's Position

There is no technical justification to preclude the routing of power cables through cable spreading areas when they are installed in such a manner to present no hazard to other cabling, generally of a lower energy level, within the area.

Regulatory Position C14

Section 5.2.1 should be supplemented as follows: "And should have independent air supplies."

Licensee's Position

Redundant standby generating units shall be placed in separate safety class structures and shall be provided with separate ventilation and combustion air systems.

Justification of Licensee's Position

The Licensee's position is an interpretation of what is believed to be the intent of the Regulatory Position.

Regulatory Position C15

Where ventilation is required, the separate safety class structures required by Section 5.3.1 should be served by independent ventilation systems.

Licensee's Position

Redundant batteries shall be housed in separate safety class structures, i.e., separate from one another, not necessarily separate from everything else within its own safety division. For example, a battery may be placed in the same safety class structure as the switchgear for that division.

Justification of Licensee's Position

The Licensee's position is an interpretation of what is believed to be the intent of the Regulatory Position.

Regulatory Position C17

Regulatory Guide Position on Section 4.6.1 "Separation from Class lE Circuits," of IEEE Std 384 (1974)

By not modifying Section 4.6.1 of IEEE Std 384 (1974) in a regulatory position, the regulatory guide has endorsed it as stated in the IEEE standard.

Licensee's Position

There is no justification for precluding the use of technically acceptable analysis to justify, on a case-by-case basis, exceptions to the generally stated criteria for separation of non-Class lE circuits, from Class lE circuits. When such analysis demonstrates that the following requirements are met, the non-Class lE circuits involved need not be classified as "associated" circuits.

For specific cases, where cable termination or routing arrangements (e.g., cables leaving cable trays in free air entering equipment or passing through conduit sleeves in walls) limit the available separation distances between non-Class lE and Class lE cables, to less than the minimum separation applicable to redundant cables in raceways, such lesser separations are permitted provided that a documented analysis is performed to demonstrate that:

a. the non-Class 1E circuits are not routed with Class 1E circuits of a redundant division or circuits "associated" with a redundant division, and

b. the Class 1E circuits involved are not degraded below an acceptable level.

The analysis will include consideration of the potential energies of the circuits involved; the physical and electrical isolation (i.e., barriers) provided for the circuits by the cable insulation, the cable shielding, and the cable jacketing systems; the degree of environmental qualification and fire retardant characteristics of the cables; and the potential for hazards in the specific area involved.

Justification of Licensee's Position

The Licensee's position is consistent with the industry consensus technical position stated in the 1977 revision to IEEE-384, Section 4.6.1(3).

REGULATORY GUIDE 1.76

DESIGN BASIS TORNADO FOR NUCLEAR POWER PLANTS

The plant design conforms to the regulatory position in Revision 0 as described in Section 3.3.

REGULATORY GUIDE 1.77

ASSUMPTIONS USED FOR EVALUATING A CONTROL ROD EJECTION ACCIDENT FOR PRESSURIZED WATER REACTORS

(Refer to Subsections 15.4.7 and 15.4.8.3 for details of this analysis.)

Westinghouse methods and criteria are documented in WCAP-7588 Revision 1A which has been reviewed and accepted by the NRC.

The results of their analyses show compliance with the Regulatory Position given in Sections C1 and C3 of Regulatory Guide 1.77, Revision 0. However, they take exception to Regulatory Guide Position C2 which implies that the rod ejection accident should be considered as an emergency condition. Westinghouse considers this a faulted condition as stated in ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants." Faulted condition stress limits will be applied for this accident.

This guide, although used in the original plant design, has been superseded by Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors".

REGULATORY GUIDE 1.78

ASSUMPTIONS FOR EVALUATING THE HABITABILITY OF A NUCLEAR POWER PLANT CONTROL ROOM DURING A POSTULATED HAZARDOUS CHEMICAL RELEASE

The Licensee complies with Revision 0 of this regulatory guide. Refer to Section 2.2 and Subsection 6.4.1 for further information.

As allowed by paragraph C.4 of Regulatory Guide (RG) 1.78, Revision 0, "The toxicity limits should be taken from appropriate authoritative sources." NUREG/CR-6624 is considered an appropriate authoritative resource and, therefore, the toxicity limits contained within may be used for periodic toxic gas surveys in place of those contained in RG 1.78, Revision 0.

REGULATORY GUIDE 1.79

PREOPERATIONAL TESTING OF EMERGENCY CORE COOLING SYSTEMS FOR PRESSURIZED WATER REACTORS

The Licensee complies with the requirements in Revision 1 of this guide. The containment spray system is tested up to the containment isolation valve while taking suction from the refueling water storage tank. In testing the RHR system under the recirculation conditions, the containment sumps are filled with cold water up to elevation 376 feet 9 inches. The adequacy of NPSH available under postaccident recirculation conditions to the RHR pumps is corrected for water elevation, temperature, and run out flow through the containment spray pumps.

Procedures to verify operability of the ECCS pumps establish proper flow-requirements during flow tests conducted at cold conditions. The capability of the pumps to deliver required flows under accident conditions has been verified by analysis to preclude any unnecessary thermal shock damage at hot operating conditions. Flow capabilities were verified using data obtained from unplanned and planned safety injection actuation performed during the testing program. Check valve operability has been evaluated to guidelines and criteria established in Table 14.2-34. See Chapter 14.0 for further discussion of preoperational testing.

REGULATORY GUIDE 1.80

PREOPERATIONAL TESTING OF INSTRUMENT AIR SYSTEMS

The air systems in the Byron/Braidwood design are designated Safety Category II, Quality Group D. As non-safety-related equipment, the air system does not come under the provision of Regulatory Guide 1.80, Revision 0. This regulatory guide was renumbered and reissued as Regulatory Guide 1.68.3. Regulatory Guide 1.80 was withdrawn on April 20, 1982.

REGULATORY GUIDE 1.81

SHARED EMERGENCY AND SHUTDOWN ELECTRIC SYSTEMS FOR MULTIUNIT NUCLEAR POWER PLANTS

The Byron/Braidwood design complies with the requirements in Revision 1 of this regulatory guide (which indicates the acceptable methods of compliance with General Design Criterion 5). The independence of each unit's onsite electrical systems are further discussed in Subsection 8.1.15.

The Licensee believes that the intent of Regulatory Guide 1.81 (Position C.1) was to disallow "normal" sharing of d-c systems, not to disallow the temporary connection of one d-c bus to a source in the other unit during periods of testing and/or maintenance. Provisions for administratively controlled manually actuated, interconnections between the nonredundant Class 1E d-c bus for each unit improves the overall reliability and availability of the d-c systems by allowing a means for manually providing power to a d-c bus at a time when it would otherwise have to be out of service (e.g., to perform a battery discharge cell, etc.) That this was the intent is evident from the "Discussion," in Regulatory Guide 1.81, Part B, second paragraph, first sentence, which reads as follows:

"Sharing of onsite power systems at multi-unit power plant sites generally results in a reduction in the number and capacity of the onsite power sources to levels below those required for the same number of units located at separate sites."

The "interconnection" provided in the Byron/Braidwood design does not result in a reduction in either the number or the capacity of the d-c power sources. . ., i.e., the number and capacity of the d-c power sources for each of the two units are exactly the same as they would be if the units were located at separate sites.

The interconnection between each Unit's Class 1E 125-Vdc systems, via the cross-tie, is limited by procedural and administrative controls. These controls ensure that combinations of maintenance and test operations will not preclude the systems capabilities to supply power to the ESF d-c loads. The criteria specifying the allowable combinations of maintenance and test operations will be governed by the plant technical specifications. Coordination between unit operations required during maintenance and testing will be governed by administrative controls.

REGULATORY GUIDE 1.82

WATER SOURCES FOR LONG-TERM RECIRCULATION COOLING FOLLOWING A LOSS-OF-COOLANT ACCIDENT

The suction screens to the containment recirculation sumps have been replaced as part of the activities to respond to NRC Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors". The replacement filters have an opening size of 1/12 inch and have been designed to comply with NRC regulations that have been published in support of NRC Generic Letter 2004-02.

The Byron and Braidwood plant design conforms to the requirements of regulatory Guide 1.82 Revision 3. Compliance with the regulatory position of this guide is discussed below:

1. PRESSURIZED WATER REACTORS

1.1 Features Needed To Minimize the Potential for Loss of NPSH

The ECC sumps, which are the source of water for such functions as ECC and containment heat removal following a LOCA, should contain an appropriate combination of the following features and capabilities to ensure the availability of the ECC sumps for long-term cooling. The adequacy of the combinations of the features and capabilities should be evaluated using the criteria and assumptions in Regulatory Position 1.3.

1.1.1 ECC Sumps, Debris Interceptors, and Debris Screens

1.1.1.1

The Byron and Braidwood Stations containment recirculation sumps include two separate sumps, fully redundant, each servicing one train of the Emergency Core Cooling System (ECCS). The replacement screens for each sump are sized to the full design basis debris load.

1.1.1.2

Two redundant sump pits are physically separated from each other and are protected from high energy piping by the solid steel cover of the trash rack structure.

The trash rack is a structure made of tube steel and angle steel supports, with side vertical stainless steel grating and a solid checkered plate cover. The top of the trash rack structure is approximately four (4) ft above the containment floor elevation of 377 ft. Design loads for the trash rack have been determined to address hydrostatic pressure, drag force, and debris impact. Dynamic effects due to design basis high energy line breaks need not be considered in the structural qualification of the trash rack, and dynamic effects from pipe breaks considered in the vicinity of the trash rack structure need not be considered. An additional load due to lead shielding has also been considered to address lead shielding activities during refueling outages. (Reference 1)

The checkered plate located on top of the trash rack structure is not permanently attached to the supporting framing members. The physical configuration of these plates provides adequate lateral restrain for the applied seismic self-weight excitation force and horizontal drag force. Furthermore, the vertical OBE and SSE accelerations are both lower than 1.0. Therefore, the dead weight of the checkered plate is adequate to resist uplift during a seismic event. Operation of the ECCS in the Recirculation Mode is assumed only in the design basis analysis for a LOCA.

1.1.1.3

The containment recirculation sumps are located below the containment floor elevation of 377 ft. This is the lowest floor elevation in containment, exclusive of the reactor vessel cavity. This maximizes the depth of the pool water above the sump screens.

A trash rack structure, vertical grating on the perimeter and checkered plate on top, protects the openings for both sumps. The sump suction pipe is located inside the sump pits. Each sump pit has a concrete slab ceiling with three openings that allow water to enter the pit. The concrete slab provides further protection for the screens and sump suction pipe. A debris interceptor plate is installed at each grating sector to prevent larger debris from accessing the sump pits.

1.1.1.4

The screens for each recirculation sump have been tested under design basis loading conditions. The trash rack will prevent debris sliding along the floor from reaching the screens.

1.1.1.5

The containment recirculation sumps are located between the Primary Shield wall and the Secondary Shield wall; this area is normally referred to as Inside Missile Barrier (IMB). Due to this location, the water leaking directly from the RCS piping, or connected piping located in the same area, does not flow through any restricted path. The same argument applies for Containment Spray wash down water that falls into the IMB area through the Steam Generators enclosures, the Reactor Coolant Pumps enclosures and the Pressurizer enclosure.

Containment Spray wash down volume between the outside of the Secondary Shield wall and the inside of the outer containment wall (referred to as Outside Missile Barrier, OMB), will not be obstructed in the path to the emergency recirculation sump. Two access openings exist at containment elevation 377 ft to control access to IMB from OMB. The size of each opening is a nominal 3 ft wide x 7 ft high; a screen door (1-1/2 inch diamond mesh) is installed at each location. This door is locked during normal plant operation to prevent personnel from entering the high radiation field in the IMB area.

Blockage of the openings to IMB is not likely because high-energy jets from the RCS will not cause direct debris in the OMB area. The Feedwater Line Break and the Main Steam Line Break accidents are the design basis accidents that may result in high-energy jets in the OMB area of containment. The ECCS response sequence to these accidents is not likely to progress further than the injection phase with the RWST as the water source. In fact, Emergency Recirculation from the containment emergency recirculation sumps is not modeled for these accidents.

Minimum containment flood level analyses have accounted for containment spray volumes that are trapped in the Refueling Cavity (Reference 8). The resulting flood levels do not degrade the post-LOCA recirculation function.

Blockage of containment drainage paths into the ECCS recirculation sumps is not a concern because there are no pipes that drain into the sumps.

1.1.1.6

The trash rack is a structure made of tube steel and angle steel supports, with side vertical stainless steel grating and a solid checkered plate cover. The top of the trash rack structure is approximately four (4) ft above the containment floor elevation of 377 ft. Design loads for the trash rack have been determined to address hydrostatic pressure, drag force, and debris impact. Dynamic effects due to design basis high energy line breaks need not be considered in the structural qualification of the trash rack, and dynamic effects from pipe breaks considered in the vicinity of the trash rack structure need not be considered. An additional load due to lead shielding has also been considered to address lead shielding activities during refueling outages. Design hydrodynamic loads includes the effects of sloshing. (Reference 1)

1.1.1.7

The recirculation sump screens at Braidwood and Byron Station are installed inside sump pits below the lowest containment floor elevation of 377 ft. The screens have been shown to be fully submerged following an accident. Air entrapment at the trash rack structure does not have any effect on the hydraulics of the sump screens.

1.1.1.8

All components of the trash rack structure are acceptable for the design basis loads consisting of hydrostatic pressure, drag force, debris impact loads, temporary lead shielding (For outage periods), rigging loads, pipe support loads, hydrodynamic loads, and seismic loading under OBE and SSE seismic events.

1.1.1.9

The sump screens and grating are made of stainless steel while the trash rack support elements are made of carbon steel. These elements are located in a dry environment and will not degrade during normal operation or for the duration of the accident mission time for the ECCS sump.

1.1.1.10

The debris interceptor structure (trash Rack) includes access openings to facilitate inspection of the trash rack, the sump screens and the sump outlets. The sump screens have been verified not to be susceptible to vortex formation during vendor testing.

1.1.1.11

The replacement sump screens have been verified by analysis and testing to result in a head loss that is adequate to maintain adequate NPSH for the RHR and CS pumps during the post-LOCA operation of the ECCS (Reference 2).

1.1.1.12

The possibility of debris-clogging flow restrictions downstream of the sump screen has been assessed to ensure adequate long term recirculation cooling, containment cooling, and containment pressure control capabilities. The downstream blockage evaluation has been performed in accordance with industry (Westinghouse Owners Group) and regulatory guidance, considering the size of the openings in the sump debris screen (1/12 inch).

(References 4, 5, and 6)

The downstream blockage evaluation has concluded that the Safety Injection throttle valves (_SI8810A-D, _SI8816A-D, _SI8822A-D) are susceptible to blockage.

The Safety Injection throttle valves have been modified to minimize the blockage potential after an accident. The modification installed a Copes-Vulcan HUSH II trim. This trim consists of an assembly of nested concentric cylinders each having a series of radially drilled holes. The orifice areas are developed by arranging the cylinders, one within the other, in an offset manner so that a series of restriction (pinch areas) and expansion areas occur in series. The total pressure is thus reduced in stages. The internal design dimensions of the trim assemblies have been set to minimize blockage due to debris that passes through the containment recirculation sump screens (hole size of 1/12 inch or 0.083 inch).

The final design of the valves' internal components has been developed based on results from extensive testing at Wyle Laboratories with debris-laden fluid. The quantity of debris was based on post-LOCA debris calculations specific to Byron and Braidwood Stations.

Testing results show a limited reduction in the valves Cv when coating debris is first added to the water mix. Testing also showed an increasing long term valve Cv after the full debris mix is added. For conservatism, the initial recorded Cv reduction is used in the hydraulic analysis (Reference 3) to calculate flow rates to the RCS for ECCS Cold Leg and Hot leg Recirculation under debris-laden conditions. The impact of the resulting flow rates on the accident analysis has been evaluated by Westinghouse and has been found to be acceptable (Reference 12).

1.1.1.13
The pump suction inlets at the sump screens have been verified by testing not to be susceptible to air ingestion or any other adverse hydraulic effect (Reference 7).

1.1.1.14
The Byron and Braidwood Station design does not include any drain pipe that would bypass the sump screens.

1.1.1.15
The design of the Byron and Braidwood sump screens, combined with the limited amount of fibrous materials inside the containment building, prevents the formation of a thin bed on the screen surfaces. This fact has been demonstrated during testing.

1.1.2 Minimizing Debris

Byron and Braidwood Stations have minimized the debris that could reach the sumps by replacing fibrous insulation within its specific Zone of Influence (ZOI) on the Steam Generators and connected piping for Byron Unit 1 and Braidwood Unit 1. Trash rack gratings has been installed at elevation 377 ft of containment to minimize the quantity of debris that enters the sumps.

1.1.2.1

Braidwood and Byron have implemented a “Containment Loose Debris Inspection” procedure (References 15 and 16). The procedures outline the steps necessary to verify the containment is free of loose debris. It is applicable for all accessible areas just prior to establishing Containment Integrity and for affected areas at the completion of any Containment entry when Containment Integrity is already set. These procedures for containment closeout necessitates that a containment walkdown be performed for housekeeping deficiencies. The procedure incorporates a list of all unresolved housekeeping and equipment discrepancies and requires that resolution be included in the plant restart documentation. The procedure also provides guidance on general cleanliness and debris inspection guidelines.

1.1.2.2

Fibrous Insulation (Thermal Wrap Trademark) on the Steam Generators and connected piping has been replaced within its Zone of Influence (ZOI) for Byron Unit 1 and Braidwood Unit 1.

Procedures already exist to clean up the work area following maintenance activities inside containment. This action prevents the generation of additional latent debris.

1.1.2.3

Bare metal surfaces inside containment that are not stainless steel are coated. Galvanized steel surfaces (i.e., scaffolds) have been accounted for in the chemical effects evaluation for sizing the replacement screens.

1.1.3 Instrumentation

Byron and Braidwood Stations do not rely on operator actions to mitigate the consequences of the accumulation of debris on the ECC sump screens.

1.1.4 Active Sump Screen System

Byron and Braidwood Station do not employ an active screen.

1.1.5 Inservice Inspection

To ensure the operability and structural integrity of the trash racks and screens, access openings can be used to inspect the ECC sump structures and the sump suction pipes. The sump screens, suction pipe, and trash racks are inspected each refueling outage in accordance with the requirements of the Technical Specifications.

1.2 Evaluation of Alternative Water Sources

Byron and Braidwood Stations do not rely on operator actions to prevent the accumulation of debris on the sump screens.

1.3 Evaluation of Long-Term Recirculation Capability

The nuclear industry has developed a methodology (Document #NEI 04-07) to perform the required evaluations with the cooperation of the Nuclear Energy Institute (NEI) and the NRC has issued a Safety Evaluation Report on the NEI document in December of 2005.

The most important elements of the NEI methodology are break selection, debris generation, and debris transport. Substantial conservatism is built into the evaluation process. For example, although application of the Leak-before-Break methodology has been approved for Byron and Braidwood on the main Reactor Coolant Loop piping, the debris generation analysis assumes the largest pipes in the Reactor Coolant System break. Byron and Braidwood have followed the NEI methodology. The results of the evaluation concluded that the existing screens must be replaced. New screens have been designed and manufactured by Control Components Inc. (CCI).

In addition, chemical reactions between the post-LOCA water and materials (Aluminum, copper, concrete) located in the containment building may result in chemical precipitants forming when the post-LOCA water temperature decreases in the latter part of the accident. This is called “chemical effects”; the design of the replacement screens is required to incorporate the impact on head loss due to “chemical effects”.

Head loss testing on a scaled version of the replacement screens has been performed using scaled quantities of debris (coating, fiber insulation, reflective metal insulation, glass). The replacement screen assemblies have been verified to have a head loss that maintains adequate margin for the Net Positive Suction Head (NPSH) for the RHR and CS pumps when they take suction from the containment recirculation sumps. The head loss testing performed by CCI also includes chemical effects.

1.3.1 Net Positive Suction Head of ECCS and Containment Heat Removal Pumps

1.3.1.1

The NPSH analysis (Reference 2) for temperatures above 200 °F assumes that the vapor pressure of the recirculation sump liquid is equal to the containment pressure. This ensures that credit is not taken for increase in the containment pressure due to the accident.

The NPSH analysis for temperatures below 200 °F credits the minimum containment air pressure that was present inside containment before the accident. No credit is taken in the NPSH analysis for increase in containment pressure due to the accident.

Below is the final NPSH margin (NPSH Available minus Screen Head Loss minus NPSH Required) considering air evolution effects:

Tsump (°F)	Screen Head Loss (ft)	RHR Pumps NPSH Margin (ft)	Braidwood CS Pumps NPSH Margin(ft)	Byron CS Pumps NPSH Margin (ft)
258.1	4.13	2.77	0.67	1.07
250.6	4.13	2.27	0.57	0.97
242.5	4.13	2.17	0.57	0.97
230.0	4.13	2.67	0.67	1.07
210.0	4.20	2.7	0.60	1.0
203.9	4.20	2.7	0.60	1.0
200.0	4.20	2.8	0.60	1.0
195.0	4.24	6.66	4.46	4.96
175.0	4.55	13.35	10.95	11.45
155.0	4.77	17.63	15.23	15.63
135.0	5.34	19.46	16.96	17.46
120.001	6.11	18.89	16.59	16.99
119.999	6.11	17.19	15.19	15.59
95.0	7.20	17.0	14.9	15.3
73.4	8.27	15.73	13.73	14.13

Additional limitations exist on the allowable head loss for the recirculation sump screens due to the screen structural limit and the pump air void fraction limits. The resulting screen head loss margins are as follows:

Tsump (°F)	Screen Head Loss (ft)	Margin Over Screen Loss: RHR Pumps (ft)	Margin Over Screen Loss: Braidwood CS Pumps (ft)	Margin Over Screen Loss: Byron CS Pumps (ft)
258.1	4.13	2.8	0.7	1.1
250.6	4.13	2.3	0.6	0.9
242.5	4.13	2.2	0.6	0.9
230.0	4.13	2.6	0.7	1.0
210.0	4.20	2.7	0.6	1.0
203.9	4.20	2.7	0.6	1.0
200.0	4.20	2.8	0.6	1.0
195.0	4.24	6.7	4.5	4.9
175.0	4.55	11.0	11.0	11.0
155.0	4.77	10.8	10.8	10.8
135.0	5.34	10.2	10.2	10.2
120.001	6.11	9.4	9.4	9.4
119.999	6.11	9.4	9.4	9.4
95.0	7.20	8.3	8.3	8.3
73.4	8.27	7.3	7.3	7.3

Notes:

- The screen vendor reports the screen head loss calculation at specific temperatures up to a maximum of 212 °F. For temperatures greater than 212 °F, the screen head loss is conservatively assumed to be 4.13 feet, the value for expected head loss at 212 °F. No credit is taken for lower head loss due to lower viscosity at higher temperatures.
- The suction piping friction losses that have been used in the calculation of available NPSH have been determined based on maximum flow rates that are larger than calculated. This assumption accounts for approximately 1 ft of margin.
- The suction piping friction losses that have been used in the calculation of available NPSH have been determined based on the maximum calculated increase in viscosity due to chemical effects. The maximum viscosity increase does not apply to temperatures above 140 °F.
-

1.3.1.2

The NPSH analysis does not credit increase in the containment pressure due to the accident.

1.3.1.3

The NPSH analysis for the CS and RH pumps does not take credit for operation of the pumps while cavitating. For the SI and CV pumps, the NPSH analysis from the RWST is more limiting.

1.3.1.4

The post-accident temperature history for the containment and recirculation sump water has been taken from the existing analyses for containment integrity (Reference 2).

1.3.1.5

The hot channel correction factor specified in ANSI/HI 1.1-1.5-1994 is not used in determining the margin between the available and required NPSH for ECCS and containment heat removal system pumps.

1.3.1.6

The calculation of available NPSH uses the containment flood level from the minimum containment flood level calculation (Reference 8). This input minimizes the height of water above the pump suction (i.e., the level of water on the containment floor). The calculated minimum flood height inside containment does not consider quantities of water that do not contribute to the sump pool (e.g., atmospheric steam, inactive water volume, pooled on the containment floor and in the refueling canal, spray droplets and other falling water, etc.).

1.3.1.7

The calculation of pipe and fitting resistance has been done using the Flowseries software (Reference 3). The nominal screen resistance without blockage by debris has been measured during model testing at the screen vendor facility.

1.3.1.8

Head loss through the sump screens has been determined based on testing of a model screen assembly, under design basis debris loading conditions (scaled accordingly).

1.3.1.9

The Calculation of available NPSH has been performed as a limiting analysis that is applicable for the duration of the LOCA event.

1.3.2 Debris Sources and Generation

1.3.2.1

A number of breaks in each high-pressure system that relies on recirculation are considered to ensure that the breaks that bound variations in debris generation by the size, quantity and type of debris are identified (Reference 9).

Based on various postulated break locations, the following break locations were evaluated per the methodology in the guidance document NEI 04-07, as modified by the NRC's Safety Evaluation Report, to maximize the postulated debris created:

1. The interim leg at the inlet to the loop D reactor coolant pump (RCP) at approximate elevation 386'-0" is the largest postulated line in containment and will affect a large amount of fiber (Transco RMI and reflective mirror insulation on the major equipment and piping inside the missile barrier). It also is the most direct path to the sump. This is the limiting break for Braidwood and Byron since it has the greatest coating debris quantity, which dominates the fiber/particulate head loss.
2. The loop A cold leg between the reactor coolant loop isolation valve and the reactor shield wall at elevation 393'-0" is chosen because it is another large break that will create the greatest mix of insulation debris types.

It is also located farther from the sump, which will create a different transport path for debris.

3. The loop D hot leg between the valve and the reactor shield wall at elevation 393’-0” is chosen because it generates the largest amount of fiber debris in Unit 1 of Braidwood and Byron.
4. Additionally, an alternate break is evaluated at the 14-inch pressurizer surge line (branch off of the reactor coolant system loop D line) at the connection to the pressurizer. This break would damage the reflective mirror insulation on most piping in loop D and a small amount in loop A, with the exception of piping near the top of the pressurizer. The loop D RCP and pressurizer insulation would also be damaged. For Unit 1 of Braidwood and Byron only, the fiber insulation on the loop D SG will also be damaged.

1.3.2.2

Insulation

The insulation for most lines and equipment is nearly identical in all four units. The majority of the insulation is Mirror RMI. Sections for the SGs in Unit 1 for both Braidwood and Byron are insulated with Transco RMI and Transco Thermal Wrap. The associated Braidwood and Byron Unit 1 SG piping connections (Main Steam, Feedwater, Auxiliary Feedwater, Steam Generator Blowdown) also have sections of Transco Thermal Wrap. The thermal wrap insulation that was located within the Zone of Influence (ZOI) for this insulation type has been replaced with reflective metal insulation for Braidwood Unit 1 and Byron Unit 1.

The ZOI characteristics for the RMI and thermal wrap were applied using the criteria established in the NRC Generic Letter (GL) 2004-02 Safety Evaluation, Table 3-2.

Coating

In accordance with the NRC’s Safety Evaluation Report for NEI 04-07, a ZOI of ten pipe diameters (10D) was used for the qualified coating. All unqualified coating was assumed to fail regardless of location inside the containment.

Foreign Material

The quantity and type of foreign material inside containment was based on walkdown data performed for both units at Byron and Braidwood. The foreign material included self-adhesive labels, stickers, placards, etc. The foreign material includes all identified foreign material in containment, and per the above referenced guidance, appropriate quantities were assumed to transport to the sump. In addition, 200 ft2 of degraded qualified coating was considered in the debris mix.

Latent Debris

A latent debris walkdown was performed at Byron and Braidwood in accordance with the NRC's Safety Evaluation Report for NEI 04-07, Section 3.5. Using a masolin cloth, samples were collected from the various surfaces at different floor elevations and when practical, different locations on each floor. When a surface was not accessible for sampling, an alternate surface was selected and noted on the walkdown report, such as circular pipe for an inaccessible circular duct. The net weight differences between the pre-sample and post-sample weight were used to statistically extrapolate the amount of latent debris for the entire containment using a 90% confidence level.

1.3.2.6

In addition to debris generated by jet forces from the pipe rupture, debris created by the resulting containment environment (thermal and chemical) have been considered in the analyses. Examples of this type of debris would be disbondment of coatings in the form of chips and particulates or formation of chemical debris (precipitants) caused by chemical reactions in the pool.

Head loss testing for the replacement screens has been performed using debris that included post-accident chemical effects.

1.3.2.7

All insulation within the ZOI has been accounted in the debris term. Continued degradation due to cascading water is irrelevant.

1.3.3 Debris Transport

The transport of the debris from the break location to the sump screen is evaluated (Reference 10) using the methods outlined in section 3.6 of NEI 04-07 as amended by the NRC SER. The means of transport considered are blowdown, washdown, pool fill, and recirculation for all types of debris. The recirculation transport analysis was performed by Sargent & Lundy using computational fluid dynamics (CFD) models developed using the computer program FLUENT. The CFD models were created by RWDI, Inc. Outputs of the CFD analysis include global (entire containment) and local (near sump pit) velocity contours, turbulent kinetic energy (TKE) contours, path lines and flow distributions for various scenarios. All particulate and coating debris was modeled as fines and 100% transports to the screen. The debris transport phenomena due to the blowdown, washdown, pool fill-up, and recirculation transport modes are summarized using debris transport logic trees consistent with the Drywell Debris Transport Study (DDTS) documented in NUREG/CR-6369, "Drywell Debris Transport Study." The debris transport logic trees consider the effect of dislocation, hold up on the floor or other structures, deposition in active or inactive pools, lift over curbs, and erosion of debris. Miscellaneous (foreign material) debris (tape, labels, etc.) is included in the debris load, and considered in the screen design as a sacrificial area. All miscellaneous debris is assumed to be 100% transportable.

The following is a summary of the overall transport fractions for all debris types:

Debris Type	Overall Transport Fraction
RMI	0.142
Qualified Coatings	1.00
Unqualified Coatings	1.00
Latent Debris	1.00
Foreign Material	1.00

The transport fractions presented above are bounding for all break locations, including a break in the RCS piping above the sump.

1.3.3.1

The debris quantities that have been used in the design basis testing for the filters are bounding values. No credit is taken for reduced debris accumulation at the filters that may be due to the actual sequence of debris accumulation at the filters during the event.

1.3.3.2

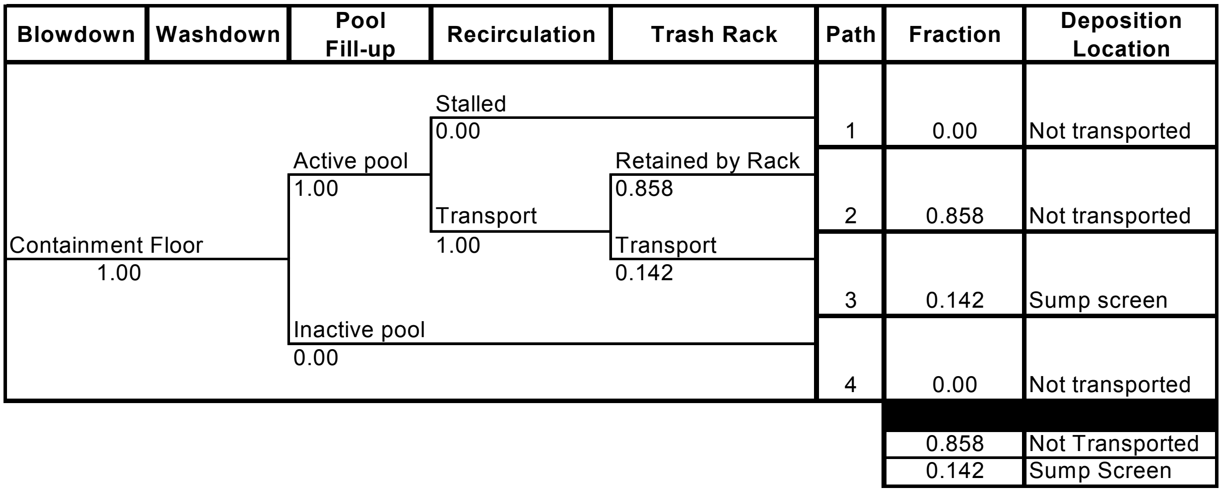
Based on the results of the CFD analysis, all (100%) coating, latent, and foreign material debris transports to the sump screen for all scenarios, except for Reflective Metal Insulation Debris.

NUREG/CR-3616 documents debris transport properties for stainless steel RMI. The following observations pertaining to RMI transport were made.

- Thick sheets of foil require higher velocities for transport than thin sheets; i.e. transport velocity tends to increase with material thickness
- Crumpled foil tends to transport at lower velocities than uncrumpled foil
- Velocity of motion of samples (crumpled or uncrumpled foil) is much less than the flow velocity
- RMI transport modes include folding, tumbling, rolling, and sliding along the floor
- RMI does not become "waterborne" during transport; i.e. a portion of the foil is always in contact with the floor. Therefore the velocity contours at 1 inch above the floor are considered acceptable for RMI
- Walls tend to hinder transport
- Interaction of foil pieces with each other often causes jamming and immobilization of the pieces; high flow velocities are then required to break up jams and resume transport
- Because RMI does not become "waterborne" during transport; i.e. a portion of the foil is always in contact with the floor, it does not cause screen blockage to a height greater than the height and width of the debris; i.e. the RMI accumulates on the floor when it transports to a screen

Results from the Computational Fluid Dynamic Analysis of containment show high velocity (>0.40 ft/s) transport paths to the sump pit area for all scenarios modeled. Therefore, all RMI debris can be transported to the vicinity of the sump, regardless of whether 1 or 2 trains are operating. However, the trash rack debris retainer has been designed to prevent large RMI debris (debris too large to pass through a 4-inch by 4-inch opening) from being transported to the sump. Per NUREG/CR-3616 (Ref.7.4.1) RMI debris transports by rolling, tumbling, and sliding and does not become “waterborne” (see §4.1.5.4). Since the large RMI debris would have to become “waterborne” to transport over the trash rack’s approximately fourteen inch long by approximately six inch high debris retainer, it will be retained and accumulate on the floor in front of the trash rack. With conservatively equating the RMI debris larger than six inches with the debris too large to pass through a 4-inch by 4-inch opening only the RMI debris less than 6 inches transports to the sump screens. Per Reference 11, a debris pile may form in front of the trash rack debris retainer. Large RMI debris may be drawn through the trash rack openings by climbing over this pile. This mode of transport would be restricted by the interaction of the foil pieces which often causes jamming and immobilization, the height of the debris pile, and the presence of the horizontal debris retainer. The conservatism included in the calculation of the amount of small debris that is transported to the sump screen will bound the minor fraction of the large debris that may transport past the trash rack by this path.

The figure below gives the Transport Logic Tree for RMI debris:



RMI transport fractions that were determined using the logic trees are summarized in the table below:

Debris Type	Size Distribution (Fraction)	Debris Transport Fraction (Applicable to All Scenarios) Before CS	Debris Transport Fraction (Applicable to All Scenarios) After CS	Fraction of Debris at Sump Screen (Applicable to All Scenarios) Before CS	Fraction of Debris at Sump Screen (Applicable to All Scenarios) After CS
Mirror RMI					
< 2 inches	0.061	1.00	1.00	0.061	0.061
>2 inches, < 6 inches	0.081	1.00	1.00	0.081	0.081
> 6 inches	0.858	0.0	0.0	0.0	0.0
Sum	1.0	-	-	0.142	0.142

Thus it can be seen that 14.2% of Mirror RMI foil debris transports to the sump screen for all scenarios.

Large RMI debris, relatively intact RMI, end covers, etc. due to RCS line breaks above the sump do not transport to the sump screen because the screen is sufficiently protected from blowdown debris by the top plate of the trash rack.

1.3.3.3
The recirculation analysis (Reference 10) considers the maximum recirculation flow rates.

The following is a summary of the overall transport fractions for all debris types:

Debris Type	Overall Transport Fraction
RMI	0.142
Qualified Coatings	1.00
Unqualified Coatings	1.00
Latent Debris	1.00
Foreign Material	1.00

The transport fractions presented above are bounding for all break locations, including a break in the RCS piping above the sump.

1.3.3.4

The debris transport analysis used computational fluid dynamics (CFD) simulations in combination with the experimental debris transport data.

The following is a summary of the overall transport fractions for all debris types:

Debris Type	Overall Transport Fraction
RMI	0.142
Qualified Coatings	1.00
Unqualified Coatings	1.00
Latent Debris	1.00
Foreign Material	1.00

The transport fractions presented above are bounding for all break locations, including a break in the RCS piping above the sump.

1.3.3.5

Both ECCS sump openings are protected by a trash rack above elevation 377 ft. The trash rack prevents heavier debris from entering the sump pits.

1.3.3.6

All debris that has been evaluated to reach the sump pits is accounted for in the head loss analysis.

1.3.3.7

The head loss evaluation assumes that the 100% debris load is present at the sump filters at the time the RHR pump suction switches over to the Recirculation sumps. Also, maximum flow rates are considered, including the flow from the Containment Spray pump.

1.3.3.8
The debris quantity that has been calculated to reach the sump screens does not apply any reduction factor due to the results of airborne or containment spray washdown debris transport analyses.

Flood level analyses inside containment show that the minimum flood level will be sufficient to fully submerge the recirculation sump filters. The minimum flood level analysis accounts for water inventory hold-up inside containment.

1.3.3.9
The effects of floating or buoyant debris have been evaluated for the integrity of the trash rack. The effects of floating or buoyant debris on head loss are not evaluated because the sump filters are fully submerged.

1.3.4 Debris Accumulation and Head Loss

1.3.4.1
The debris accumulation on the sump filters accounts for the total debris quantity that has been calculated. The full debris quantity (scaled accordingly) has been used in the vendor head loss testing.

1.3.4.2
The containment recirculation sump screens are entirely located within the sump pit below the containment floor elevation of 377 ft. Calculations for minimum containment flood level have demonstrated that the sump screens will be submerged at the time of ECCS Switchover to the containment recirculation sump. The CS pumps are switched over to the containment recirculation sumps later than the RHR pumps. The flood level increases as the LOCA event progresses; thus, the sump screens will be submerged at the time of CS Switchover.

The head loss through the sump screens has been determined by testing a model of the screens under debris loading conditions scaled accordingly from the debris quantities that have been calculated specific to Byron and Braidwood.

1.3.4.3
The NPSH analysis at high temperature follows the current Byron and Braidwood licensing basis. The containment pressure and sump water vapor pressure are assumed to be equal. This assumption assures that credit is not taken for increases in containment pressure due to the accident.

As part of the chemical effects evaluations related to head loss through the containment recirculation sump strainers, the NPSH analysis for the RHR pumps has been performed at low temperatures. In accordance with the requirements specified in Regulatory Guide 1.1, the NPSH analysis at low temperatures assumes the containment atmospheric pressure is equal to the minimum containment atmospheric pressure that would be present inside containment before the Loss of Coolant Accident (LOCA) event. This analysis does not credit calculated increases in containment pressure as a result of the LOCA.

1.3.4.4

The sump screens at Byron and Braidwood are fully submerged at the times the RH and CS pumps' suctions are switched over to the containment recirculation sumps.

1.3.4.5

The head loss through the recirculation sump screens has been determined by testing. The head loss through the recirculation screen assembly, downstream of the screens, has been determined via a Computational Fluid Dynamic (CFD) analysis. (Reference 7). The total head loss through the recirculation sump screens has been determined to be 4.20 ft at a temperature of 200 °F.

1.3.4.6

The screen head loss has been determined by testing based on limiting debris quantities specific to Byron and Braidwood.

References:

1. Design Analysis #6.1.2.6-BRW-06-0029-S (Braidwood), #6.1.2.6-BYR06-031 (Byron)
2. Design Analysis #BRW-06-0035-M (Braidwood), #BYR06-058 (Byron)
3. Design Analysis #BRW-06-0016-M (Braidwood), # BYR06-029 (Byron)
4. Design Analysis #BRW-05-0061-M (Braidwood), #BYR05-043 (Byron)
5. Design Analysis #BRW-05-0063-M (Braidwood), #BYR05-061 (Byron)
6. Design Analysis #BRW-05-0084-M (Braidwood), #BYR06-017 (Byron)
7. Design Analysis #3 SA-096.018
8. Design Analysis #SI-90-01
9. Design Analysis #BRW-05-0059-M (Braidwood), #BYR05-041 (Byron)
10. Design Analysis #BRW-05-0060-M (Braidwood), #BYR05-042 (Byron)
11. Design Analysis #BRW-06-0015-M (Braidwood), #BYR06-025 (Byron)
12. Design Analysis #CAE-07-49/CCE-07-48
13. Letter from K. R. Jury (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "Supplement to Exelon Response to NRC Generic Letter 2004-02, 'Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors,' " dated May 31, 2006
14. Letter from R. F. Kuntz (U. S. Nuclear Regulatory Commission) to C. M. Crane (Exelon Generation Company, LLC), "Byron Station, Unit No.1 and Braidwood Station, Unit 2 - Requested Extension of Completion Schedule for NRC Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized-Water Reactors,' " dated July 21, 2006
15. Procedure BwOS TRM 2.5.b.1 for Braidwood
16. Procedure BOSR Z.5.b.1-1 for Byron

REGULATORY GUIDE 1.83

INSERVICE INSPECTION OF PRESSURIZED WATER REACTOR STEAM GENERATOR TUBES

Regulatory Guide 1.83 describes an acceptable method of complying with the Commission's regulations with regard to inservice inspection of pressurized water reactor steam generator tubes. The plant design includes the features of Regulatory Position C.1.

The preservice and inservice inspection of steam generator tubing was conducted in accordance with Regulatory Positions C.2 through C.8 of Regulatory Guide 1.83, Revision 1, as modified by the Technical Specifications for Byron/Braidwood. Nuclear Energy Institute (NEI) 97-06 "Steam Generator Program Guidelines," supercedes NRC Regulatory Guide 1.83 for inservice inspection requirements. NEI 97-06 was approved for use by the NRC via License Amendments 150 and 179 for Byron and License Amendments 144 and 172 for Braidwood.

REGULATORY GUIDE 1.84

DESIGN AND FABRICATION CODE CASE ACCEPTABILITY ASME SECTION III DIVISION 1

The Licensee complies with the regulatory position. This regulatory guide lists those section III ASME code cases relevant to design and fabrication that are generally acceptable to the NRC for implementation in the licensing of light-water-cooled nuclear power plants.

Code cases explain the intent of code rules or provide alternative requirements under special conditions. Implementation of individual code cases is limited to the requirements as specified in the inquiry and reply sections of each code case. The ASME considers the use of code cases to be optional for the user and not a mandatory requirement. Use of this regulatory guide is optional.

Approval of code cases listed in this regulatory guide is by code case number and date of ASME approval. Components ordered to a specific version of a code case need not be changed because a subsequent revision to the code case is listed as the approved version in subsequent revisions of the regulatory guide. Similarly, components ordered to a code case that was previously approved for use need not be changed because the code case has been subsequently annulled.

Code cases on the approved list may be applied to components that were in the process of construction prior to the effective date of the code case within the limits specified in the code case and applicable regulations, or recommended in other regulatory guides.

Code cases listed in this regulatory guide are generically acceptable for implementation. Beginning with Revision 16 to this regulatory guide, it is no longer necessary to obtain NRC approval to use code cases listed in the regulatory guide.

Code cases not listed in this regulatory guide cannot be implemented unless formal approval is obtained from the Commission in accordance with footnote 6 of the Codes and Standards Rule, 10CFR 50.55a.

Components with long lead times were ordered prior to the original effective dates for Regulatory Guides 1.84 and 1.85. Nevertheless, there are no known examples of code cases being applied to components, except those approved by either Regulatory Guide 1.84 or 1.85, with the following exceptions or special conditions:

Code Case 1528: Fracture toughness information for this code used in the construction of the steam generators and pressurizers has been supplied to the NRC by WCAP-9292, March 1978, "Dynamic Fracture Toughness of ASME SA-508 Class 2a and ASME SA-533 Grade A Class 2 Base and Heat Affected Zone Material and Applicable Weld Metals."

Code Case 1637: This code case was used for the purchase of heat exchanger tubing. Authorization for its use was obtained from the NRC.

Refer to Subsections 5.4.2 and 5.2.1 for further information.

In addition, the following code cases have been approved by the NRC. This list is subject to change based on regulatory guide revisions. Code cases approved for use by Regulatory Guide 1.84 need not be listed here prior to being implemented.

CODE CASE	REGULATORY GUIDE REVISION	ASME APPROVAL DATE	TITLE
N-272	18	05/15/80	Compiling Data Report Records, Section III, Division 1
N-275	18	05/18/80	Repair of Welds, Section III, Division 1
N-292	19	01/05/81	Depositing Weld Metal Prior to Preparing Ends for Welding, Section III, Division 1, Class 1, 2, and 3
N-340	N/A	06/17/82	Alternate Rules for Examination of Weld Edge Preparation, Section III, Division 1, Classes 1, 2, MC, and CS. (Licensee to Provide Justification Each Time Code Case N-340 is Used.)
N-403	N/A	02/14/85	Reassembly of Subsection NF Component and Piping Supports, Section III, Division 1
N-411	24	09/17/84	Alternative Damping Values for Seismic Analysis of Class 1, 2, and 3 Piping, Section III, Division 1
N-413	24	02/14/85	Minimum Size of Fillet Welds for Subsection NF Linear Type Supports, Section III, Division 1

The status of code case approval is continually changing; however, the rules for use of this regulatory guide normally do not change. Therefore, the above discussions are applicable to any revision of this regulatory guide, provided the limitations of the regulatory guide revision are adhered to.

REGULATORY GUIDE 1.85

MATERIALS CODE ACCEPTABILITY
ASME SECTION III DIVISION 1

The Licensee complies with the regulatory position. This regulatory guide lists those Section III ASME code cases relevant to materials and testing that are generally acceptable to the NRC for implementation in the licensing of light-water-cooled nuclear power plants.

Code cases explain the intent of Code rules or provide alternative requirements under special conditions. Implementation of individual code cases is limited to the requirements as specified in the inquiry and reply sections of each code case. The ASME considers the use of code cases to be optional for the user and not a mandatory requirement. Use of this regulatory guide is optional.

Approval of code cases listed in this regulatory guide is by code case number and date of ASME approval. Components ordered to a specific version of a code case need not be changed because a subsequent revision to the code case is listed as the approved version in subsequent revisions of the regulatory guide. Similarly, components ordered to a code case that was previously approved for use need not be changed because the code case has been subsequently annulled.

Code cases on the approved list may be applied to components that were in the process of construction prior to the effective date of the code case within the limits specified in the code case and applicable regulations, or recommended in other regulatory guides.

Code cases listed in this regulatory guide are generically acceptable for implementation. Beginning with Revision 16 to this regulatory guide, it is no longer necessary to obtain NRC approval to use code cases listed in the regulatory guide.

Code cases not listed in this regulatory guide cannot be implemented unless formal approval is obtained from the Commission in accordance with footnote 6 of the Codes and Standards Rule, 10CFR 50.55a.

Components with long lead times were ordered prior to the original effective dates for Regulatory Guides 1.84 and 1.85. Nevertheless, there are no known examples of code cases being applied to components except those approved by either Regulatory Guide 1.84 or 1.85, with the following exceptions or special conditions:

Code Case 1528: Fracture toughness information for this code used in the construction of the steam generators and pressurizers has been supplied to the NRC by WCAP-9292, March 1978, "Dynamic Fracture Toughness of ASME SA-508 Class 2a and ASME SA-533 Grade A Class 2 Base and Heat Affected Zone Material and Applicable Weld Metals."

Code Case 1637: This code case was used for the purchase of heat exchanger tubing. Authorization for its use was obtained from the NRC.

Code Case N-242-1: Paragraphs 1 through 4 of this code case are used in the acceptance of limited amounts of materials in ASME Section III systems.

Refer to Subsections 5.4.2 and 5.2.1 for further information.

In addition, the following code cases have been approved by the NRC. This list is subject to change based on regulatory guide revisions. Code cases approved for use by Regulatory Guide 1.85 need not be listed here prior to being implemented.

CODE CASE	REGULATORY GUIDE REVISION	ASME APPROVAL DATE	TITLE
N-242-1	18	04/10/80	Material Certification, Section III, Division 1, Classes 1, 2, 3, MC, and CS Construction
N-295	19	01/15/81	Use of Previously Produced Material, NCA-1140

The status of code case approval is continually changing; however, the rules for use of this regulatory guide normally do not change. Therefore, the above discussions are applicable to any revision of this regulatory guide, provided the limitations of the regulatory guide revision are adhered to.

REGULATORY GUIDE 1.86

TERMINATION OF OPERATING LICENSES FOR NUCLEAR REACTORS

Revision 0 of Regulatory Guide 1.86 will be complied with by the regulatory staff for the termination of operating licenses at the end of the station design life.

REGULATORY GUIDE 1.87

GUIDANCE FOR CONSTRUCTION OF CLASS 1 COMPONENTS IN ELEVATED-TEMPERATURE REACTORS

This guide is not pertinent to this application, since the Byron/Braidwood reactors are not "high-temperature reactors."

REGULATORY GUIDE 1.88

COLLECTION, STORAGE, AND MAINTENANCE OF NUCLEAR POWER PLANT QUALITY ASSURANCE RECORDS

The requirements of Regulatory Guide 1.88 have been incorporated in Regulatory Guide 1.28, Revision 3. Regulatory Guide 1.88 was withdrawn on June 17, 1991.

The Licensee complies with the intent of Regulatory Guide 1.28 Revision 3, but applies it to ANSI/ASME NQA-1-1994.

REGULATORY GUIDE 1.89

QUALIFICATION OF CLASS 1E EQUIPMENT FOR NUCLEAR POWER PLANTS

NSSS Scope

For Westinghouse NSSS Class 1E equipment, Westinghouse has met the requirements of IEEE-323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations" including the Nuclear Power Engineering Committee (NPEC) Position Statement of July 24, 1975, and Regulatory Guide 1.89, Revision 0, by providing an appropriate combination of the following: type testing, operating experience, qualification by analysis, and on-going qualification. This commitment has been satisfied by NRC review and approval of Westinghouse Topical Report WCAP-8587.

Non-NSSS Scope

The extent of the Licensee's commitment to comply with the requirements of Regulatory Guide 1.89 is presented in Subsections 3.11.2 and 8.1.16.

The Licensee follows the provisions in Revision 1 of Regulatory Guide 1.89 to qualify replacement electric equipment installed subsequent to February 22, 1993. Revision 1 provides NRC-accepted reasons to allow exception to qualification requirements.

REGULATORY GUIDE 1.90

INSERVICE INSPECTION OF PRESTRESSED CONCRETE CONTAINMENT STRUCTURES WITH GROUTED TENDONS

The containment design does not use grouted tendons. Thus, this guide is not applicable to Byron/Braidwood.

REGULATORY GUIDE 1.91

EVALUATION OF EXPLOSIONS POSTULATED TO OCCUR ON TRANSPORTATION ROUTES NEAR NUCLEAR POWER PLANTS

The plant design conforms to Revision 1 of this regulatory guide as described in Subsection 2.2.3.2.

REGULATORY GUIDE 1.92

COMBINING MODAL RESPONSES AND SPATIAL COMPONENTS IN SEISMIC RESPONSE ANALYSIS

The plant design conforms to Revision 1 of this regulatory guide as described in Subsections 3.7.2 and 3.7.3.7.

REGULATORY GUIDE 1.93

AVAILABILITY OF ELECTRIC POWER SOURCES

Availability of electric power sources is discussed in the Technical Specifications.

The Licensee complies with the requirements in Revision 0 of this guide with the following exceptions and clarifications:

> Regulatory Positions C.1, C.2, and C.4 refer to a 72-hour time interval for power operation when the available power sources are less than the "Limiting Conditions for Operation."
>
> The operating time limits delineated in regulatory positions C.1 through C.5 are explicitly for corrective maintenance activities only. These operating time limits should not be construed to include preventive maintenance activities that require the incapacitation of any required electric power source. Therefore, per this guide, preventive maintenance should be scheduled for performance during cold shutdown and/or refueling periods.
>
> The Licensee has determined that performance of preventive maintenance activities on the system auxiliary transformers and Emergency Diesel Generators may be safely performed with both units at power. Therefore, system auxiliary transformer and Emergency Diesel Generator preventive maintenance activities may be performed during periods other than cold shutdown and/or refueling.

REGULATORY GUIDE 1.94

QUALITY ASSURANCE REQUIREMENTS FOR INSTALLATION, INSPECTION, AND TESTING OF STRUCTURAL CONCRETE AND STRUCTURAL STEEL DURING THE CONSTRUCTION PHASE OF NUCLEAR POWER PLANTS

Regulatory Guide 1.94 endorsed ANSI N45.2.5, Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants. NQA-1-1994, Subpart 2.5, Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete, Structural Steel, Soils, and Foundations for Nuclear Power Plants supersedes this commitment to Regulatory Guide 1.94 and ANSI N45.2.5 as documented in the Quality Assurance Topical Report (NO-AA-10). For specific information relating to concrete standards, refer to Appendix B.

REGULATORY GUIDE 1.95

PROTECTION OF NUCLEAR POWER PLANT CONTROL ROOM OPERATORS AGAINST AN ACCIDENTAL CHLORINE RELEASE

The Licensee complies with the requirements in Revision 1 of this guide, as discussed in Subsection 6.4.4.

REGULATORY GUIDE 1.96

DESIGN OF MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEMS FOR BOILING WATER REACTOR NUCLEAR POWER PLANTS

The requirements of this guide are not applicable to pressurized water reactors.

REGULATORY GUIDE 1.97

INSTRUMENTATION FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS TO ASSESS PLANT AND ENVIRONS CONDITIONS DURING AND FOLLOWING AN ACCIDENT

Compliance with Revision 3 of Regulatory Guide 1.97 was discussed in two letters. The Preliminary Report on compliance to Regulatory Guide 1.97 is in a letter from K.A. Ainger of CECo to H.R. Denton of the NRC dated February 27, 1987. The Final Report is in a letter from Steve Hunsader of CECo to T. E. Murley of the NRC dated September 1, 1987. These transmittals, which furnished a report of compliance with Regulatory Guide 1.97, met the license conditions for the Byron/Braidwood Stations. Refer to UFSAR Sections 6.2, 6.5, 7.5, 11.5, E.21, E.30, and E.75 for further discussion.

The hydrogen monitoring system was originally designed to meet the requirements of Regulatory Guide 1.97 Category 1 instruments. Regulatory Guide 1.97 Category 1 is intended for key variables that most directly indicate the accomplishment of a safety function for design basis accident events and provides for full qualification, redundancy, and continuous real-time display and requires onsite (standby) power. Based on a revision to 10 CFR 50.44 which eliminated the design basis loss-of-coolant hydrogen release, the hydrogen monitors have been reclassified as Regulatory Guide 1.97 Category 3 instruments.

REGULATORY GUIDE 1.98

ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A RADIOACTIVE OFF-GAS SYSTEM FAILURE IN A BOILING WATER REACTOR

The requirements of this guide are not applicable to pressurized water reactors.

REGULATORY GUIDE 1.99

RADIATION EMBRITTLEMENT OF REACTOR VESSEL MATERIALS

Regulatory Guide 1.99 was issued after procurement of the reactor vessels.

The vessel materials used in the construction of the reactor vessels have been evaluated using the methods provided within Regulatory Guide 1.99, Revision 2. The end-of-life adjusted reference temperature (ART) at the 1/4 thickness (1/4T) position in the vessel wall is less than 200 °F for each of the reactor vessels. The ART predictions, therefore, are in agreement with Regulatory Position C.3.

Refer to UFSAR Section 5.3 for further discussion.

REGULATORY GUIDE 1.100

SEISMIC QUALIFICATION OF ELECTRIC EQUIPMENT FOR NUCLEAR POWER PLANTS

NSSS Scope

The Licensee is in compliance with the objectives in Revision 1 of Regulatory Guide 1.100. The Westinghouse program for seismic qualification of safety-related electrical equipment is delineated in WCAP-8587, Revision 1, which has been reviewed by the NRC. For further details refer to Section 3.10.

Non-NSSS Scope

The Licensee complies with the objectives in Revision 1 of Regulatory Guide 1.100. The Licensee's approach to seismic qualification of Class 1E equipment is discussed in Section 3.10.

Revision 1 of Regulatory Guide 1.100 was in effect when the operating license applications were docketed.

REGULATORY GUIDE 1.101

EMERGENCY PLANNING AND PREPAREDNESS FOR NUCLEAR POWER REACTORS

The guidance provided by Regulatory Guide 1.101, Revision 3, was utilized in the preparation of the Licensee's emergency response plans. The Licensee complies with this regulatory guide as described in the Exelon Nuclear Standardized Radiological Emergency Plan.

REGULATORY GUIDE 1.102

FLOOD PROTECTION FOR NUCLEAR POWER PLANTS

The plant design conforms to Revision 1 of this regulatory guide as described in Sections 2.4 and 3.4.

REGULATORY GUIDE 1.103

POST-TENSIONED PRESTRESSING SYSTEMS FOR CONCRETE REACTOR VESSELS AND CONTAINMENTS

The plant design conforms to the regulatory positions in Revision 1 as described in Subsection 3.8.1 and Appendix B.3. The requirements remain in effect even though the regulatory guide was withdrawn on July 8, 1981. The regulatory positions are now covered by one or more national standards.

REGULATORY GUIDE 1.104

OVERHEAD CRANE HANDLING SYSTEMS FOR NUCLEAR POWER PLANTS

Regulatory Guide 1.104 was withdrawn on August 16, 1979 because the information it contained was published in NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants." Byron and Braidwood Stations follow the recommendations of NUREG-0554.

REGULATORY GUIDE 1.105

INSTRUMENT SETPOINTS

The Licensee complies with the regulatory position in Revision 1 with the following exceptions keyed to paragraph numbers in the position. Revision 1 was in effect when the operating license applications were docketed.

Position C.5, requires locking devices on instrument setpoint adjustment mechanisms. We have not specified such locking devices on our instrument data sheets, so that these would only be available to the extent that they are standard equipment. In general, locking devices are not required to maintain stable instrument setpoints and we believe that setpoint stability will not be improved by providing locking devices.

Position C.6, requires documentation of the assumptions used in selecting setpoint values and the margins between the setpoints and the limiting safety system values. The documentation is to include definition of instrument setpoint drift rate and the relationship of the drift rate to testing intervals. The Byron/Braidwood design conforms to this position only to the degree that setpoints are documented on the instrument data sheets along with instrument range and the maximum range of the parameter being measured. With respect to the other requirements of Position C.6, generic drift rates are not generally available for any instruments since drift rates would be affected by the particular service to which the instrument was subjected. Testing intervals are set on the basis of past experience with the specific instrument types in question.

REGULATORY GUIDE 1.106

THERMAL OVERLOAD PROTECTION FOR ELECTRIC MOTORS ON MOTOR-OPERATED VALVES

The Licensee complies with the requirements in Revision 1 of this regulatory guide. The Licensee has selected the method described in Regulatory Position C.2.

REGULATORY GUIDE 1.107

QUALIFICATIONS FOR CEMENT GROUTING FOR PRESTRESSING TENDONS IN CONTAINMENT STRUCTURES

The Byron/Braidwood containment design does not use grouted tendons; therefore, this guide is not applicable to Byron and Braidwood stations.

REGULATORY GUIDE 1.108

PERIODIC TESTING OF DIESEL GENERATORS USED AS ONSITE ELECTRIC POWER SYSTEMS AT NUCLEAR POWER PLANTS

The guidance in Regulatory Guide 1.108 has been updated and incorporated into Revision 3 of Regulatory Guide 1.9. Regulatory Guide 1.108 was withdrawn on July 19, 1993.

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

The Licensee complies with the position in Revision 1 of this regulatory guide as presented in Subsections 11.2.3.3 and 11.3.3.7.

REGULATORY GUIDE 1.110

COST BENEFIT ANALYSIS FOR RADWASTE SYSTEMS FOR LIGHT-WATER-COOLED NUCLEAR POWER REACTORS

The Licensee complies with the Annex to 10 CFR 50 Appendix I; therefore, this guide is not applicable.

REGULATORY GUIDE 1.111

METHODS FOR ESTIMATING ATMOSPHERIC TRANSPORT AND DISPERSION OF GASEOUS EFFLUENTS IN ROUTINE RELEASES FROM LIGHT-WATER-COOLED REACTORS

The Licensee complies with the requirements in Revision 1 of this guide. This is discussed further in Section 2.3.

REGULATORY GUIDE 1.112

CALCULATION OF RELEASES OF RADIOACTIVE MATERIALS IN GASEOUS AND LIQUID EFFLUENTS FROM LIGHT-WATER-COOLED POWER REACTORS

The Licensee complies with the requirements in Revision 0-R of this guide. This is discussed further in Sections 11.2 and 11.3.

REGULATORY GUIDE 1.113

ESTIMATING AQUATIC DISPERSION OF EFFLUENTS FROM ACCIDENTAL AND ROUTINE REACTOR RELEASES FOR THE PURPOSE OF IMPLEMENTING APPENDIX I

The Licensee complies with the position in Revision 1 of this regulatory guide as presented in Subsections 2.4.12, 2.4.13.3, and 11.2.3.

REGULATORY GUIDE 1.114

GUIDANCE TO OPERATORS AT THE CONTROLS AND TO SENIOR OPERATORS IN THE CONTROL ROOM OF A NUCLEAR POWER UNIT

The Licensee complies with the requirements in Revision 2 of this guide. Refer to Subsection 13.1.2.2 for further information.

REGULATORY GUIDE 1.115

PROTECTION AGAINST LOW TRAJECTORY TURBINE MISSILES

The Licensee meets the objectives set forth in Revision 1 of this regulatory guide as presented in Section 3.5 and Section 10.2.3.

REGULATORY GUIDE 1.116

QUALITY ASSURANCE REQUIREMENTS FOR INSTALLATION, INSPECTION, AND TESTING OF MECHANICAL EQUIPMENT AND SYSTEMS

Regulatory Guide 1.116 endorsed ANSI N45.2.8, Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems. NQA-1-1994, Subpart 2.8, Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems for Nuclear Power Plants supersedes this commitment to Regulatory Guide 1.116 and ANSI N45.2.8 as documented in the Quality Assurance Topical Report (NO-AA-10). Refer to Topical Report NO-AA-10 for further information on the Quality Assurance Program.

REGULATORY GUIDE 1.117

TORNADO DESIGN CLASSIFICATION

This guide is applicable only to construction permit applications docketed after May 30, 1978. The Byron/Braidwood construction permit application was docketed prior to this date. For a discussion of the Byron/Braidwood design, refer to Section 3.2.

REGULATORY GUIDE 1.118

PERIODIC TESTING OF ELECTRIC POWER AND PROTECTION SYSTEMS

The Licensee complies with the regulatory positions in Revision 3 of this regulatory guide with the following exception:

Regulatory Position C

Exception is taken to including the signal conditioning and actuation logic during the conduct of periodic RTS and ESFAS response time test. Implementation of WCAP-14036-P-A, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests", October 1998, allows the allocation of bounding response time values for these portions of the protection channels based on engineering data.

Exception is taken to including the pressure and differential pressure sensors during the conduct of periodic RTS and ESFAS response time tests. Implementation of WCAP-13632, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements", August 1995, allows the allocation of bounding response time values for these portions of the protection channels based on engineering data.

Regulatory Position C.2

Exception is taken to the limitations placed on the use of jumpers or any other alterations, such as lifting leads, utilized to support safety system testing. In order to accomplish certain tests, it is necessary to use jumpers or lifted leads to simulate desired logic circuit conditions. The safe use of these alterations is ensured with detailed procedures, which include independent verification of the temporary circuit alteration and subsequent restoration.

REGULATORY GUIDE 1.119

SURVEILLANCE PROGRAM FOR NEW FUEL ASSEMBLY DESIGNS

This regulatory guide is not applicable to Byron and Braidwood Stations. It was withdrawn on June 23, 1977 because the NRC Staff believed that fuel surveillance programs should be plant specific and handled on a case-by-case basis rather than in a detailed generic manner.

REGULATORY GUIDE 1.120

FIRE PROTECTION GUIDELINES FOR NUCLEAR POWER PLANTS

The Licensee's positions on the regulatory positions in Revision 1 are provided in detail in Chapter 3.0 of the Exelon Generation Company report, "Byron/Braidwood Stations Fire Protection Report in Response to Appendix A of BTP APCSB 9.5-1" (current amendment).

REGULATORY GUIDE 1.121

BASES FOR PLUGGING DEGRADED PWR STEAM GENERATOR TUBES

The minimum acceptable wall thickness at which the tube must be removed from service by plugging satisfies the requirements of Regulatory Guide 1.121, Revision 0. The Licensee complies with the regulatory position with the following comments and exceptions keyed to paragraph numbers in the position:

Position C.1

The term "unacceptable defects" is interpreted as applying to those imperfections resulting from service induced mechanical or chemical degradation of the tube walls which have penetrated to a depth in excess of the plugging limit.

Position C.2.a(2) and C.2.a(4)

Westinghouse has documented its opinions on Regulatory Guide 1.121 by corporate letter that the Licensee concurs with. A major exception to this position is the margin of 3 against tube failure for normal operation. Tube failure is defined as plastic deformation of a crack to the extent that the sides of the crack open to a nonparallel, elliptical configuration. The tubing can sustain added internal pressure beyond those values before reaching a condition of gross failure. We have interpreted this to apply as an operating limit for the plant and consider that it introduces a conflict to the established conditions for plant operation as identified in the plant technical specifications. A factor of 3 is quite often used in ASME Code Design guidelines. These code practices apply to the design of hardware and to the analyses done on these designs. Conditions which occur during operation of the equipment and which may affect the equipment so that design values no longer apply, are not directly addressed by the initial code requirements. That is one reason why plant Technical Specifications have been generated to establish safe limits of operation for power station equipment. The ASME code is not applicable to the operational criteria of steam generator tubing. Our tubing design and tubing in the design condition has margins in excess of 3. In summary, we satisfy the margin of 3 if it were used in a Code sense as new equipment design. Moreover, we do not believe that this margin should be utilized as a limiting condition for normal operation.

Position C.2.b

In cases where sufficient inspection data exists to establish a degradation allowance, the rate used will be an average time rate determined from the mean of the test data. The combined effect of these requirements would be to establish a maximum permissible primary-to-secondary leak rate which may be below the threshold of detection with current methods of measurement. Westinghouse has determined the maximum acceptable length of a through-wall-crack based on secondary pipe break accident loadings which are typically twice the magnitude of normal operating pressure loads. Westinghouse will use a leak rate associated with the crack size determined on the basis of accident loadings.

Where requirements for minimum wall are markedly different for different areas of the tube bundle, e.g., U-bend area versus straight length in Westinghouse designs, alternate plugging limits may be established to address the varying requirements in a manner that does not require unnecessary plugging of tubes.

Position C.3.e(6)

Westinghouse supplied computer code names and references rather than the actual codes.

Nuclear Energy Institute (NEI) 97-06 “Steam Generator Program Guidelines,” as approved by the NRC via License Amendments 150 and 179 for Byron and License Amendments 144 and 172 for Braidwood, contains additional requirements for tube integrity in accordance with Technical Specification 3.4.19.

REGULATORY GUIDE 1.122

DEVELOPMENT OF FLOOR DESIGN RESPONSE SPECTRA FOR SEISMIC DESIGN OF FLOOR-SUPPORTED EQUIPMENT OR COMPONENTS

The plant design conforms to Revision 1 of this regulatory guide as described in Subsection 3.7.2.

REGULATORY GUIDE 1.123

QUALITY ASSURANCE REQUIREMENTS FOR CONTROL OF PROCUREMENT OF ITEMS AND SERVICES FOR NUCLEAR POWER PLANTS

The requirements in Regulatory Guide 1.123 have been incorporated in Regulatory Guide 1.28, Revision 3. Regulatory Guide 1.123 was withdrawn on June 17, 1991.

The Licensee complies with the intent of Regulatory Guide 1.28 Revision 3, but applies it to ANSI/ASME NQA-1-1994.

REGULATORY GUIDE 1.124

SERVICE LIMITS AND LOADING COMBINATIONS FOR CLASS 1 LINEAR-TYPE COMPONENT SUPPORTS

The design of the Byron/Braidwood NSSS components supports is in compliance with all of the applicable regulatory positions contained in Revision 1 of this regulatory guide. See Sections 3.6, 3.7, 3.9, and 3.10 for further information.

REGULATORY GUIDE 1.125

PHYSICAL MODELS FOR DESIGN AND OPERATION OF HYDRAULIC STRUCTURES AND SYSTEMS FOR NUCLEAR POWER PLANTS

This regulatory guide is applicable to construction permit applications docketed after March 1977. The Byron/Braidwood construction permit application was docketed prior to this date. There are no safety-related hydraulic model tests used in the design.

REGULATORY GUIDE 1.126

AN ACCEPTABLE MODEL AND RELATED STATISTICAL METHODS FOR THE ANALYSIS OF FUEL DENSIFICATION

Regulatory Guide 1.126 allows the use of NRC-approved vendor models rather than the model provided in the guide. The densification model presented in WCAP-8218 (proprietary) was approved by the NRC. The nonproprietary models in WCAP-8219 and WCAP-8264 are station-specific models based on the approved model.

REGULATORY GUIDE 1.127

INSPECTION OF WATER-CONTROL STRUCTURES ASSOCIATED WITH NUCLEAR POWER PLANTS

The Licensee meets the requirements of the regulatory position in Revision 1 of this guide.

REGULATORY GUIDE 1.128

INSTALLATION DESIGN AND INSTALLATION OF LARGE LEAD STORAGE BATTERIES FOR NUCLEAR POWER PLANTS

The Licensee complies with the requirements in Revision 1 of this guide with the exceptions and/or clarifications to the regulatory positions identified and justified below:

Regulatory Position C-1

In Subsection 4.1.4, "Ventilation," instead of the second sentence, the following should be used:

> "The ventilation system shall limit hydrogen concentration to less than two percent by volume at any location within the battery area."

Licensee's Position

The ventilation requirements set forth in IEEE Std. 484-1987 are adequate.

Justification of Licensee's Position

IEEE 484-1987 requires that the ventilation system limit hydrogen accumulation to less than 2% of the total volume of the battery area. This Regulatory Position would require that hydrogen accumulation be limited to less than 2% at any location within the battery area. The ventilation requirements as set forth in IEEE 484-1987 are entirely adequate. The "2% at any location" requirement would be almost impossible to verify and might even require the installation of ducts, vanes, and/or auxiliary fans so as to ensure that every "nook and cranny" is thoroughly purged.

The battery area ventilation system is designed to maintain the hydrogen concentration below 2% with a "run-away" charger (i.e., a charger delivering its full-rated output into a fully-charged battery, thereby causing gassing of all cells). Thus, any significant hydrogen build-up in the battery area would require two failures (a failure of the ventilation system, and a failure of the charger), both of which will be annunciated in the main control room.

Regulatory Position C-2

In Subsection 4.2.1, "Location," Item 1, the general requirement that the battery be protected against fire should be supplemented with the applicable recommendations in Regulatory Guide 1.120, "Fire Protection Guidelines for Nuclear Power Plants."

Licensee's Position

The reference to Regulatory Guide 1.120 is inappropriate because this regulatory guide is in the "comment" stage.

Justification of Licensee's Position

The battery location and protection against fire will be described in the Fire Protection Report in Response to Branch Technical Position APCSB 9.5.1 in lieu of Regulatory Guide 1.120. The location and fire protection requirements set forth in IEEE 484-1987 are adequate.

In reference to Regulatory Guide 1.120, Revision 1, (November 1977), Section C.6(g), Page 20, "Safety-Related Battery Rooms," Licensee's comments are as follows:

(a) This paragraph seems to imply that all safety-related batteries are to be located in separately - enclosed rooms. It is the Licensee's position that it should not be necessary that battery rooms be separated from other areas of the plant by barriers having a minimum fire rating of three hours. Such barriers would be necessary only if the batteries were in a separate fire protection zone. There is nothing wrong with a design wherein the battery is located in an open area so long as the battery is protected from mechanical damage; e.g., the battery may be located in an electrical equipment room but protected by an enclosing fence.

(b) The location of d-c switchgear and inverters in the electrical equipment room described above is a satisfactory arrangement.

Regulatory Position C-3

Items 1 through 5 of Subsection 4.2.2, "Mounting," should be supplemented with the following:

"6. Restraining channel beams and tie rods shall be electrically insulated from the cell case and shall also be in conformance with Item 2 above regarding moisture and acid resistance."

In addition, the general requirement in Item 5 to use IEEE 344-1975 should be supplemented by Regulatory Guide 1.100, "Seismic Qualification of Electric Equipment for Nuclear Power Plants."

Licensee's Position

Restraining channel beams and tie rods need not be electrically insulated from the cell case.

Justification of Licensee's Position

The expense for the addition of electrical insulation to the restraining channel beams and tie rods is unwarranted. Heat from an accident that can damage lead plates and vaporize electrolyte could also melt insulation on restraining channels and tie rods. In addition, rubber or plastic for insulation purposes will significantly increase the combustible fuel loading in the battery area and thus add to the fire hazard.

REGULATORY GUIDE 1.129

MAINTENANCE, TESTING AND REPLACEMENT OF LARGE LEAD STORAGE BATTERIES FOR NUCLEAR POWER PLANTS

The Licensee complies with Revision 1 of the regulatory guide as described in the Technical Specifications for the C&D batteries at Byron Station.

> The Licensee complies in general with the intent of the requirements in regulatory position C.1 of this guide. However, the Licensee performs a modified performance discharge test, as described in IEEE 450-1995. The modified test is performed in lieu of the service test and performance discharge test required by this regulatory guide because the discharge rate of the modified performance discharge test envelops the load cycle of the service test.

The Licensee complies with Revision 1 of the regulatory guide for the C&D batteries at Braidwood Station with the following comments and exceptions as described in the Technical Specifications.

> The Licensee complies in general with the intent of the requirements in regulatory position C.1 of this guide. However, the Licensee performs a modified performance discharge test, as described in IEEE Standard 450-1995. The modified test is performed in lieu of the service test and performance discharge test required by this regulatory guide, because the discharge rate of the modified performance discharge test envelops the load cycle of the service test.

REGULATORY GUIDE 1.130

SERVICE LIMITS AND LOADING COMBINATIONS FOR CLASS 1 PLATE-AND-SHELL-TYPE COMPONENT SUPPORTS

The design of Byron/Braidwood NSSS component supports is in compliance with all of the applicable regulatory positions contained in Revision 1 of this regulatory guide. Refer to Sections 3.6, 3.7, 3.9, and 3.10 for further information.

REGULATORY GUIDE 1.131

QUALIFICATION TESTS OF ELECTRIC CABLES, FIELD SPLICES AND CONNECTIONS FOR LIGHT-WATER-COOLED NUCLEAR POWER REACTORS

The Licensee complies with the regulatory position in Revision 0 with the following comments and exceptions keyed to paragraph numbers in the position:

1. Regulatory Position C-1

   The position status that in lieu of Section 1.3.4.2.3 of IEEE 383, "Other Design Basis Events", the following should be used: "The remainder of the complete spectrum of design basis events (e.g., events such as a steam line break) shall be considered in case they represent different types of more severe hazards to cable operation."

   Licensee Response: All safety-related cable is qualified for the anticipated environments detailed in Section 3.11 of the B/B-UFSAR. Steamline breaks are addressed in Subsection 3.6.1.3 of the B/B-UFSAR.

2. Regulatory Position C-10

   The position states that in lieu of the first sentence of Section 2.5.4.4.1 of IEEE 383, the following should be used: The ribbon gas burner shall be mounted horizontally such that the flame impinges on the specimen midway between the tray rungs and so that the burner face is in front of and 4 inches from the cable and approximately 2 feet above the bottom of the tray."

   Licensee Response: The ribbon gas burner was mounted so that the burner face was in front of and 3 inches from the cable, as set forth in IEEE 383, Section 2.5.4.4.1.

3. Regulatory Position C-11

   The position states that in lieu of Section 2.5.4.4.3 of IEEE 383 the following should be used: "Flame size will normally be achieved when the propane flow is 27.8 standard ft per hour and the air flow is 139 standard ft per hour."

   Licensee Response: Flame size was achieved using the schematic arrangement and pressures as set forth in IEEE 383, Section 2.5.4.4.3.

REGULATORY GUIDE 1.132

SITE INVESTIGATIONS FOR FOUNDATIONS OF NUCLEAR POWER PLANTS

I. BYRON STATION

All Byron Station site investigations were performed prior to June 1, 1978, with the exception of structure specific exploration consisting of eleven borings; (IC-1 through IC-11) performed on June 12, 13, and 14, 1978, three borings on December 14, 15, and 16, 1981, and four borings on March 17, 18, and 19, 1982, along a portion of the essential service water pipeline. The site investigations of these borings conform to the guidelines set forth in Revision 1 of the regulatory guide. The site investigations performed by the Licensee prior to the date of the regulatory guide implementation conform to the guidelines set forth in this regulatory guide because the sampling and exploration methods conform to the ASTM (American Society for Testing and Materials) procedures or other generally accepted procedures for foundation investigations at the time the work was performed. For details see Byron Sections 2.4 and 2.5.

II. BRAIDWOOD STATION

All Braidwood Station site investigation work was performed prior to June 1, 1978. However, the site investigations performed by the Licensee prior to the date of the regulatory guide implementation conform to the guidelines set forth in Revision 1 of this regulatory guide in that the investigation methods conform to the ASTM (American Society for Testing and Materials) procedures or other generally accepted procedures for foundation investigations at the time the work was performed. For details see Braidwood Sections 2.4 and 2.5.

REGULATORY GUIDE 1.133

LOOSE-PART DETECTION PROGRAM FOR THE PRIMARY SYSTEM OF LIGHT-WATER-COOLED REACTORS

The loose parts detection system is in compliance with the regulatory position in Revision 1 with the following exceptions and clarifications keyed to paragraph numbers in the regulatory position.

Section C.l.a. Sensor Location

Byron/Braidwood is in compliance with this section.

Section C.l.b System Sensitivity

The manufacturer states that preliminary tests on the system demonstrate compliance with this section.

Section C.l.c. Channel Separation

The sensors, cables, and line drivers are all physically separated from each other, but the twisted-shielded pairs running back to the control room are routed in the same cable division.

Section C.l.d. Data Acquisition System

Byron/Braidwood is in compliance with this section.

Section C.l.e. Alert Level

Byron/Braidwood is in compliance with this section.

Section C.l.f. Capability for Sensor Channel Operability Tests

There is no specific channel test for the system, but sensors on the reactor can be checked when the rods are moved.

Section C.1.g. Operability for Seismic and Environmental Conditions

The loose parts monitoring system has not been demonstrated to be capable of performing its function following all seismic events that do not require plant shutdown, up to and including the operating basis earthquake (OBE).

Section C.1.h. Quality of System Components

The components have not been demonstrated to have a 40-year design life, but the regulatory guide permits setting up a replacement schedule to replace these items. The Byron/ Braidwood design permits this.

Section C.1.i. System Repair

Byron/Braidwood is in compliance with this section.

Section C.5 Technical Specification for the Loose-Part Detection System

The requirements of the Loose-Part Detection System were relocated from Technical Specifications to the Technical Requirements Manual in Braidwood Technical Specification Amendment No. 98 and Byron Technical Specification Amendment No. 106 because the Loose-Part Detection System does not meet any of the criteria for inclusion described in 10 CFR 50.36(c)(2)(ii). The requirement to prepare and submit a special report to the Commission if an inoperable channel cannot be restored within 30 days has been revised to require the special report be submitted to the Plant Operations Review Committee.

Section C.6 Notification of a Loose Part

The requirement to notify the Commission if the presence of a loose part is confirmed is no longer applicable since the requirements of the Loose-Part Detection System were removed from Technical Specifications.

REGULATORY GUIDE 1.134

MEDICAL EVALUATION OF LICENSED PERSONNEL FOR NUCLEAR POWER PLANTS

The Licensee complies with the regulatory position in Revision 2 of this guide.

REGULATORY GUIDE 1.135

NORMAL WATER LEVEL AND DISCHARGE AT NUCLEAR POWER PLANTS

The plant design conforms to the regulatory positions in Revision 0 of this regulatory guide as described in Subsections 2.4.3, 2.4.8, and 2.4.11.

REGULATORY GUIDE 1.136

MATERIAL FOR CONCRETE CONTAINMENTS

The plant design conforms to the regulatory position in Revision 0 as described in Appendix B. Regulatory Position C.2 is not applicable since grouted tendon systems are not used. Revision 0 was the current revision of the regulatory guide when the construction permit was issued.

REGULATORY GUIDE 1.137

FUEL-OIL SYSTEMS FOR STANDBY DIESEL GENERATORS

The Licensee complies with the requirements in Revision 1 of this regulatory guide with the following exceptions:

a. For Byron, certain components or parts of the Diesel Oil System which are not available as ASME Section III items are classified as safety-related, non-ASME. This exception is taken because some original equipment vendors for fuel oil components are no longer ASME Section III suppliers. All safety-related components maintain seismic qualification and conform to Regulatory Guide 1.26 and 10 CFR Appendix B. All safety-related Diesel Oil system piping remains ASME Section III piping.

b. For Braidwood, the guidance contained in Generic Letter 89-09 is utilized to procure items, originally constructed to ASME Section III, which are no longer available as ASME Section III. The original ASME Class 3 classification for Diesel Oil System components and parts is maintained for those items originally constructed to this code class.

c. Regulatory Position C.1.e(1) and (2)

   Byron and Braidwood Stations perform pressure testing for those portions of the fuel oil system originally designed to Section III, Subsection ND of the Code in accordance with the applicable Edition and Addenda of Section XI (including code cases) as specified in 10 CFR 50.55a(g) and the Station ISI Program Plan.

d. Regulatory Position C.2

   Byron and Braidwood Station Technical Specification 5.5.13, Diesel Fuel Oil Testing Program, and Technical Requirements Manual Appendix M, Diesel Fuel Oil Testing Program establish and implement the requirements discussed in this regulatory position related to sampling and testing of new and stored fuel oil.

Refer to Technical Specification Section 5.5.13 and LCOs 3.8.1 and 3.8.3 and Technical Requirements Manual Appendix M.

REGULATORY GUIDE 1.138

LABORATORY INVESTIGATIONS OF SOILS FOR ENGINEERING ANALYSIS AND DESIGN OF NUCLEAR POWER PLANTS

I. BYRON STATION

All of the Byron Station laboratory tests on soils and rocks for determining soil and rock properties were performed prior to the regulatory guide implementation date of December 1, 1978, with the exception of laboratory tests of soil along a portion of the ESW pipeline in 1982. The laboratory tests performed in 1982 conform to the requirements of Regulatory Guide 1.138, Revision 0. The Licensee's laboratory investigations of soils and rocks prior to December 1, 1978 conform to the guidelines set forth in the regulatory guide in that the ASTM (American Society for Testing and Materials) procedures or other generally accepted procedures were used in performing the laboratory testing at the time the work was performed. For details, see Byron Subsections 2.5.4 and 2.5.5.

II. BRAIDWOOD STATION

The only laboratory tests performed since December 1, 1978 at the Braidwood Station were on soils along a portion of the essential service water pipeline and several areas of the exterior dike embankment. The essential service water pipeline is Safety Category I, the exterior dike embankment is non-safety-related.

The laboratory testing of soils by the Licensee since December 1, 1978 conforms to the requirements of Regulatory Guide 1.138, Revision 0. (No rock has been tested since December 1, 1978.) The laboratory testing of soil and rock by the Licensee prior to December 1, 1978 conforms to the guidelines set forth in Regulatory Guide 1.138 in that the ASTM (American Society for Testing and Materials) procedures or other generally accepted procedures for laboratory testing of soil and rock were used at the time the work was performed. For details see Braidwood Subsections 2.5.4, 2.5.5, and 2.5.6.

REGULATORY GUIDE 1.139

GUIDANCE FOR RESIDUAL HEAT REMOVAL

The Licensee complies with the requirements in Revision 0 of this guide, in that the Byron and Braidwood Stations are designed for safe shutdown concurrent with a single failure in one of the redundant ESF divisions. However, the Licensee defines the term safe shutdown as meaning hot standby.(1) Refer to Subsection 5.4.7 for further information.

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(1)As defined in the Technical Specifications.

REGULATORY GUIDE 1.140

DESIGN, TESTING, AND MAINTENANCE CRITERIA FOR NORMAL VENTILATION EXHAUST SYSTEM AIR FILTRATION AND ADSORPTION UNITS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS

The design of the non-safety-related filter systems meet the requirements in Revision 0 of this guide, except as noted below. Revision 0 of Regulatory Guide 1.140 was in effect when the operating license application was docketed. The exceptions are keyed to paragraph numbers in the regulatory position.

1a and 1b - The equipment and components (excluding charcoal, filter pads and separator pads) are designed to withstand a maximum of 40-year integrated radiation dose and worst-case anticipated continuous service, rather than 40 years of continuous service.

2a - All of the exhaust filter systems contain prefilters, HEPA filters, fan and associated instrumentation. Charcoal adsorbers are only used when iodine is anticipated to be present, with heaters for over 70% relative humidity air streams.

2b - The purge system exhaust filter train is designed for 43,900 cfm. Filter efficiency was tested at this capacity. The filter train consists of two banks with a grating between the filter banks. Each filter bank is three filters high and seven wide.

2f and 3f - Filter Housings

All non-ESF filter housings, exclusive of the TSC and post-LOCA purge units are at negative pressure with respect to their surroundings, and are located in areas which are low airborne radiation environments. Any in-leakage will not adversely affect Appendix I releases, hence, the housings were not leak tested to the ANSI-N509 requirements. However, all of the filter mounting frames were leak tested in accordance with ANSI N510-80. The TSC unit housing is located in an area where the airborne radiation level of the room air may exceed that of the air within the housing; however, it is at positive pressure with respect to the surroundings, hence, it was not tested to ANSI-N509 requirements. The filter mounting frames were leak tested in accordance with ANSI N510-80. The post-LOCA purge unit housing was leak tested to ANSI-N509-80 requirements.

2f and 3f - Ductwork

All of the ductwork upstream of the non-ESF filter units is under negative pressure with respect to its surroundings. Ductwork upstream of the filter units, except ductwork upstream of the TSC filter unit, is located in areas of low airborne radioactivity. Any in-leakage will not adversely affect Appendix I releases, hence, it was not tested to the ANSI-N509 requirements.

The ductwork upstream of the TSC filter unit is located in the HVAC equipment room. The quality of the equipment room environment is the same as that of the outside air which is within the duct. Any leakage will be filtered prior to its release to the TSC environment, hence, this ductwork was not tested to the ANSI-N509 requirements.

The following ductwork was leak tested to the ANSI-N509-80 requirements:

1. Positive pressure ductwork for the laboratory exhaust filter unit outside the laboratory HVAC equipment room.

2. The positive pressure ductwork from the radwaste building exhaust filter unit in the auxiliary building.

3. The positive pressure ductwork from the volume reduction system area ventilation exhaust filter that is located in the radwaste building.

4. The positive pressure ductwork from the post-LOCA purge filter unit.

5. All non-safety-related system ductwork that is required to operate and is under pressure within the control room boundary during an abnormal or accident condition.

6. TSC negative pressure duct sections outside the protected space where in-leakage would not normally be filtered.

All remaining ductwork meets the exception of ANSI N509 or has negligible impact on ALARA practices and therefore, was not leak tested. Positive pressure radwaste building exhaust ductwork in the auxiliary building was tested before the radwaste volume reduction system was put into operation. The fan peak pressure test was not performed. For systems that have no isolation devices, the fans are provided with high differential pressure trips or high/low flow trips.

3a - The components of the heaters are manufactured and assembled as per Section 5.5 of ANSI N509-76, similar to the requirements of heaters in safety-related filter systems, but the traceability of the components is not established as it would be in the case of safety-related heaters. Thus, no complete qualification program was done.

3e - Bubble tight dampers are not provided for TSC intake isolation.

5b - Airflow distribution and air-aerosol mixing tests were not performed on the non-entry type filter units. Airflow distribution tests were performed on all entry-type filter trains to ensure that the airflow through any individual filter element does not exceed 120% of the element's rated capacity.

Filtration unit airflow capacity tests were performed at the system design pressure range corresponding to clean and dirty filter pressure losses. Tests were performed at 1.25 times dirty filter conditions to verify system stability only. Filter pressure losses for airflow capacity tests were simulated without filters in place.

The miscellaneous filter tank vents system (VF)- lower flow limit criterion is no longer applied.

5c - Silicone sealant was used as a permanent sealant for HVAC ductwork.

HEPA filter bypass leakage is tested to less than 1%.

5d - Charcoal adsorber bypass leakage is tested to less than 1%.

5c and 5d - Periodic testing for 100% recirculating systems located within reactor containments will be performed per ANSI N510-1980 Table 1.

In-place bypass leakage testing will not be performed on the Containment Charcoal Filter Unit Subsystem following the replacement of charcoal or HEPA filters.

6a(3) Laboratory tests will be performed per the requirements of Table 2 of Regulatory Guide 1.140 with the exception that the temperature will be 30° C, and ASTM D3803-1989 will be used to test for the methyl iodide removal efficiency.

REGULATORY GUIDE 1.140

DESIGN, TESTING, AND MAINTENANCE CRITERIA FOR NORMAL VENTILATION EXHAUST SYSTEM AIR FILTRATION AND ADSORPTION UNITS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS

The design of the non-safety-related filter systems meet the requirements in Revision 0 of this guide, except as noted below. Revision 0 of Regulatory Guide 1.140 was in effect when the operating license application was docketed. The exceptions are keyed to paragraph numbers in the regulatory position.

1a and 1b - The equipment and components (excluding charcoal, filter pads and separator pads) are designed to withstand a maximum of 40-year integrated radiation dose and worst-case anticipated continuous service, rather than 40 years of continuous service.

2a - All of the exhaust filter systems contain prefilters, HEPA filters, fan and associated instrumentation. Charcoal adsorbers are only used when iodine is anticipated to be present, with heaters for over 70% relative humidity air streams.

2b - The purge system exhaust filter train is designed for 43,900 cfm. Filter efficiency was tested at this capacity. The filter train consists of two banks with a grating between the filter banks. Each filter bank is three filters high and seven wide.

2f and 3f - Filter Housings

All non-ESF filter housings, exclusive of the TSC unit are at negative pressure with respect to their surroundings, and are located in areas which are low airborne radiation environments. Any inleakage will not adversely affect Appendix I releases, hence, the housings were not leak tested to the ANSI-N509 requirements. However, all of the filter mounting frames were leak tested in accordance with ANSI N510-80. The TSC unit housing is located in an area where the airborne radiation level of the room air may exceed that of the air within the housing; however, it is at positive pressure with respect to the surroundings, hence, it was not tested to ANSI-N509 requirements. The filter mounting frames were leak tested in accordance with ANSI N510-80.

2f and 3f - Ductwork

Ductwork is designed, constructed, and tested in accordance with the intent of Section 5.10 of ANSI N509-976. The longitudinal seams, however, were either seal welded or mechanical lock type (Pittsburgh lock with sealants). Silicone sealant is used as a permanent sealant in HVAC ductwork.

All of the ductwork upstream of the non-ESF filter units is under negative pressure with respect to its surroundings. Ductwork upstream of the filter units, except ductwork upstream of the TSC filter unit, is located in areas of low airborne radioactivity. Any in-leakage will not adversely affect Appendix I releases, hence, it was not tested to the ANSI-N509 requirements.

The ductwork upstream of the TSC filter unit is located in the HVAC equipment room. The quality of the equipment room environment is the same as that of the outside air which is within the duct. Any leakage will be filtered prior to its release to the TSC environment, hence, this ductwork was not tested to the ANSI-N509 requirements.

The following ductwork was leak tested to the ANSI N509-80 requirements:

1. Positive pressure ductwork for the laboratory exhaust filter unit outside the laboratory HVAC equipment room.

2. The positive pressure ductwork from the radwaste building exhaust filter unit in the auxiliary building.

3. The positive pressure ductwork from the volume reduction system area ventilation exhaust filter that is located in the radwaste building.

4. The positive pressure ductwork from the post-LOCA purge filter unit.

5. TSC negative pressure duct sections outside the protected space where in-leakage would not normally be filtered.

All remaining ductwork meets the exception of ANSI N509 or has negligible impact on ALARA practices and therefore, was not leak tested. Positive pressure radwaste building exhaust ductwork in the auxiliary building was tested before the radwaste volume reduction system was put into operation.

Fan peak pressure tests were not performed. For systems that have an isolation device, the fans are provided with high differential pressure trips or high/low flow trips.

3a - The components of the heaters are manufactured and assembled as per Section 5.5 of ANSI N509-76, similar to the requirements of heaters in safety-related filter systems, but the traceability of the components is not established as it would be in the case of safety-related heaters. Thus, no complete qualification program was done.

3e - Bubble tight dampers are not provided for TSC intake isolation.

5b - Airflow distribution and air-aerosol mixing tests were not performed on the non-entry type filter units. Airflow distribution tests were performed on all entry-type filter trains to ensure that the airflow through any individual filter element does not exceed 120% of the element's rated capacity.

Filtration unit airflow capacity tests were performed at the system design pressure range corresponding to clean and dirty filter pressure losses. The midpoint filter drop test was not performed. Tests were performed at 1.25 times dirty filter conditions to verify system stability only. Filter pressure losses for airflow capacity tests were simulated without filters in place.

The miscellaneous filter tank vents system (VF), mini-purge filter unit and the post-LOCA purge filter unit (VQ) air capacity tests were performed to verify that maximum flow is not greater than 110% of design. The lower flow limit criterion was not applied.

5c - Silicone sealants or other temporary patching material was not used in the non-ESF filter housings. Silicone sealant is used, however, as a permanent sealant for HVAC ductwork.

A sampling rate of less than 1 cfm was employed for diocrylophthalate (DOP) testing filter systems larger than 1000 cfm.

HEPA filter bypass leakage is tested to less than 1%.

5d - Charcoal adsorber bypass leakage is tested to less than 1%.

5c and 5d - Periodic testing for 100% recirculating systems located within reactor containments will be performed per ANSI N510-1980 Table 1.

In-place bypass leakage testing will not be performed on the Containment Charcoal Filter Unit Subsystem following the replacement of charcoal or HEPA filters.

6a(2) All carbon furnished prior to 1985 as part of the original specification for atmospheric clean-up filtration units was tested to the requirements of Table 5-1 of ANSI N509-1976. All replacement carbon or original carbon furnished in 1985 or later will be tested to the requirements of Table 5-1 of ANSI N509-1980. With the exception that the laboratory test for methyl iodine penetration at 30°C, 95% relative humidity is less than 1%.

6a(3) Laboratory tests will be performed per the requirements of Table 2 of Regulatory Guide 1.140 with the exception that the temperature will be 30°C and ASTM D3803-1989 will be used to test for the methyl iodide removal efficiency.

REGULATORY GUIDE 1.141

CONTAINMENT ISOLATION PROVISIONS FOR FLUID SYSTEMS

The Licensee complies with the requirements in Revision 0 of this regulatory guide, as further explained below:

a. Phase A and Phase B Conditions are different from those listed. Refer to Drawings 108D685.

b. There are differences between various figures in Appendix B and containment isolation features for various systems. Refer to diagrams of the various systems. Appendix C pertains to diagram legend and symbols, to which the Licensee conforms with minor exceptions. Appendix D pertains to a valve maintenance program which the Licensee does not agree to implement. The Licensee agrees with the exceptions which the guide has taken to ANSI N271-1976.

REGULATORY GUIDE 1.142

SAFETY-RELATED CONCRETE STRUCTURES FOR NUCLEAR POWER PLANTS (OTHER THAN REACTOR VESSELS AND CONTAINMENTS)

The Licensee is in compliance with Revision 0 of Regulatory Guide 1.142, which was in effect at the time of construction, with the following clarifications:

1. Position C.7 requires compliance with ANSI Standard N45.2.5-1974, i.e., two test cylinders per 100 cubic yards of concrete tested at 28 days with a minimum of one test per day for each class of concrete. The Applicant's position is to take six test cylinders per 150 cubic yards of concrete tested in pairs at 7, 28, and 91 days with a minimum of one test per day for each class of concrete. This position is in compliance with ACI-318-71 and ACI-318-77.

   At Byron and Braidwood six standard cylinders for compressive testing were prepared from concrete samples representing every 150 cubic yards of concrete placed in Category I structures other than the containment. These specimens are tested for compressive strength at 7, 28, and 91 days.

   Concrete acceptance is based on the 91 day results; however, the 7- and 28-day results are used for monitoring the compressive strength development ages. Requirements in ACI-318 and ACI-301 are intended to cover commercial structures, in which the total number of samples is small because the total volume of concrete used is also small.

   For the large volume of concrete used in a nuclear power plant, a frequency of "every 150 cu. yd." results in a much higher confidence level and reliability than the "every 100 cu. yd." in ACI-301.

   The rate at which concrete was placed varied in a range of 50 cubic yards per hour up to 240 cubic yards per hour. This rate was governed by the size and location of the concrete element being placed and the method of placement which was used.

The onsite concrete batching plant has more production quality control and lends itself to a more consistent product than commercial concrete produced by the ready mix industry. The referenced ACI-301 and ACI-318 requirements have been designed for ready mix industry conditions.

The frequencies for testing fresh concrete (slump and air content) in ACI-301 and ACI-318 are 100 cubic yards and 150 cubic yards, respectively. For Byron and Braidwood, a frequency of every 50 cubic yards was used for testing slump, air content and temperature, as in Table B of ANSI N45.2.5-1974. In addition, the tightened sampling frequency implemented (testing of every truck) any time the properties of the fresh concrete were out of the allowable limits and the positive actions available to reject individual trucks (Table B.1-5) and to stop production (Subsection B.1.10), further reduced the probability that substandard concrete was placed.

ACI 349-76, "Code Requirements for Safety-Related Concrete Structures," establishes a compressive strength test frequency of one for every 150 cubic yards of concrete placed for safety-related structures other than the containment.

Section 4.3.1 of ACI 349-76 allows an increase in the number of cubic yards representative of a single test by 50 cubic yards for each 100 psi lower than a standard deviation of 600 psi.

Table CC-5200-1 of the Summer 1981 Addenda of the ASME Boiler and Pressure Vessel Code, Section III, Div. 2 allows a testing frequency of every 200 cubic yards if the average strength of at least the latest 30 consecutive compressive strength test exceeds the specified strength f_c' by an

amount expressed as:

$$f_{cr} = f_c' + 1.419\,(f_c' / 8.69) \qquad \text{(A1.142-1)}$$

The average compressive strength consistently exceeded this f_{cr} for all the concrete placed.

2. Position C.8 requires minimum pressure testing of embedded piping in accordance with ACI-318-71. The Licensee's position is that all Category I embedded piping is tested in accordance with ASME Section III and all Category II embedded piping is tested in accordance with ANSI B31.1.

3. Position C.9 has been complied with by the Licensee. However, the load factor for R_o used in the ACI combinations 1, 2, and 3 is different than the load factor for R_o given

in SRP Section 3.8.3. The load factor used in the UFSAR combinations is in compliance with the load factor required by the SRP.

Load combination equations 2b' and 3b' of SRP Section 3.8.4 have been complied with by the equation numbers 6 and 5, respectively of Table 3.8-10. Note 2 of the UFSAR table when applied to equation number 6 of the UFSAR reduced this equation to equation 2b' of the SRP with the exception of the load factor for dead load D. The load factor used in the UFSAR is higher than the load factor used in the SRP when the seismic load and the dead load are in the same direction. This will result in a more conservative design. If the dead load and seismic load are not in the same direction, the load factor for D is in compliance with position C.11 and ACI-349-76 Section 9.3.3.

In similar manner, using Note 2 of Table 3.8-10, equation 3b' can be reduced to equation number 5.

The NRC review and acceptance of Subsection 3.5.1.5 (Braidwood) evaluated the ductility ratios in accordance with SRP 3.5.3 and Regulatory Guide 1.142, Revision 1.

REGULATORY GUIDE 1.143

DESIGN GUIDANCE FOR RADIOACTIVE WASTE MANAGEMENT SYSTEMS, STRUCTURES AND COMPONENTS INSTALLED IN LIGHT-WATER-COOLED NUCLEAR POWER PLANTS

The Licensee complies with the requirements in Revision 0, which was in effect when the operating license application was docketed. Further information is provided in Subsection 11.2.1.11.

A review of the design of the volume reduction system shows that it conforms to Regulatory Position 1.2 with the following exceptions:

a. Those exceptions listed in Topical Report No. AECC-2-P(NP) including amendments.

b. No high level alarm has been provided for the contaminated oil tank (0VR04T) because the switches that allow the tank to be filled are mounted locally and it requires approximately 2 minutes to fill the tank.

c. No high level alarm has been provided for the flush water recovery tank (0VR09T). The high level switch starts the flush water recovery tank pump (0VR30M).

REGULATORY GUIDE 1.144

AUDITING OF QUALITY ASSURANCE PROGRAMS FOR NUCLEAR POWER PLANTS

The requirements in Regulatory Guide 1.144 have been incorporated in Regulatory Guide 1.28, Revision 3. Regulatory Guide 1.144 was withdrawn on June 17, 1991.

The Licensee complies with the intent of Regulatory Guide 1.28 Revision 3, but applies it to ANSI/ASME NQA-1-1994.

REGULATORY GUIDE 1.145

ATMOSPHERIC DISPERSION MODELS FOR POTENTIAL ACCIDENT CONSEQUENCE ASSESSMENTS AT NUCLEAR POWER PLANTS

Regulatory Guide 1.145 had not been issued at the time of the original SAR submittal. Values for χ/Q were calculated using the NRC guidance available. This guidance was provided in Section 2.3.4.2 of the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, Revision 1, issued October 1972.

Values for χ/Q were calculated using the observed onsite meteorological record, and the 5% and 50% probability values were selected and reported in the SAR. This approach was the common practice at the time, and it was reviewed and accepted by the NRC.

The NRC published Revision 0 of Regulatory Guide 1.145 in August 1979. The methodology for calculating χ/Q differs from the older methods, but results in values of χ/Q similar to those calculated earlier. The NRC does not require licensees using the older method to recalculate the χ/Q values using Regulatory Guide 1.145.

REGULATORY GUIDE 1.146

QUALIFICATION OF QUALITY ASSURANCE PROGRAM AUDIT PERSONNEL FOR NUCLEAR POWER PLANTS

The requirements in Regulatory Guide 1.146 have been incorporated in Regulatory Guide 1.28, Revision 3. Regulatory Guide 1.146 was withdrawn on June 17, 1991.

The Licensee complies with the intent of Regulatory Guide 1.28 Revision 3, but applies it to ANSI/ASME NQA-1-1994.

REGULATORY GUIDE 1.147

INSERVICE INSPECTION CODE CASE ACCEPTABILITY
ASME SECTION XI, DIVISION 1

This regulatory guide lists those Section XI ASME code cases that are generally acceptable to the NRC for implementation in the inservice inspection of light-water-cooled nuclear power plants. The Licensee complies with the regulatory position.

Code cases explain the intent of code rules or provide alternative requirements under special conditions. Implementation of individual code cases is limited to the requirements as specified in the inquiry and reply sections of each code case. The ASME considers the use of code cases to be optional for the user and not a mandatory requirement. Use of this regulatory guide is optional.

Approval of code cases listed in this regulatory guide is by code case number and date of ASME approval. Code cases to be applied during an inspection interval or preservice inspection that were previously approved for use need not be changed because a subsequent revision of the code case is listed as the approved version in subsequent revisions of this regulatory guide. Similarly, code cases to be applied during an inspection interval or preservice inspection that were previously approved for use need not be changed because the code case has been subsequently annulled. A code case that was approved for a particular situation and not for a generic application should be used only for the approved situation, because annulment of such a code case could result in situations that would not meet code requirements. New revisions to code cases must be accepted by the NRC prior to their use.

Code cases listed in this regulatory guide are generically acceptable for implementation in the inservice inspection program. Beginning with Revision 6 to this regulatory guide, it is no longer necessary to obtain NRC approval to use code cases listed in the regulatory guide. Use of such code cases should be noted in the applicable inservice inspection program plan and/or procedures.

Code cases not listed in this regulatory guide cannot be implemented in the inservice inspection program unless formal written approval of a specific relief request is obtained from the Commission or other formal approval is obtained from the Commission in accordance with footnote 6 of the Codes and Standards Rule, 10CFR 50.55a.

REGULATORY GUIDE 1.149

NUCLEAR POWER PLANT SIMULATION FACILITIES FOR USE IN OPERATOR TRAINING, LICENSE EXAMINATIONS, AND APPLICANT EXPERIENCE REQUIREMENTS

The requirements of ANSI-ANS-3.5-2009 and the clarifications to that document contained in Revision 4 of Regulatory Guide 1.149, Section C have been incorporated into the Exelon Nuclear Training procedures that cover maintenance of simulator fidelity and configuration management. The requirements and actions prescribed in these procedures have been implemented at the Byron and Braidwood simulators. Revision 4 of the regulatory guide, issued April 2011, requires that operating tests be administered on an approved or certified simulator.

REGULATORY GUIDE 1.150

ULTRASONIC TESTING OF REACTOR VESSEL WELDS DURING PRESERVICE AND INSERVICE EXAMINATIONS

On February 4, 2008, Regulatory Guide 1.150 requirements for ultrasonic testing of reactor vessel welds were superseded by 10 CFR 50.55a(g)(6)(ii)(C)(1). Ultrasonic testing of reactor vessel welds are conducted using any of the following:

1. Welds that are specified by 10 CFR 50.55a to be examined using the requirements of American Society of Mechanical Engineers (ASME) Section XI, Appendix VIII or the Appendix VIII requirements are modified by 10 CFR 50.55a or;

2. Welds specified by ASME Section XI to be examined using the requirements of Appendix VIII or are demonstrated as an acceptable alternative using the methods described in ASME Section XI, IWA-2240 (as allowed by 10 CFR 50.55a) or;

3. Welds allowed by ASME Nuclear Code Cases to be examined using the requirements of Appendix VIII or alternate methods. These code cases are implemented using approved relief requests or are approved for use in Regulatory Guide 1.147 or;

4. Welds allowed by NRC approved relief requests to be examined using the requirements of an alternative method.

REGULATORY GUIDE 1.151

INSTRUMENT SENSING LINES

Regulatory Guide 1.151 is not applicable to Byron and Braidwood stations because the construction permit application was issued before the implementation date.

REGULATORY GUIDE 1.153

CRITERIA FOR POWER, INSTRUMENTATION, AND CONTROL PORTIONS OF SAFETY SYSTEMS

Regulatory Guide 1.153 is not applicable to Byron and Braidwood stations because the construction permit application was docketed before the implementation date.

REGULATORY GUIDE 1.154

FORMAT AND CONTENT OF PLANT-SPECIFIC PRESSURIZED THERMAL SHOCK SAFETY ANALYSIS REPORTS FOR PRESSURIZED WATER REACTORS

The stated purpose of Regulatory Guide 1.154 is to provide recommended methods of assessing the risk due to pressurized thermal shock (PTS) events for proposed operation of the plant with reactor vessel RT_{PTS} above the screening criteria. Since the reactor vessels do not exceed the NRC PTS screening criteria for both design and extended license vessel life, Regulatory Guide 1.154 is not applicable.

REGULATORY GUIDE 1.155

STATION BLACKOUT

The plant design conforms to the regulatory position described in Regulatory Guide 1.155, Revision 0. This is discussed further in Subsection 8.3.1.1.2.2.

REGULATORY GUIDE 1.156

ENVIRONMENTAL QUALIFICATION OF CONNECTION ASSEMBLIES FOR NUCLEAR POWER PLANTS

Regulatory Guide 1.156 is not applicable to Byron and Braidwood stations due to the dates that the construction permit was issued and the operating license application was docketed.

REGULATORY GUIDE 1.157

BEST-ESTIMATE CALCULATIONS OF EMERGENCY CORE COOLING SYSTEM PERFORMANCE

Byron Station and Braidwood Station were subsequently licensed via License Amendment Nos. 118 and 112, respectively, to allow use of the generically approved Westinghouse Best-Estimate large break loss-of-coolant accident (LBLOCA) analysis methodology as the methodology used to perform LBLOCA analyses for Byron and Braidwood Stations. This methodology is described in WCAP-12945-P-A and was approved by the NRC in a letter from R. C. Jones, NRC, to N. J. Liparulo, Westinghouse Electric Corporation, "Acceptance for Referencing of the Topical Report WCAP-12945(P), 'Westinghouse Code Qualification Document For Best Estimate Loss of Coolant Analysis,'" dated June 28, 1996. Therefore, the Byron Station and Braidwood Station specific analysis satisfies the acceptance criteria of 10 CFR 50.46, conforms to the requirements of 10 CFR 50, Appendix K, "ECCS Evaluation Models," Section II, "Required Documentation," and meets the guidance of Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," dated May 1989. The method of analysis for large break is discussed in Section 15.6.5.2.1.2.

REGULATORY GUIDE 1.158

QUALIFICATION OF SAFETY-RELATED LEAD STORAGE BATTERIES FOR NUCLEAR POWER PLANTS

Replacement safety-related lead storage batteries purchased subsequent to February 28, 1989 are qualified in accordance with the provisions of IEEE Standard 535-1986, which is endorsed by Regulatory Guide 1.158, Revision 0.

REGULATORY GUIDE 1.159

ASSURING THE AVAILABILITY OF FUNDS FOR DECOMMISSIONING NUCLEAR REACTORS

Decommissioning costs include the cost of decontamination, dismantling, and site restoration in accordance with NRC guidelines. Illinois law requires public utility operators of nuclear power plants to establish external trusts to hold funds to cover the costs of the eventual decommissioning of nuclear power plants. The Illinois Commerce Commission has approved Exelon Generation Company's method of funding its obligations with respect to decommissioning costs and required Exelon Generation Company to contribute future decommissioning fund collections to the trusts annually.

The guidance provided in Regulatory Guide 1.159 was issued after the Licensee's plan was approved.

REGULATORY GUIDE 1.160

MONITORING THE EFFECTIVENESS OF MAINTENANCE AT NUCLEAR POWER PLANTS

The Licensee complies with the requirements in Regulatory Guide 1.160, Revision 2, through the implementation of 10 CFR 50.65. The regulatory guide endorses NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 2, as an acceptable method for implementing the maintenance rule.

The maintenance rule has been implemented using the referenced documents, with the following exceptions and clarifications:

NUMARC 93-01 recommends that industry-wide operating experience be reviewed for plant-specific applicability during the scoping process. Events that have occurred at similarly configured plants should be considered to identify non-safety related systems, structures, and components (SSC) that meet the scoping criteria. Specific events were not directly reviewed during the scoping process. Industry operating experience was considered indirectly in the scoping process via the expert panelists with operating licenses and system engineering experience. These panelists programmatically receive all pertinent operating experiences through routine routing and requalification training. The expert panel used their knowledge of this industry experience to answer the scoping screening criteria. In addition, where necessary, the expert panel considered reasonable hypothetical scenarios to determine if an SSC met the scoping criteria, even if a specific event was not identified to verify inclusion in the scope of the rule.

NUMARC 93-01 recommends monitoring maintenance preventable functional failures (MPFFs). Byron Station monitors all functional failures and does not classify certain failures as "maintenance preventable." The classification "maintenance preventable" is often subjective and could result in inconsistent evaluations for the need for (a)(1) goal setting. The use of all functional failures is more conservative than the recommendations of NUMARC 93-01 (use of MPFFs) in that all functional failures, not just those attributed to maintenance related reasons, are considered when evaluating the need for goals. Though conservative, this is considered to result in more effective implementation because it focuses on fixing performance problems rather than categorizing them. This approach is also consistent with the unavailability monitoring, which tracks unavailability due to all causes, not just those associated with maintenance activities. In addition, monitoring all functional failures simplifies the monitoring process. It removes the subjective aspect of MPFF determination, which often provides little value in selecting appropriate corrective actions. It is also consistent with the needs of the probabilistic risk assessment group, which uses plant-specific failure data attributed to all causes, not just MPFFs. During the pilot inspections, the NRC approved of this approach.

NUMARC 93-01 recommends that "The historical data used to determine the performance of SSCs consists of that data for a period of at least two fuel cycles or 36 months, whichever is less." In several cases involving the initial evaluation of SSC performance, the historical data sources for SSC availability and reliability were not amenable to the exact assessment of performance. Information needed to make a correct maintenance rule determination may not have been documented. Consequently, historical information dating from the start of cycle 6 for Unit 1 and cycle 5 for Unit 2 for Byron was used to the extent possible. Candidates for the (a)(1) category were based on this review and the recommendations by the site maintenance rule owner and senior station management.

The details of maintenance rule implementation and compliance are described in station procedures.

REGULATORY GUIDE 1.161

EVALUATION OF REACTOR PRESSURE VESSELS WITH CHARPY UPPER-SHELF ENERGY LESS THAN 50 FT-LB

Regulatory Guide 1.161 does not apply since the Charpy upper-shelf energy is predicted to remain above 50 ft-lb.

Appendix G to 10 CFR 50 requires that the predicted Charpy upper-shelf energy at end of life be above 50 ft-lb. Using the method in Regulatory Guide 1.99, Revision 2, the predicted Charpy upper-shelf energy of the weld metal at the end of life will be greater than 50 ft-lb.

REGULATORY GUIDE 1.162

FORMAT AND CONTENT OF REPORT FOR THERMAL ANNEALING OF REACTOR PRESSURE VESSELS

Regulatory Guide 1.162 does not apply since there is no intention to perform thermal annealing of the reactor pressure vessels.

REGULATORY GUIDE 1.163

PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM

The Licensee follows the guidance in Revision 0 of Regulatory Guide 1.163 as modified by approved exceptions in Technical Specification 5.5.16 for a performance-based leak-test program and leakage-rate test methods, procedures and analyses that are used to comply with the performance-based Option B in Appendix J to 10 CFR Part 50.

REGULATORY GUIDE 1.165

IDENTIFICATION AND CHARACTERIZATION OF SEISMIC SOURCES AND DETERMINATION OF SAFE SHUTDOWN EARTHQUAKE GROUND MOTION

This guide is applicable only to applications for construction permits, operating licenses, combined licenses, or design certifications submitted after January 10, 1997. The Byron/Braidwood documents were submitted prior to this date.

REGULATORY GUIDE 1.166

PREEARTHQUAKE PLANNING AND IMMEDIATE NUCLEAR POWER PLANT OPERATOR POSTEARTHQUAKE ACTIONS

This guide is applicable only to applications for construction permits, operating licenses, combined licenses, or design certifications submitted after January 10, 1997. The Byron/Braidwood documents were submitted prior to this date.

REGULATORY GUIDE 1.167

RESTART OF A NUCLEAR POWER PLANT SHUT DOWN BY A SEISMIC EVENT

This guide is applicable only to applications for construction permits, operating licenses, combined licenses, or design certifications submitted after January 10, 1997. The Byron/Braidwood documents were submitted prior to this date.

REGULATORY GUIDE 1.181

CONTENT OF THE UPDATED FINAL SAFETY ANALYSIS REPORT IN ACCORDANCE WITH 10 CFR 50.71(e)

As part of the ongoing effort to improve the quality of the UFSAR, the guidelines provided in Nuclear Energy Institute (NEI) 98-03, "Guidelines for Updating Final Safety Analysis Reports," Revision 1, June 1999, as endorsed by NRC Regulatory Guide 1.181, "Content of the Updated Final Safety Analysis Report in Accordance with 10CFR50.71(e)," Revision 0, September 1999, are used to further improve the content of the UFSAR. While the UFSAR will continue to follow the general organizational recommendations, i.e., format, specified in Regulatory Guide 1.70, Revision 2, the reorganization options described in NEI 98-03 will be used to simplify information contained in the UFSAR to improve its focus, clarity, and maintainability.

REGULATORY GUIDE 1.183

ALTERNATIVE RADIOLOGICAL SOURCE TERMS FOR EVALUATING DESIGN BASIS ACCIDENTS AT NUCLEAR POWER REACTORS

The Licensee complies with Revision 0 of the regulatory position with comments and exceptions as listed in the following UFSAR Tables:

- Table A1.183-1
- Table A1.183-2
- Table A1.183-3
- Table A1.183-4
- Table A1.183-5
- Table A1.183-6
- Table A1.183-7

TABLE A.1.183-1

CONFORMANCE WITH REGULATORY GUIDE 1.183 MAIN SECTIONS

RG Section	RG Position	Analysis	Comments
3.1	The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty. The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 or ORIGEN-ARP. Core inventory factors (Ci/MWt) provided in TID 14844 and used in some analysis computer codes were derived for low burnup, low enrichment fuel and should not be used with higher burnup and higher enrichment fuels.	Conforms	ORIGEN 2.1 based methodology was used to determine the bounding core inventory. These source terms were evaluated at end-of-cycle and at beginning of cycle (100 effective full power days (EFPD) to achieve equilibrium) conditions. The worst-case inventory was used for each of the selected 60 isotopes for the RADTRAD analyses. These values were then converted to units of Ci/MWt. Accident analyses are based on a 3658.3 MWt power level, based on the current accident analysis design basis allowance for instrument uncertainty. Source terms are based on an 18-month fuel cycle with 542.9 EFPD per cycle.
3.1	For the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the core average inventory should be used. For DBA events that do not involve the entire core, the fission product inventory of each of the damaged fuel rods is determined by dividing the total core inventory by the number of fuel rods in the core. To account for differences in power level across the core, radial peaking factors from the facility's core operating limits report (COLR) or Technical Specifications should be applied in determining the inventory of the damaged rods.	Conforms	Peaking factors of 1.7 are used for DBA events that do not involve the entire core, with fission product inventories for damaged fuel rods determined by dividing the total core inventory by the number of fuel rods in the core.
3.1	No adjustment to the fission product inventory should be made for events postulated to occur during power operations at less than full rated power or those postulated to occur at the beginning of core life. For events postulated to occur while the facility is shutdown, e.g., a fuel handling accident, radioactive decay from the time of shutdown may be modeled.	Conforms	No adjustments for less than full power are made in any analyses.

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CONFORMANCE WITH REGULATORY GUIDE 1.183 MAIN SECTIONS

RG Section	RG Position	Analysis	Comments
3.2	The core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases for DBA LOCAs are listed in Table 1 for BWRs and Table 2 for PWRs. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1. TABLE 2 PWR Core Inventory Fraction Released Into Containment Group / Gap Release Phase / Early In-Vessel Phase / Total Noble Gases 0.05 0.95 1.0 Halogens 0.05 0.35 0.4 Alkali Metals 0.05 0.25 0.3 Tellurium Metals 0.00 0.05 0.05 Ba, Sr 0.00 0.02 0.02 Noble Metals 0.00 0.0025 0.0025 Cerium Group 0.00 0.0005 0.0005 Lanthanides 0.00 0.0002 0.0002 Footnote 10: The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak rod burnup up to 62,000 MWD/MTU. The data in this section may not be applicable to cores containing mixed oxide (MOX) fuel.	Conforms	The release fractions from Regulatory Position 3.1, Table 2 are used. Footnote 10 criteria are met.
3.2	For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 3. The release fractions from Table 3 are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor. Table 3[11] Non-LOCA Fraction of Fission Product Inventory in Gap Group / Fraction I-131 0.08 Kr-85 0.10 Other Noble Gases 0.05 Other Halogens 0.05 Alkali Metals 0.12	Exception taken (as approved in a previous submittal by another Licensee)	The analysis does not fully comply with Note 11 of Table 3 since typical Byron and Braidwood core designs indicate that there are fuel assemblies that exceed the 6.3 kW/ft while >54GWD/MTU. Previous analyses (ANS 5.4) for TMI-1 have shown that those fuel assemblies exceeding these limits had no increase in gap release fractions of concern. Therefore, doubling of the

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RG Section	RG Position	Analysis	Comments
	Footnote 11: The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for rods with burnups that exceed 54 GWD/MTU. As an alternative, fission gas release calculations performed using NRC-approved methodologies may be considered on a case-by-case basis. To be acceptable, these calculations must use a projected power history that will bound the limiting projected plant-specific power history for the specific fuel load. For the BWR rod drop accident and the PWR rod ejection accident, the gap fractions are assumed to be 10% for iodines and noble gases.		gap fractions in Table 3 is conservative as used and approved in the Fort Calhoun AST submittal. Peaking factor of 1.7 used for DBA events that do not involve the entire core.
3.3	Table 4 tabulates the onset and duration of each sequential release phase for DBA LOCAs at PWRs and BWRs. The specified onset is the time following the initiation of the accident (i.e., time = 0). The early in-vessel phase immediately follows the gap release phase. The activity released from the core during each release phase should be modeled as increasing in a linear fashion over the duration of the phase. For non-LOCA DBAs, in which fuel damage is projected, the release from the fuel gap and the fuel pellet should be assumed to occur instantaneously with the onset of the projected damage. Table 4 LOCA Release Phases Phase / PWRs Onset / PWRs Duration / BWRs Onset / BWRs Duration Gap Release: 30 sec / 0.5 hr / 2 min / 0.5 hr Early In-Vessel: 0.5 hr / 1.3 hr / 0.5 hr / 1.5 hr	Conforms	The PWR durations from Table 4 are used. The LOCA activity released from the core is modeled in a linear fashion over the duration of the release phases. Non-LOCA DBAs are modeled as an instantaneous release from the fuel.
3.3	For facilities licensed with leak-before-break methodology, the onset of the gap release phase may be assumed to be 10 minutes. A licensee may propose an alternative time for the onset of the gap release phase, based on facility-specific calculations using suitable analysis codes or on an accepted topical report shown to be applicable for the	Not Applicable	Neither Byron nor Braidwood use leak-before-break methodology for design bases dose analyses.

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CONFORMANCE WITH REGULATORY GUIDE 1.183 MAIN SECTIONS

RG Section	RG Position	Analysis	Comments
	specific facility. In the absence of approved alternatives, the gap release phase onsets in Table 4 should be used.		
3.4	Table 5 lists the elements in each radionuclide group that should be considered in design basis analyses. Table 5 Radionuclide Groups Group / Elements Noble Gases: Xe, Kr Halogens: I, Br Alkali Metals: Cs, Rb Tellurium Group: Te, Sb, Se, Ba, Sr Noble Metals: Ru, Rh, Pd, Mo, Tc, Co Lanthanides: La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am Cerium: Ce, Pu, Np	Conforms	The nuclides used are the 60 identified as being potentially important dose contributors to total effective dose equivalent (TEDE) in the RADTRAD code, which encompasses those listed in RG 1.183, Table 5. The Co-58 and Co-60 values used are those from the RADTRAD defaults (activation products). All other isotope activities were determined using ORIGEN.
3.5	Of the radioiodine released from the reactor coolant system (RCS) to the containment in a postulated accident, 95 percent of the iodine released should be assumed to be cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. This includes releases from the gap and the fuel pellets. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form. The same chemical form is assumed in releases from fuel pins in FHAs and from releases from the fuel pins through the RCS in DBAs other than FHAs or LOCAs. However, the transport of these iodine species following release from the fuel may affect these assumed fractions. The accident-specific appendices to this regulatory guide provide additional details.	Conforms	This guidance was applied in the analyses. (95 percent of the iodine released should be assumed to be cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.)
3.6	The amount of fuel damage caused by non-LOCA design basis events should be analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel clad is breached. Although the NRC staff has traditionally relied upon the departure from nucleate boiling ratio (DNBR) as a fuel damage criterion, licensees may propose other methods to the NRC staff, such as those based upon enthalpy deposition, for estimating fuel damage	Conforms	The currently licensed and approved assumptions regarding the amount of fuel damage for non-LOCA design basis events is used in the AST analyses.

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RG Section	RG Position	Analysis	Comments
	for the purpose of establishing radioactivity releases.		
4.1.1	The dose calculations should determine the TEDE. TEDE is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. The calculation of these two components of the TEDE should consider all radionuclides, including progeny from the decay of parent radionuclides that are significant with regard to dose consequences and the released radioactivity.	Conforms	TEDE is calculated, with significant progeny included.
4.1.2	The exposure-to-CEDE factors for inhalation of radioactive material should be derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers" (Ref. 19). Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Ref. 20), provides tables of conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the CEDE.	Conforms	Federal Guidance Report 11 dose conversion factors (DCFs) are used.
4.1.3	For the first 8 hours, the breathing rate of persons offsite should be assumed to be 3.5 x 10^{-4} cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.8 x 10^{-4} cubic meters per second. After that and until the end of the accident, the rate should be assumed to be 2.3 x 10^{-4} cubic meters per second.	Conforms	The values that correspond to the rounded values in Section 4.1.3 of RG 1.183 are used.
4.1.4	The DDE should be calculated assuming submergence in semi-infinite cloud assumptions with appropriate credit for attenuation by body tissue. The DDE is nominally equivalent to the effective dose equivalent (EDE) from external exposure if the whole body is irradiated uniformly. Since this is a reasonable assumption for submergence exposure situations, EDE may be used in lieu of DDE in determining the contribution of external dose to the TEDE. Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil" (Ref. 21), provides external EDE conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the EDE.	Conforms	Federal Guidance Report 12 conversion factors are used.
4.1.5	The TEDE should be determined for the most limiting person at the EAB. The maximum EAB TEDE for any two-hour period following the start of the radioactivity release should be determined and used in determining	Conforms	The maximum two-hour LOCA EAB dose starts as follows:

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CONFORMANCE WITH REGULATORY GUIDE 1.183 MAIN SECTIONS

RG Section	RG Position	Analysis	Comments
	compliance with the dose criteria in 10 CFR 50.67. The maximum two-hour TEDE should be determined by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments for successive two-hour periods. The maximum TEDE obtained is submitted. The time increments should appropriately reflect the progression of the accident to capture the peak dose interval between the start of the event and the end of radioactivity release (see also Table 6). Footnote 14: With regard to the EAB TEDE, the maximum two-hour value is the basis for screening and evaluation under 10 CFR 50.59. Changes to doses outside of the two-hour window are only considered in the context of their impact on the maximum two-hour EAB TEDE.		<u>Containment Leakage</u>: 11.01 rem TEDE 0.3 to 2.3 hours <u>ECCS Leakage</u>: 1.20 rem TEDE 1.8 to 3.8 hours Conservatively, the maximum 2-hour period dose was determined by adding the maximum 2-hour dose for each of the components listed above even though they do not occur simultaneously.
4.1.6	TEDE should be determined for the most limiting receptor at the outer boundary of the low population zone (LPZ) and should be used in determining compliance with the dose criteria in 10 CFR 50.67.	Conforms	This guidance is applied in the analyses through the use of the RADTRAD computer code.
4.1.7	No correction should be made for depletion of the effluent plume by deposition on the ground.	Conforms	No such corrections are made in the analyses.
4.2.1	The TEDE analysis should consider all sources of radiation that will cause exposure to control room personnel. The applicable sources will vary from facility to facility, but typically will include: Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility, Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope, Radiation shine from the external radioactive plume released from the facility, Radiation shine from radioactive material in the reactor containment, Radiation shine from radioactive material in systems and components inside or external to the control room envelope, e.g., radioactive material buildup in recirculation filters.	Conforms	The principal source of dose within the control room is due to airborne activity within the CR. The dose contributions from the other sources, such as direct shine, were also considered.

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CONFORMANCE WITH REGULATORY GUIDE 1.183 MAIN SECTIONS

RG Section	RG Position	Analysis	Comments
4.2.2	The radioactive material releases and radiation levels used in the control room dose analysis should be determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values, unless these assumptions would result in non-conservative results for the control room.	Conforms	The source term, transport, and release methodology is the same for both the control room and offsite locations.
4.2.3	The models used to transport radioactive material into and through the control room, and the shielding models used to determine radiation dose rates from external sources, should be structured to provide suitably conservative estimates of the exposure to control room personnel.	Conforms	This guidance is applied in the analyses.
4.2.4	Credit for engineered safety features that mitigate airborne radioactive material within the control room may be assumed. Such features may include control room isolation or pressurization, or intake or recirculation filtration. Refer to Section 6.5.1, "ESF Atmospheric Cleanup System," of the SRP (Ref. 3) and Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post-accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" (Ref. 25), for guidance.	Conforms	Engineered safety features that mitigate airborne radioactive material within the control room are credited. These features are qualified and acceptable per the referenced guidance. Control Room and intake and recirculation filtration are credited. Radiation isolation mode has been analyzed with initiation within 30 minutes. After this period, credit is taken for HEPA and charcoal adsorber efficiencies.
4.2.5	Credit should generally not be taken for the use of personal protective equipment or prophylactic drugs. Deviations may be considered on a case-by-case basis.	Conforms	Such credits are not taken.
4.2.6	The dose receptor for these analyses is the hypothetical maximum exposed individual who is present in the control room for 100% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days. For the duration of the event, the breathing rate of this individual should be assumed to be 3.5×10^{-4} cubic meters per second.	Conforms	Standard occupancy factors and breathing rate are used throughout the analyses.

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CONFORMANCE WITH REGULATORY GUIDE 1.183 MAIN SECTIONS

RG Section	RG Position	Analysis	Comments
4.2.7	Control room doses should be calculated using dose conversion factors identified in Regulatory Position 4.1 above for use in offsite dose analyses. The DDE from photons may be corrected for the difference between finite cloud geometry in the control room and the semi-infinite cloud assumption used in calculating the dose conversion factors. The following expression may be used to correct the semi-infinite cloud dose, DDE_{∞}, to a finite cloud dose, DDE_{finite} , where the control room is modeled as a hemisphere that has a volume, V, in cubic feet, equivalent to that of the control room (Ref. 22). $DDE_{finite} = \frac{DDE_{\infty} V^{0.338}}{1173}$	Conforms	The equation given is utilized for finite cloud correction when calculating external doses due to the airborne activity inside the control room.
4.3	The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737 (Ref. 2). Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE. Integrated radiation exposure of plant equipment should be determined using the guidance of Appendix I of this guide.	Conforms	TSC habitability has been re-determined using AST and has been determined acceptable. The EOF is sufficiently far away from the site (outside the LPZ) such that analysis is not required.
5.1.1	The evaluations required by 10 CFR 50.67 are re-analyses of the design basis safety analyses and evaluations required by 10 CFR 50.34; they are considered to be a significant input to the evaluations required by 10 CFR 50.92 or 10 CFR 50.59. These analyses should be prepared, reviewed, and maintained in accordance with quality assurance programs that comply with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.	Conforms	These analyses were prepared as specified in the guidance. These analyses have been prepared and reviewed in accordance with a quality assurance program that complies with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.

TABLE A.1.183-1

CONFORMANCE WITH REGULATORY GUIDE 1.183 MAIN SECTIONS

RG Section	RG Position	Analysis	Comments
5.1.2	Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by Technical Specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.	Conforms	Accident mitigation features credited in these analyses are classified as safety-related, are required to be operable by Technical Specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. Single active failures and loss of offsite power were also considered where required.
5.1.3	The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but be non-conservative in another portion of the same analysis.	Conforms	Conservative assumptions are used. The effects of tolerance values were evaluated. Those values that produce the highest doses were used in the analyses.
5.1.4	Licensees should ensure that analysis assumptions and methods are compatible with the AST and the TEDE criteria.	Conforms	Analysis assumptions and methods are compatible with the AST and the TEDE criteria per this guidance.

TABLE A.1.183-1

CONFORMANCE WITH REGULATORY GUIDE 1.183 MAIN SECTIONS

RG Section	RG Position	Analysis	Comments
5.3	Atmospheric dispersion values (χ/Q) for the EAB, the LPZ, and the control room that were approved by the staff during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified by this guide. Methodologies that have been used for determining χ/Q values are documented in Regulatory Guides 1.3 and 1.4, Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and the paper, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19". References 22 *[Murphy - Campe]* and 28 *[RG 1.145]* (of RG 1.183) should be used if the FSAR χ/Q values are to be revised or if values are to be determined for new release points or receptor distances. Fumigation should be considered where applicable for the EAB and LPZ. For the EAB, the assumed fumigation period should be timed to be included in the worst 2-hour exposure period. The NRC computer code PAVAN implements Regulatory Guide 1.145 and its use is acceptable to the NRC staff. The methodology of the NRC computer code ARCON96 is generally acceptable to the NRC staff for use in determining control room χ/Q values.	Conforms	New atmospheric dispersion values (χ/Q) for the EAB, the LPZ, control room, and the TSC were developed, using meteorological data for the years 1994-1998. ARCON96 and PAVAN were used with these data to determine control room, EAB, and LPZ atmospheric dispersion values. Since there is no tall stack, no fumigation is considered. Control room χ/Qs were developed in conformance with the guidance provided in RG 1.194.

TABLE A.1.183-2

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX A (LOSS-OF-COOLANT ACCIDENT)

RG Section	RG Position	Analysis	Comments
1	Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	Conforms (with exception relating to Table 3, footnote 11)	*Fission Product Inventory:* Bounding core source terms are developed using ORIGEN-2.1 based methodology. *Release Fractions:* Release fractions are per Table 2 of RG 1.183, and are implemented by RADTRAD. Non-LOCA Table 3 release fractions are doubled in the analyses to account for the effects of exceeding the LHGR value described in footnote 11. *Timing of Release Phases:* Release Phases are per Table 4 of RG 1.183, and are implemented by RADTRAD. *Radionuclide Composition:* Radionuclide grouping is per Table 5 of RG 1.183, as implemented in RADTRAD. *Chemical Form:* Treatment of release chemical form is per RG 1.183, Section 3.5.
2	If the sump or suppression pool pH is controlled at values of 7 or greater, the chemical form of radioiodine released to the containment should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine species, including those from iodine re-evolution, for sump or suppression pool pH values less than 7 will be evaluated on a case-by-case basis. Evaluations of pH should consider the effect of acids and bases created during the LOCA event, e.g., radiolysis products. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.	Conforms	The stated distributions of iodine chemical forms are used in the analyses. The post-LOCA containment sump pH has previously been evaluated, including consideration of the effects of acids and bases created during the LOCA event, the effects of key fission product releases, and the impact of NaOH injection. Containment sump pH remains above 7 for at least 30 days.

TABLE A.1.183-2

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX A (LOSS-OF-COOLANT ACCIDENT)

RG Section	RG Position	Analysis	Comments
3.1	The radioactivity released from the fuel should be assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment in PWRs or the drywell in BWRs as it is released. This distribution should be adjusted if there are internal compartments that have limited ventilation exchange. The suppression pool free air volume may be included provided there is a mechanism to ensure mixing between the drywell to the wetwell. The release into the containment or drywell should be assumed to terminate at the end of the early in-vessel phase.	Conforms	The radioactivity release from the fuel is assumed to instantaneously and homogeneously mix throughout the containment air space as it is released. Recirculation fans provide a mixing mechanism within the containment.
3.2	Reduction in airborne radioactivity in the containment by natural deposition within the containment may be credited. Acceptable models for removal of iodine and aerosols are described in Chapter 6.5.2, "Containment Spray as a Fission Product Cleanup System," of the Standard Review Plan (SRP), NUREG-0800 (Ref. A-1) and in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments" (Ref. A-2). The latter model is incorporated into the analysis code RADTRAD (Ref. A-3).	Conforms	The RADTRAD computer program, including the Powers Natural Deposition algorithm based on NUREG/CR-6189, is used for modeling aerosol deposition in Containment. No natural deposition is assumed for elemental or organic iodine. The lower bound (10%) level of deposition credit is used.
3.3	Reduction in airborne radioactivity in the containment by containment spray systems that have been designed and are maintained in accordance with Chapter 6.5.2 of the SRP (Ref. A-1) may be credited. Acceptable models for the removal of iodine and aerosols are described in Chapter 6.5.2 of the SRP and NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays"1 (Ref. A-4). This simplified model is incorporated into the analysis code RADTRAD (Refs. A-1 to A-3). The evaluation of the containment sprays should address areas within the primary containment that are not covered by the spray drops. The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow exists between these regions, is assumed to be two turnovers of the unsprayed regions per hour, unless other rates are justified. The containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90% of the volume and if adequate mixing of unsprayed compartments can be shown.	Conforms	A qualified Containment Spray System is an available design feature at both Byron and Braidwood. The conservatively analyzed containment volume is 2.85E6 cubic feet, with 82.5% of this volume sprayed. The sprayed volume is 2.35125E6 cubic feet, unsprayed volume is 4.9875E5 cubic feet. Transfer between these two volumes is provided by the Containment Fan Coolers. The flow rate is 65,000 cfm per fan for a total of 130,000 cfm.

TABLE A.1.183-2

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX A (LOSS-OF-COOLANT ACCIDENT)

RG Section	RG Position	Analysis	Comments
	The SRP sets forth a maximum decontamination factor (DF) for elemental iodine based on the maximum iodine activity in the primary containment atmosphere when the sprays actuate, divided by the activity of iodine remaining at some time after decontamination. The SRP also states that the particulate iodine removal rate should be reduced by a factor of 10 when a DF of 50 is reached. The reduction in the removal rate is not required if the removal rate is based on the calculated time-dependent airborne aerosol mass. There is no specified maximum DF for aerosol removal by sprays. The maximum activity to be used in determining the DF is defined as the iodine activity in the columns labeled "Total" in Tables 1 and 2 of this guide multiplied by 0.05 for elemental iodine and by 0.95 for particulate iodine (i.e., aerosol treated as particulate in SRP methodology).		It is assumed that after the end of the core activity release process the aerosols would continue to be removed at a λ of 6.0 hr^{-1} until an overall DF of 50 is achieved. The current SRP 6.5.2 based assessment of elemental iodine removal coefficients during containment spray will continue to be used. The spray removal coefficient was determined to be 30.3 hr^{-1}. Per SRP 6.5.2 this value is reduced to 20 hr^{-1}. For aerosol removal the DF of 50 is reached at 2.21 hours. From that point until 8 hours, the removal coefficient of 0.6 hr^{-1} is used. For elemental iodine removal, the DF of 100 is reached at 1.926 hours.
3.4	Reduction in airborne radioactivity in the containment by in-containment recirculation filter systems may be credited if these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. A-5 and A-6). The filter media loading caused by the increased aerosol release associated with the revised source term should be addressed.	Not Applicable	Not applicable for Byron or Braidwood. In-containment recirculation filters are not credited in the analyses.
3.5	Reduction in airborne radioactivity in the containment by suppression pool scrubbing in BWRs should generally not be credited. However, the staff may consider such reduction on an individual case basis. The evaluation should consider the relative timing of the blowdown and the fission product release from the fuel, the force driving the release through the pool, and the potential for any bypass of the suppression pool (Ref. 7). Analyses should consider iodine re-evolution if the suppression pool liquid pH is not maintained greater than 7.	Not Applicable	Not applicable for a PWR

TABLE A.1.183-2

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX A (LOSS-OF-COOLANT ACCIDENT)

RG Section	RG Position	Analysis	Comments
3.6	Reduction in airborne radioactivity in the containment by retention in ice condensers, or other engineering safety features not addressed above, should be evaluated on an individual case basis. See Section 6.5.4 of the SRP (Ref. A-1).	Not Applicable	Neither Byron nor Braidwood have ice condensers. No other removal mechanisms are credited other than natural deposition.
3.7	The primary containment (i.e., drywell for Mark I and II containment designs) should be assumed to leak at the peak pressure Technical Specification leak rate for the first 24 hours. For PWRs, the leak rate may be reduced after the first 24 hours to 50% of the Technical Specification leak rate. For BWRs, leakage may be reduced after the first 24 hours, if supported by plant configuration and analyses, to a value not less than 50% of the Technical Specification leak rate. Leakage from subatmospheric containments is assumed to terminate when the containment is brought to and maintained at a subatmospheric condition as defined by Technical Specifications. For BWRs with Mark III containments, the leakage from the drywell into the primary containment should be based on the steaming rate of the heated reactor core, with no credit for core debris relocation. This leakage should be assumed during the two-hour period between the initial blowdown and termination of the fuel radioactivity release (gap and early in-vessel release phases). After two hours, the radioactivity is assumed to be uniformly distributed throughout the drywell and the primary containment.	Conforms	The analyses follow the guidance for PWRs (the analyzed leak rate may be reduced after the first 24 hours to 50% of the Technical Specification leak rate). Neither Byron nor Braidwood have subatmospheric containments.
3.8	If the primary containment is routinely purged during power operations, releases via the purge system prior to containment isolation should be analyzed and the resulting doses summed with the postulated doses from other release paths. The purge release evaluation should assume that 100% of the radionuclide inventory in the reactor coolant system liquid is released to the containment at the initiation of the LOCA. This inventory should be based on the Technical Specification reactor coolant system equilibrium activity. Iodine spikes need not be considered. If the purge system is not isolated before the onset of the gap release phase, the release fractions associated with the gap release and early in-vessel phases should be considered as applicable.	Conforms	The Byron and Braidwood containments can be considered to be routinely purged during power operation. Therefore, the resulting purge dose contribution is summed with the postulated doses from other release paths.

TABLE A.1.183-2

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX A (LOSS-OF-COOLANT ACCIDENT)

RG Section	RG Position	Analysis	Comments
4.1	Leakage from the primary containment should be considered to be collected, processed by engineered safety feature (ESF) filters, if any, and released to the environment via the secondary containment exhaust system during periods in which the secondary containment has a negative pressure as defined in Technical Specifications. Credit for an elevated release should be assumed only if the point of physical release is more than two and one-half times the height of any adjacent structure.	Conforms	No leakage is assumed to be collected for processing. Containment leakage is assumed to be released as a diffuse area source per RG 1.194. Since neither Byron nor Braidwood have a “tall stack,” elevated releases are not assumed.
4.2	Leakage from the primary containment is assumed to be released directly to the environment as a ground-level release during any period in which the secondary containment does not have a negative pressure as defined in Technical Specifications.	Conforms	For EAB and LPZ doses, ground level releases are assumed. For Control Room doses, releases are based on zero-velocity vent release assumptions (ground-level equivalent).
4.3	The effect of high wind speeds on the ability of the secondary containment to maintain a negative pressure should be evaluated on an individual case basis. The wind speed to be assumed is the 1-hour average value that is exceeded only 5% of the total number of hours in the data set. Ambient temperatures used in these assessments should be the 1-hour average value that is exceeded only 5% or 95% of the total numbers of hours in the data set, whichever is conservative for the intended use (e.g., if high temperatures are limiting, use those exceeded only 5%).	Conforms	Although Byron and Braidwood are single-containment PWRs (no secondary containments), the evaluation was performed relative to the Aux Building. The bounding 250 foot elevation wind speed exceeded only 5% of the time at Byron and Braidwood is approximately 25.2 mph. Based on representative average surface pressure coefficients for

TABLE A.1.183-2

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX A (LOSS-OF-COOLANT ACCIDENT)

RG Section	RG Position	Analysis	Comments
			rectangular buildings, a wind speed of greater than 32.2 mph would be required before the TS 3.7.12 minimum negative 0.25 inches water gauge Aux Building pressure would be positive relative to outside air pressures at any building surface.
4.4	Credit for dilution in the secondary containment may be allowed when adequate means to cause mixing can be demonstrated. Otherwise, the leakage from the primary containment should be assumed to be transported directly to exhaust systems without mixing. Credit for mixing, if found to be appropriate, should generally be limited to 50%. This evaluation should consider the magnitude of the containment leakage in relation to contiguous building volume or exhaust rate, the location of exhaust plenums relative to projected release locations, the recirculation ventilation systems, and internal walls and floors that impede stream flow between the release and the exhaust.	N/A	Byron and Braidwood are PWRs with no secondary containment.
4.5	Primary containment leakage that bypasses the secondary containment should be evaluated at the bypass leak rate incorporated in the Technical Specifications. If the bypass leakage is through water, e.g., via a filled piping run that is maintained full, credit for retention of iodine and aerosols may be considered on a case-by-case basis. Similarly, deposition of aerosol radioactivity in gas-filled lines may be considered on a case-by-case basis.	N/A	Byron and Braidwood are PWRs with no secondary containment. Therefore, all containment leakage is released directly to the environment.
4.6	Reduction in the amount of radioactive material released from the secondary containment because of ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	Conforms	Credited ESF ventilation systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02.

TABLE A.1.183-2

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX A (LOSS-OF-COOLANT ACCIDENT)

RG Section	RG Position	Analysis	Comments
5.1	With the exception of noble gases, all the fission products released from the fuel to the containment (as defined in Tables 1 and 2 of this guide) should be assumed to instantaneously and homogeneously mix in the primary containment sump water (in PWRs) or suppression pool (in BWRs) at the time of release from the core. In lieu of this deterministic approach, suitably conservative mechanistic models for the transport of airborne activity in containment to the sump water may be used. Note that many of the parameters that make spray and deposition models conservative with regard to containment airborne leakage are non-conservative with regard to the buildup of sump activity.	Conforms	With the exception of noble gases, all the fission products released from the fuel to the containment are assumed to instantaneously and homogeneously mix in the reactor building sump water at the time of release from the core.
5.2	The leakage should be taken as two times the sum of the simultaneous leakage from all components in the ESF recirculation systems above which the Technical Specifications, or licensee commitments to item III.D.1.1 of NUREG-0737 (Ref. A-8), would require declaring such systems inoperable. The leakage should be assumed to start at the earliest time the recirculation flow occurs in these systems and end at the latest time the releases from these systems are terminated. Consideration should also be given to design leakage through valves isolating ESF recirculation systems from tanks vented to atmosphere, e.g., emergency core cooling system (ECCS) pump miniflow return to the refueling water storage tank.	Conforms	ECCS leakage is analyzed at a rate twice that allowed. ECCS leakage is a minor contributor to LOCA doses from the Byron and Braidwood plants. The accident analysis basis is 276,000 cc/hour. This leak rate is considered an upper bound that would still allow ECCS operability after 30 days, without makeup (a 12% inventory loss).
5.3	With the exception of iodine, all radioactive materials in the recirculating liquid should be assumed to be retained in the liquid phase.	Conforms	With the exception of iodine, all radioactive materials in ECCS liquids are assumed to be retained in the liquid phase.

TABLE A.1.183-2

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX A (LOSS-OF-COOLANT ACCIDENT)

RG Section	RG Position	Analysis	Comments
5.4	If the temperature of the leakage exceeds 212°F, the fraction of total iodine in the liquid that becomes airborne should be assumed equal to the fraction of the leakage that flashes to vapor. This flash fraction, FF, should be determined using a constant enthalpy, h, process, based on the maximum time-dependent temperature of the sump water circulating outside the containment: $FF = \frac{h_{f1} - h_{f2}}{h_{fg}}$ *Where:* h_{f1} is the enthalpy of liquid at system design temperature and pressure; h_{f2} is the enthalpy of liquid at saturation conditions (14.7 psia, 212°F); and h_{fg} is the heat of vaporization at 212°F.	Conforms	The temperature of the leakage exceeds 212°F for a period of less than 24 hours. Therefore, a flashing factor of 10% is assumed.
5.5	If the temperature of the leakage is less than 212°F or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and area ventilation rates.	Conforms	ECCS leakage flashing fractions are assumed to be 10% for the duration of the accident.
5.6	The radioiodine that is postulated to be available for release to the environment is assumed to be 97% elemental and 3% organic. Reduction in release activity by dilution or holdup within buildings, or by ESF ventilation filtration systems, may be credited where applicable. Filter systems used in these applications should be evaluated against the guidance of Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	Conforms	The credited Control Room intake charcoal and HEPA filters meet the requirements of RG 1.52 and Generic Letter 99-02. These are credited at 95% efficiency for elemental and organic iodines. Aerosol removal efficiencies are assumed to be 99% based on the HEPA/charcoal combination.

TABLE A.1.183-2
CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX A (LOSS-OF-COOLANT ACCIDENT)

RG Section	RG Position	Analysis	Comments
			The filter efficiency for the Auxiliary Building exhaust is 90.0%. These filters meet the requirements of RG 1.52 and Generic Letter 99-02
7.0	The radiological consequences from post-LOCA primary containment purging as a combustible gas or pressure control measure should be analyzed. If the installed containment purging capabilities are maintained for purposes of severe accident management and are not credited in any design basis analysis, radiological consequences need not be evaluated. If the primary containment purging is required within 30 days of the LOCA, the results of this analysis should be combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. Reduction in the amount of radioactive material released via ESF filter systems may be taken into account provided that these systems meet the guidance in Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	Conforms	Although the UFSAR discusses containment purge for hydrogen control, this will not be considered as a release pathway during a LOCA. The system that would be used for hydrogen purge is not safety grade and thus does not meet the system operating requirements for a design basis accident.

TABLE A.1.183-3

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX B (FUEL HANDLING ACCIDENT)

RG Section	RG Position	Analysis	Comments
1	Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	Conforms	See Table A.1.183-1 for conformance with Regulatory Guide 1.183 Appendix B (Fuel Handling Accident). The bounding inventory of fission products in the reactor core and available for release to the containment is based on the maximum full power operation of the core with current licensed values for fuel enrichment, fuel burnup, and a core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty. Additional conservatisms are added as discussed in Table A.1.183-1 of this document to ensure the bounding source term was determined.
1.1	The number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case. This analysis should consider parameters such as the weight of the dropped heavy load or the weight of a dropped fuel assembly (plus any attached handling grapples), the height of the drop, and the compression, torsion, and shear stresses on the irradiated fuel rods. Damage to adjacent fuel assemblies, if applicable (e.g., events over the reactor vessel), should be considered.	Conforms	The number of fuel rods damaged is equal to one fuel assembly. As currently described in UFSAR Section 15.7.4 (Fuel Handling Accidents), the accident is defined as the drop of a spent fuel assembly (SFA) onto the spent fuel pool floor or the core, resulting in the postulated rupture of the cladding of all fuel rods in one assembly.

TABLE A.1.183-3

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX B (FUEL HANDLING ACCIDENT)

RG Section	RG Position	Analysis	Comments
1.2	The fission product release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods is assumed to be instantaneously released. Radionuclides that should be considered include xenons, kryptons, halogens, cesiums, and rubidiums.	Exception taken(alternative treatment used)	Since several fuel assemblies exceed the guidance outlined in Footnote 11, the gap release fractions are doubled for conservatism. This treatment (previously approved for Fort Calhoun is conservative as previously discussed in Table A of this Compliance Table (Item 3.2).
1.3	The chemical form of radioiodine released from the fuel to the spent fuel pool should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. The CsI released from the fuel is assumed to completely dissociate in the pool water. Because of the low pH of the pool water, the iodine re-evolves as elemental iodine. This is assumed to occur instantaneously. The NRC staff will consider, on a case-by-case basis, justifiable mechanistic treatment of the iodine release from the pool.	Conforms	All iodine added to the reactor vessel or spent fuel pool is assumed to instantaneously dissociate and re-evolve as elemental iodine and treated appropriately with regard to pool pH and is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide.
2	If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 57% elemental and 43% organic species. If the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-by-case method (Ref. B-1)	Conforms	The analyzed water depth above damaged fuel is 23 feet. This value corresponds to the minimum depth of water coverage over the top of irradiated fuel assemblies seated in the spent fuel pool racks within the spent fuel pool, as per TS 3.7.14.

TABLE A.1.183-3

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX B (FUEL HANDLING ACCIDENT)

RG Section	RG Position	Analysis	Comments
			Therefore, an overall DF of 200 is used per this guidance. Iodine above the water is assumed to be composed of 57% elemental and 43% organic species
3	The retention of noble gases in the water in the fuel pool or reactor cavity is negligible (i.e., decontamination factor of 1). Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor).	Conforms	DF = 1 for noble gas isotopes; DF = infinite for particulate radionuclides.
4.1	The radioactive material that escapes from the fuel pool to the fuel building is assumed to be released to the environment over a 2-hour time period.	Conforms	The release is assumed to occur over a two-hour period.
4.2	A reduction in the amount of radioactive material released from the fuel pool by engineered safety feature (ESF) filter systems may be taken into account provided these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-2, B-3). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses.	Conforms	All ESF filtration systems credited in the analyses are qualified in accordance with the references cited in this section.
4.3	The radioactivity release from the fuel pool should be assumed to be drawn into the ESF filtration system without mixing or dilution in the fuel building. If mixing can be demonstrated, credit for mixing and dilution may be considered on a case-by-case basis. This evaluation should consider the magnitude of the building volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the pool, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the pool and the exhaust plenums.	Conforms	As per RG 1.183, the release from the fuel building to the environment is assumed over a 2-hour time period. To assure this, the refueling floor exhaust rate is set artificially high at 5 times this value or 0.118 air changes per minute during Control Room emergency mode of operation.

TABLE A.1.183-3

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX B (FUEL HANDLING ACCIDENT)

RG Section	RG Position	Analysis	Comments
5.1	If the containment is isolated during fuel handling operations, no radiological consequences need to be analyzed.	Not Applicable	Containment isolation is not credited in the analysis.
5.2	If the containment is open during fuel handling operations, but designed to automatically isolate in the event of a fuel handling accident, the release duration should be based on delays in radiation detection and completion of containment isolation. If it can be shown that containment isolation occurs before radioactivity is released to the environment, no radiological consequences need to be analyzed.	Conforms	Automatic containment isolation is not credited. Therefore, a radiological consequence analysis is performed.
5.3	If the containment is open during fuel handling operations (e.g., personnel air lock or equipment hatch is open), the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period. Note 3: *The staff will generally require that Technical Specifications allowing such operations include administrative controls to close the airlock, hatch, or penetrations within 30 minutes. Such administrative controls will generally require that a dedicated individual be present, with the necessary equipment available, to restore containment closure should a fuel handling accident occur. Radiological analyses should generally not credit this manual isolation.*	Conforms (with site-specific exceptions as noted in Attachment 5 of the submittal)	The radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period.
5.4	A reduction in the amount of radioactive material released from the containment by ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-2 and B-3). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses.	Conforms	For non-Recently Irradiated Fuel, no filtration of the radioactive gas released from the pool or automatic isolation of the accident location is assumed, with essentially all of the activity reaching the refueling floor airspace exhausted to the environment within two hours after the accident.

TABLE A.1.183-3

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX B (FUEL HANDLING ACCIDENT)

RG Section	RG Position	Analysis	Comments
			For Recently Irradiated Fuel, an additional FHA analysis was performed with containment closure established or with the FHB ventilation system operable. The results of this analysis also met the limits of 10 CFR 50.67 assuming a minimum decay time of six hours. The six-hour minimum decay time is inconsequential as it is physically impossible to remove the reactor head and move fuel within the first six hours after the reactor is subcritical.
5.5	Credit for dilution or mixing of the activity released from the reactor cavity by natural or forced convection inside the containment may be considered on a case-by-case basis. Such credit is generally limited to 50% of the containment free volume. This evaluation should consider the magnitude of the containment volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the reactor cavity, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the reactor cavity and the exhaust plenums.	Conforms	The activity is instantaneously released from the fuel into the containment and is assumed to mix with 100% of the containment volume to calculate a hypothetical release rate with which to remove nearly all the activity within a two-hour period. This creates a conservative release rate over the two-hour release period.

TABLE A.1.183-4

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX H (PWR ROD EJECTION ACCIDENT)

RG Section	RG Position	Analysis	Comments
1	Assumptions acceptable to the NRC staff regarding core inventory are in Regulatory Position 3 of this guide. For the rod ejection accident, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10% of the core inventory of the noble gases and iodines is in the fuel gap. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and the assumption that 100% of the noble gases and 25% of the iodines contained in that fraction are available for release from containment. For the secondary system release pathway, 100% of the noble gases and 50% of the iodines in that fraction are released to the reactor coolant.	Conforms	The CREA core source terms are those associated with a DBA power level of 3658.3 MWth, which includes an additional 2% power over that of the full licensed power to account for uncertainty. The sudden rod ejection and localized temperature spike associated with the CREA results in the damage of 10% of the core. Only 2.5 % of the damaged core releases melted fuel activity, i.e., 0.00250 of the total core melts. Therefore, the source term available for release is associated with this fraction of melted fuel and the fraction of core activity existing in the gap. A peaking factor of 1.7 is also applied.
2	If no fuel damage is postulated for the limiting event, a radiological analysis is not required as the consequences of this event are bounded by the consequences projected for the loss-of-coolant accident (LOCA), main steam line break, and steam generator tube rupture.	Not Applicable	Since fuel damage is postulated, a radiological consequence analysis is performed.

TABLE A.1.183-4

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX H (PWR ROD EJECTION ACCIDENT)

RG Section	RG Position	Analysis	Comments
3	Two release cases are to be considered. In the first, 100% of the activity released from the fuel should be assumed to be released instantaneously and homogeneously through the containment atmosphere. In the second, 100% of the activity released from the fuel should be assumed to be completely dissolved in the primary coolant and available for release to the secondary system.	Conforms	For Case 1, the ejected control rod is assumed to breach the reactor pressure vessel (RPV), effectively causing the equivalent of a small break loss of coolant accident. In this case, all activity from damaged fuel that has been mixed with the primary coolant of the Reactor Coolant System (RCS) leaks directly to the containment volume. This flashed release is assumed to instantaneously and homogeneously mix with the containment atmosphere, and is available for release to the environment via a Containment leak rate limit, or L_a. For Case 2, no breach of the RPV is assumed following the rod ejection. In this case, RCS integrity is maintained and all activity from damaged fuel that has been mixed with the RCS leaks to the secondary side coolant through the Steam Generator (SG) tubes via the Tech. Spec. primary to secondary coolant leakage rate of 1.0 gpm.

TABLE A.1.183-4

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX H (PWR ROD EJECTION ACCIDENT)

RG Section	RG Position	Analysis	Comments
			From here, activity is available for release to the environment by steaming of the SG Power-Operated Relief Valves (PORVs). In addition to the activity released from the primary to secondary coolant, pre-existing Tech. Spec. iodine activity in the secondary coolant system is assumed to also be released.
4	The chemical form of radioiodine released to the containment atmosphere should be assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. If containment sprays do not actuate or are terminated prior to accumulating sump water, or if the containment sump pH is not controlled at values of 7 or greater, the iodine species should be evaluated on an individual case basis. Evaluations of pH should consider the effect of acids created during the rod ejection accident event, e.g., pyrolysis and radiolysis products. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.	Conforms	All iodine released from the SGs is conservatively assumed to be of the elemental species. This is done for RADTRAD simulation considerations, and is consistent with the RG 1.183, because elemental and organic iodine are identically treated by the computer model.
5	Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.	Conforms	All iodine released from the SGs is assumed to be of the elemental species. This is done for RADTRAD simulation considerations, and is consistent with the RG 1.183 specification of 97% elemental and 3% organic, because elemental and organic iodine are identically treated by the computer model.
6	Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material in and from the containment are as follows.	Conforms	(See sections 6.1 and 6.2)

TABLE A.1.183-4

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX H (PWR ROD EJECTION ACCIDENT)

RG Section	RG Position	Analysis	Comments
6.1	A reduction in the amount of radioactive material available for leakage from the containment that is due to natural deposition, containment sprays, recirculating filter systems, dual containments, or other engineered safety features may be taken into account. Refer to Appendix A to this guide for guidance on acceptable methods and assumptions for evaluating these mechanisms.	Conforms	The RADTRAD computer program, including the Powers Natural Deposition algorithm based on NUREG/CR-6189, is used for modeling aerosol deposition in Containment. No natural deposition is assumed for elemental or organic iodine. The lower bound (10%) level of deposition credit is used. Decay of radioactivity is credited in all compartments, prior to release. This is implemented in RADTRAD using the half-lives in the Nuclide Inventory File (NIF). The RADTRAD decay plus daughter option is used. In reality, daughter products such as xenon from iodines or iodines from tellurium are unlikely to readily escape from the fuel matrix in which the parent iodine or tellurium is contained. Nevertheless, the RADTRAD feature to include daughter effects is selected for conservatism. No credit for containment spray is taken.

TABLE A.1.183-4

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX H (PWR ROD EJECTION ACCIDENT)

RG Section	RG Position	Analysis	Comments
6.2	The containment should be assumed to leak at the leak rate incorporated in the Technical Specifications at peak accident pressure for the first 24 hours, and at 50% of this leak rate for the remaining duration of the accident. Peak accident pressure is the maximum pressure defined in the Technical Specifications for containment leak testing. Leakage from subatmospheric containments is assumed to be terminated when the containment is brought to a subatmospheric condition as defined in Technical Specifications.	Conforms	The containment is assumed to leak at the leak rate incorporated in the Technical Specifications at peak accident pressure for the first 24 hours, and at 50% of this leak rate for the remaining duration of the accident.
7.1	A leak rate equivalent to the primary-to-secondary leak rate Limiting Condition for Operation specified in the Technical Specifications should be assumed to exist until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms	The leak rate equivalent to the primary-to-secondary leak rate Limiting Condition for Operation specified in the Technical Specifications is assumed to exist until shutdown cooling is in operation and releases from the steam generators have been terminated.
7.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate Technical Specifications. These tests typically are based on cooled liquid. The facility's instrumentation used to determine leakage typically is located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft3).	Conforms	The density is assumed to be 1.0 gm/cc (62.4 lbm/ft3)
7.3	All noble gas radionuclides released to the secondary system are assumed to be released to the environment without reduction or mitigation.	Conforms	Noble gases are released without reduction or mitigation.
7.4	The transport model described in assumptions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.	Conforms	The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E was utilized for iodine and particulates.

TABLE A.1.183-5

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX G (PWR LOCKED ROTOR ACCIDENT)

RG Section	RG Position	Analysis	Comments
1	Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are in Regulatory Position 3 of this regulatory guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.	A conservative exception is taken regarding release fractions	The analysis does not fully comply with Note 11 of Table 3 since typical Byron and Braidwood core designs indicate that there are fuel assemblies that exceed the 6.3 kW/ft while >54GWD/MTU. Previous analyses (ANS 5.4) for TMI-1 have shown that those fuel assemblies exceeding these limits had no increase in gap release fractions of concern. Therefore, doubling of the “Other Noble Gases’, “Other Halogens”, and “Alkali Metals” gap fractions in Table 3 is conservative as used and approved in the Fort Calhoun AST submittal. Additionally, a peaking factor of 1.7 is used for DBA events that do not involve the entire core.
2	If no fuel damage is postulated for the limiting event, a radiological analysis is not required as the consequences of this event are bounded by the consequences projected for the main steam line break outside containment.	Not Applicable	Fuel damage is assumed. Therefore, a specific analysis is performed.
3	The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.	Conforms	The activity is assumed to be released instantaneously and homogeneously through the primary coolant.

TABLE A.1.183-5

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX G (PWR LOCKED ROTOR ACCIDENT)

RG Section	RG Position	Analysis	Comments
4	The chemical form of radioiodine released from the fuel should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking.	Conforms	Iodine chemical form is in accordance with this guidance (97% elemental, 3% organic iodines).
5.1	The primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate Limiting Condition for Operation specified in the Technical Specifications. The leakage should be apportioned between the steam generators in such a manner that the calculated dose is maximized.	Conforms	Neither Byron nor Braidwood have implemented alternative repair criteria. Therefore, the primary-to-secondary leak rate in the steam generators is assumed to be the leak rate Limiting Condition for Operation specified in the Technical Specifications. The design basis leak rate is 0.218 gpm per intact SG, totaling 0.654 gpm.
5.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate Technical Specifications. These tests are typically based on cool liquid. Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft3).	Conforms	The density is assumed to be 1.0 gm/cc (62.4 lbm/ft3)

TABLE A.1.183-5

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX G (PWR LOCKED ROTOR ACCIDENT)

RG Section	RG Position	Analysis	Comments
5.3	The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212° F). The release of radioactivity should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms	The steaming release and primary-to-secondary coolant leakage is postulated to end at 40 hours, when the RCS and secondary loop have equilibrated.
5.4	The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power.	Conforms	A coincident loss of offsite power is assumed.
5.5	All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Conforms	Noble gases are released without reduction or mitigation.
5.6	The transport model described in assumptions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.	Conforms	The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E was utilized for iodine and particulates.

TABLE A.1.183-6

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX E (PWR MAIN STEAM LINE BREAK)

RG Section	RG Position	Analysis	Comments
1	Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this regulatory guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. The fuel damage estimate should assume that the highest worth control rod is stuck at its fully withdrawn position.	Conforms	No fuel damage is postulated to occur during the MSLB (see Section 2 below).
2	If no or minimal[2] fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by the Technical Specifications. Two cases of iodine spiking should be assumed. Footnote 2: The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum Technical Specification values, whichever maximizes the radiological consequences. In determining dose equivalent I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.	Conforms	The activity assumed in the analysis is based on the activity associated with the maximum Technical Specification values. In determining dose equivalent I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes is included.
2.1	A reactor transient has occurred prior to the postulated main steam line break (MSLB) and has raised the primary coolant iodine concentration to the maximum value (typically 60 μCi/gm DE I-131) permitted by the Technical Specifications (i.e., a pre-accident iodine spike case).	Conforms	This analyzed case involves a 60 μCi/gm pre-accident Iodine spike, consistent with the Technical Specification operational Reactor Coolant System (RCS) activity concentration limit for an assumed spike. All of the spike activity is homogeneously mixed in the primary coolant, prior to accident initiation.

TABLE A.1.183-6

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX E (PWR MAIN STEAM LINE BREAK)

RG Section	RG Position	Analysis	Comments
2.2	The primary system transient associated with the MSLB causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (typically 1.0 µCi/gm DE I-131) specified in Technical Specifications (i.e., concurrent iodine spike case). A concurrent iodine spike need not be considered if fuel damage is postulated. The assumed iodine spike duration should be 8 hours. Shorter spike durations may be considered on a case-by-case basis if it can be shown that the activity released by the 8-hour spike exceeds that available for release from the fuel gap of all fuel pins.	Conforms (Iodine spike has been determined to last for 6 hours instead of 8)	This case involves an accident initiated iodine spike that occurs concurrently with the release of fluid from the primary and secondary coolant systems. This spike results in a release rate from the operating limit defective fuel fraction that is 500 times the normal rate. Conservative Byron and Braidwood analyses have shown that after 6 hours the total iodine gap activity of the defective fuel will have been completely released into the primary coolant.
3	The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.	Conforms	The released activity is assumed to be dispersed instantaneously and homogeneously through the primary coolant.
4	The chemical form of radioiodine released from the fuel should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking.	Conforms	Iodine chemical form is in accordance with this guidance (i.e., 97% elemental and 3% organic).

TABLE A.1.183-6

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX E (PWR MAIN STEAM LINE BREAK)

RG Section	RG Position	Analysis	Comments
5.1	For facilities that have not implemented alternative repair criteria (see Ref. E-1, DG-1074), the primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate Limiting Condition for Operation specified in the Technical Specifications. For facilities with traditional generator specifications (both per generator and total of all generators), the leakage should be apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized.	Conforms	Neither Byron nor Braidwood have implemented alternative repair criteria. Therefore, the primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate Limiting Condition for Operation specified in the Technical Specifications. Activity that originates in the primary RCS is released to the secondary coolant by means of the primary-to-secondary coolant leak rate. This design basis leak rate value is 0.218 gpm, per intact SG, totaling 0.654 gpm, and 0.5 gpm for the faulted SG with the broken steam line.
5.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of the parameter being converted. The ARC leak rate correlations are generally based on the collection of cooled liquid. Surveillance tests and facility instrumentation used to show compliance with leak rate Technical Specifications are typically based on cooled liquid. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft3).	Conforms	The density is assumed to be 1.0 gm/cc (62.4 lbm/ft3).
5.3	The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212°F). The release of radioactivity from unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms	The steaming release and primary-to-secondary coolant leakage is postulated to end at 40 hours, when the RCS and secondary loop have equilibrated.

TABLE A.1.183-6

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX E (PWR MAIN STEAM LINE BREAK)

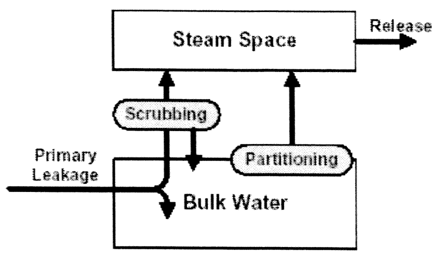
RG Section	RG Position	Analysis	Comments
5.4	All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Conforms	Noble gases are released without reduction or mitigation.
5.5	The transport model described in this section should be utilized for iodine and particulate releases from the steam generators. This model is shown in Figure E-1 and summarized below: **Figure E-1** **Transport Model** 	Conforms	The transport model described in this section is utilized for iodine and particulate releases from the steam generators.
5.5.1	A portion of the primary-to-secondary leakage will flash to vapor, based on the thermodynamic conditions in the reactor and secondary coolant. During periods of steam generator dryout, all of the primary-to-secondary leakage is assumed to flash to vapor and be released to the environment with no mitigation. With regard to the unaffected steam generators used for plant cooldown, the primary-to-secondary leakage can be assumed to mix with the secondary water without flashing during periods of total tube submergence.		Primary to secondary coolant leakage through the faulted steam generator conservatively goes directly to the environment, without mixing with any secondary coolant. Therefore, under the assumed dry-out conditions, no partitioning of any nuclides is expected to occur in this release pathway. For all post-accident releases through the

TABLE A.1.183-6

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX E (PWR MAIN STEAM LINE BREAK)

RG Section	RG Position	Analysis	Comments
			PORVs of the intact SG loops, the mechanism for release to the environment is steaming of the secondary coolant. Because of this release dynamic, a reduction is taken in the amount of activity released to the environment based on partitioning of nuclides between the liquid and gas states of water. For Iodine, the partitioning factor of 0.01 was taken directly from RG 1.183. Reviewing the specified AST release fractions, it is concluded that the only nuclides other than iodines to be released from the core source term are noble gas nuclides. Because of the volatility of noble gases, no partitioning is assumed for any such isotopes.
5.5.2	The leakage that immediately flashes to vapor will rise through the bulk water of the steam generator and enter the steam space. Credit may be taken for scrubbing in the generator, using the models in NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident" (Ref. E-2), during periods of total submergence of the tubes.	Conforms	See Comments for Section 5.5.1.
5.5.3	The leakage that does not immediately flash is assumed to mix with the bulk water.	Conforms	See Comments for Section 5.5.1.

TABLE A.1.183-6

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX E (PWR MAIN STEAM LINE BREAK)

RG Section	RG Position	Analysis	Comments
5.5.4	The radioactivity in the bulk water is assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient. A partition coefficient for iodine of 100 may be assumed. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators.	Conforms	The specified partition coefficient is used in the analysis.
5.6	Operating experience and analyses have shown that for some steam generator designs, tube uncovery may occur for a short period following any reactor trip (Ref. E-3). The potential impact of tube uncovery on the transport model parameters (e.g., flash fraction, scrubbing credit) needs to be considered. The impact of emergency operating procedure restoration strategies on steam generator water levels should be evaluated.	Conforms	See Comments for Section 5.5.1.

TABLE A.1.183-7

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX F (PWR STEAM GENERATOR TUBE RUPTURE ACCIDENT)

RG Section	RG Position	Analysis	Comments
1	Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are in Regulatory Position 3 of this guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.	Conforms	See Table A.1.183-1 for conformance with Regulatory Guide 1.183 Appendix F. The bounding inventory of fission products in the reactor core and available for release to the containment is based on the maximum full power operation of the core with current licensed values for fuel enrichment, fuel burnup, and a core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty. Additional conservatisms are added as discussed in Table A.1.183-1 of this document to ensure the bounding source term was determined.
2	If no or minimal[2] fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by Technical Specification. Two cases of iodine spiking should be assumed. Footnote #2: The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum Technical Specification values, whichever maximizes the radiological consequences. In determining dose equivalent I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.	Conforms	The design basis assumes no fuel damage for the postulated SGTR event. For this SGTR accident, the source terms are defined by the Technical Specification activity release rates from a maximum failed fuel fraction assumed during operation, which are characterized by the equilibrium 1.0 μCi/gm Dose Equivalent (DE) I-131 iodine activity concentration in the primary reactor coolant system. The noble gas inventory in the RCS is based on operation at the Technical Specification limit of 603 micro-Ci/gm DE Xe-133.

TABLE A.1.183-7

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX F (PWR STEAM GENERATOR TUBE RUPTURE ACCIDENT)

RG Section	RG Position	Analysis	Comments
			Because no fuel damage is assumed for this accident, only iodine and noble gas isotopes are modeled to contribute to dose. To identify the worst-case SGTR accident, however, two different cases of iodine spiking are analyzed, per regulatory guidance (Pre-Accident Iodine Spike and Concurrent Iodine Spike).
2.1	A reactor transient has occurred prior to the postulated steam generator tube rupture (SGTR) and has raised the primary coolant iodine concentration to the maximum value (typically 60 µCi/gm DE I-131) permitted by the Technical Specifications (i.e., a pre-accident iodine spike case).	Conforms	This analyzed case involves a 60 µCi/gm pre-accident Iodine spike, consistent with the Technical Specification operational Reactor Coolant System (RCS) activity concentration limit for an assumed spike. All of the spike activity is homogeneously mixed in the primary coolant, prior to accident initiation.
2.2	The primary system transient associated with the SGTR causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 335 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (typically 1.0 µCi/gm DE I-131) specified in Technical Specifications (i.e., concurrent iodine spike case). A concurrent iodine spike need not be considered if fuel damage is postulated. The assumed iodine spike duration should be 8 hours. Shorter spike durations may be considered on a case-by-case basis if it can be shown that the activity released by the 8-hour spike exceeds that available for release from the fuel gap of all fuel pins.	Conforms	The second analyzed case involves an accident initiated iodine spike that occurs concurrently with the release of fluid from the primary and secondary coolant systems. This spike results in a release rate from defective fuel that is 335 times the normal rate, and lasts for an 8-hour duration.

TABLE A.1.183-7

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX F (PWR STEAM GENERATOR TUBE RUPTURE ACCIDENT)

RG Section	RG Position	Analysis	Comments
3	The activity released from the fuel, if any, should be assumed to be released instantaneously and homogeneously through the primary coolant.	Conforms	Mixing in the primary coolant is assumed to be instantly and homogeneously.
4	Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.	Conforms	Such iodine releases are assumed to be 97% elemental and 3% organic.
5.1	The primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate Limiting Condition for Operation specified in the Technical Specifications. The leakage should be apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized.	Conforms	Activity that originates in the primary RCS is released to the secondary coolant by means of the primary-to-secondary coolant leak rate. This design basis leak rate value is 0.218 gpm per intact SG, totaling 0.654 gpm.
5.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate Technical Specifications. These tests are typically based on cool liquid. Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft3).	Conforms	The density is assumed to be 1.0 gm/cc (62.4 lbm/ft3)
5.3	The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212° F). The release of radioactivity from the unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms	Release of activity terminates when shutdown cooling has been established.
5.4	The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power.	Conforms	A coincident loss of offsite power is assumed.
5.5	All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Conforms	Noble gases are released without reduction or mitigation.
5.6	The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.	Conforms	The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E was utilized for iodine and particulates.

Regulatory Guide 1.194

Atmospheric Relative Concentrations For Control Room Radiological Habitability Assessments At Nuclear Power Plants

The Licensee complies with Revision 0 of the regulatory position with comments and exceptions as listed in UFSAR Table A1.194-1.

TABLE A.1.194-1

CONFORMANCE WITH REGULATORY GUIDE 1.194 (DIFFUSE AREA SOURCE GUIDANCE)

RG Section	RG Position	Analysis	Comments
3.2.4	Examples of possible area sources are postulated releases from the surface of a reactor or a secondary containment building. A reasonable approach (is) to model the building surface as a vertical planar area source. This approach is not intended to address dispersion resulting from building-induced turbulence. Treatment of a release as a diffuse source will be acceptable for design basis calculations if the guidance herein is followed.	Conforms	Introductory information excerpted - no requirements.
3.2.4.1	Diffuse source modeling should be used only for those situations in which the activity being released is homogeneously distributed throughout the building and when the assumed release rate from the building surface would be reasonably constant over the surface of the building. For example, steam releases within a turbine building with roof ventilators or louvered walls would generally not be suitable for modeling as a diffuse source. (See Regulatory Positions 3.2.4.7 and 3.2.4.8.)	Conforms	Used only for containment building, where the situation would comply with this guidance.
3.2.4.2	Since leakage is more likely to occur at a penetration, analysts must consider the potential impact of building penetrations exposed to the environment* within this modeled area. If the penetration release would be more limiting, the diffuse area source model should not be used. Releases from personnel air locks and equipment hatches exposed to the environment, or containment purge releases prior to containment isolation, may need to be treated differently. It may be necessary to consider several cases to ensure that the χ/Q value for the most limiting location is identified. *Penetrations that are enclosed within safety-related structures need not be considered in this evaluation if the release would be captured and released via a plant ventilation system, as ventilation system releases should have already been addressed as a separate release point.	Conforms	Containment radioactivity releases through penetrations and personnel/equipment hatches into the auxiliary building are served by otherwise un-credited HEPA filters and charcoal adsorbers before release through the plant vent. This filtration more than offsets differences between plant vent and containment diffuse area source χ/Qs. Therefore, unfiltered containment diffuse area treatment can be conservatively applied to these penetrations. Leakage through the secondary personnel/equipment hatch would be unfiltered, but the hatch is located on the far

TABLE A.1.194-1

CONFORMANCE WITH REGULATORY GUIDE 1.194 (DIFFUSE AREA SOURCE GUIDANCE)

RG Section	RG Position	Analysis	Comments
			side of the containment buildings with respect to the Control Room intakes, with χ/Qs more favorable than the containment building diffuse source χ/Qs. Containment vent penetrations are exhausted through the plant vent, but such leakage, if not directly into the auxiliary building, would be past miniflow or normal ventilation HEPA filters, or the post-LOCA purge HEPA filters and charcoal adsorbers. Therefore, unfiltered containment diffuse area treatment can be conservatively applied to these penetrations. Containment purge supply penetration leakage, if not directly into the auxiliary building, would be to the auxiliary building supply air intake such that it would be drawn back into the auxiliary building. Therefore, unfiltered containment diffuse area treatment can be conservatively applied to these penetrations.

TABLE A.1.194-1

CONFORMANCE WITH REGULATORY GUIDE 1.194 (DIFFUSE AREA SOURCE GUIDANCE)

RG Section	RG Position	Analysis	Comments
			Penetration leakage into the steam tunnel is not from the containment atmosphere, due to the barrier provided by the steam generators.
3.2.4.3	The total release rate (e.g., Ci/second) from the building atmosphere is to be used in conjunction with the diffuse area source χ/Q in assessments. This release rate is assumed to be equally distributed over the entire diffuse source area from which the radioactivity release can enter the environment. For freestanding containments, this would be the entire periphery above grade or above a building that surrounds the lower elevations of the containment. When a licensee can justify assuming collection of a portion of the release from the containment within the surrounding building, the total release from the containment may be apportioned between the exposed and enclosed building surfaces. Similarly, if the building atmosphere release is modeled through more than one simultaneous pathway (e.g., drywell leakage and main steam safety valve leakage in a BWR), only that portion of the total release released through the building surface should be used with the diffuse area χ/Q. The release rate should not be averaged or otherwise apportioned over the surface area of the building. For example, reducing the release rate by 50 percent because only 50 percent of the surface faces the control room intake would be inappropriate.	Conforms	The containment buildings are freestanding containments, so the source area is the entire periphery above grade.
3.2.4.4	ARCON96 uses two initial diffusion coefficients entered by the user to represent the area source. There are insufficient field measurements to mechanistically model these initial diffusion coefficients. The following deterministic equations should be used in the absence of site-specific empirical data.*	Conforms	ARCON96 and the two equations are utilized.

TABLE A.1.194-1

CONFORMANCE WITH REGULATORY GUIDE 1.194 (DIFFUSE AREA SOURCE GUIDANCE)

RG Section	RG Position	Analysis	Comments
	$\text{Sigma } Y_o = \dfrac{\text{Width source area}}{6}$ $\text{Sigma } Z_o = \dfrac{\text{Height source area}}{6}$ *See Regulatory Position 7 regarding the use of site-specific empirical measurements.		
3.2.4.5	The height and width of the area source (e.g., the building surface) are taken as the maximum vertical and horizontal dimensions of the above-grade building cross-sectional area perpendicular to the line of sight from the building center to the control room intake (see Figure 2). These dimensions are projected onto a vertical plane perpendicular to the line of sight and located at the closest point on the building surface to the control room intake. The release height is set at the vertical center of the projected plane. The source-to-receptor distance (slant path) is measured from this point to the control room intake.	Conforms	
3.2.4.6	Intentional releases from a secondary containment (e.g., standby gas treatment systems (SGTS) at BWR reactors) or annulus ventilation systems in dual containment structures should be treated as a ground-level release or an elevated stack release, as appropriate. The diffuse area source model may be appropriate for time intervals for which the secondary containment or annulus ventilation system is not capable of maintaining the requisite negative pressure differential specified in Technical Specifications or in the UFSAR. Secondary containment bypass leakage (i.e., leakage from the primary containment that bypasses the secondary containment and is not collected by the SGTS) should be treated as a ground-level release or an elevated stack release, as appropriate.	Not applicable	
3.2.4.7	A second possible application of the diffuse area source model is determining a χ/Q value for multiple (i.e., 3 or more) roof vents. This treatment would be appropriate for configurations in which (1) the vents are in a close arrangement, (2) no individual vent is significantly* closer to the control room intake than the center of the area source, (3) the release rate from each vent is approximately the same, and (4) no credit is taken for plume rise. The	Not applicable	

TABLE A.1.194-1

CONFORMANCE WITH REGULATORY GUIDE 1.194 (DIFFUSE AREA SOURCE GUIDANCE)

RG Section	RG Position	Analysis	Comments
	distance to the receptor is measured from the closest point on the perimeter of the assumed area source. For assumed areas that are not circular, the area width is measured perpendicular to the line of sight from the center of the assumed source to the control room intake. The initial diffusion coefficient sigma Yo is found by Equation 3; sigma Zo is assumed to be 0.0. * The degree of significance will depend on the radius or width of the assumed area and the proximity of the vent cluster to the control room intake. As the radius decreases or the distance from the cluster to the control room intake increases, the less significance the position of any one vent has.		
3.2.4.8	A third possible application of the diffuse area source model is determining a χ/Q value for large louvered panels or large openings (e.g., railway doors on BWR Mark I plants) on vertical walls. This treatment would be appropriate for a louvered panel or opening when (1) the release rate from the building interior is essentially equally dispersed over the entire surface of the panel or opening and (2) assumptions of mixing, dilution, and transport within the building necessary to meet condition 1 are supported by the interior building arrangement. The staff has traditionally not allowed credit for mixing and holdup in turbine buildings because of the buoyant nature of steam releases and the typical presence of high volume roof exhaust ventilators. The distance to the receptor and the release height is measured from the center of the louvered panel or opening. Initial diffusion coefficients are found using Equations 3 and 4 assuming the width and height is that of the panel or opening rather than that of the building. If the area source and the intake are on the same building surface such that wind flows along the building surface would transport the release to the intake, the initial dispersion coefficient will need to be adjusted. If the included angle between the source-receptor line of sight and the vertical axis of the assumed source is less than 45 degrees, sigma Y_o should be set to 0.0. If the included angle between the source receptor line of sight and the horizontal axis of the assumed source is less than 45 degrees, sigma Z_o should be set to 0.0.	Not applicable	

REGULATORY GUIDE 1.196

CONTROL ROOM HABITABILITY AT LIGHT-WATER NUCLEAR POWER REACTORS

The Licensee complies with the requirements in Revision 0 of this regulatory guide with the following exceptions and clarifications:

The Control Room Envelope Habitability Program is governed by Technical Specification (TS) 5.5.18, "Control Room Envelope Habitability Program," approved via TS Amendment No. 146 for Braidwood Station, Units 1 and 2 and TS Amendment No. 151 for Byron Station, Units 1 and 2. This TS Amendment modified TS requirements related to control room envelope habitability in accordance with TS Task Force (TSTF) Traveler TSTF-448, Revision 3, "Control Room Habitability." The implementation of a Control Room Envelope (CRE) Habitability Program is the result of a regulatory commitment made in response to NRC Generic Letter (GL) 2003-01, "Control Room Habitability." The CRE Habitability Program was implemented as a result of findings at facilities that existing Technical Specifications may not be adequate to ensure the requirements of 10 CFR 50 Appendix A GDC 19 are met as described in GL 2003-01.

Survey of chemical sources is to be performed at least once per 6 years as part of the periodic assessment of CRE habitability required by TS 5.5.18.

Regulatory Guide (RG) 1.196 references RG 1.78, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," Revision 1. As described in this Appendix to the UFSAR, Braidwood and Byron comply with Revision 0 of RG 1.78. Compliance with RG 1.78 is further described in Section 2.2 and subsection 6.4.1.

As allowed by paragraph C.4 of Regulatory Guide (RG) 1.78, Revision 0, "The toxicity limits should be taken from appropriate authoritative sources." NUREG/CR-6624 is considered an appropriate authoritative resource and, therefore, the toxicity limits contained within may be used for periodic toxic gas surveys in place of those contained in RG 1.78, Revision 0.

REGULATORY GUIDE 1.197

DEMONSTRATING CONTROL ROOM ENVELOPE INTEGRITY AT NUCLEAR POWER REACTORS

The Licensee has only committed to the testing methods and frequencies as specified in Sections C.1 and C.2 of Revision 0 to this regulatory guide. The requirements for determining the unfiltered air inleakage past the CRE boundary into the CRE is defined in Technical Specification 5.5.18, "Control Room Envelope Habitability Program."

REGULATORY GUIDE 8.2

GUIDE FOR ADMINISTRATIVE PRACTICES IN RADIATION MONITORING

Administrative procedures and practices of radiation monitoring are based on 10 CFR 20 and Regulatory Guide 8.2, Revision 0.

REGULATORY GUIDE 8.7

INSTRUCTIONS FOR RECORDING AND REPORTING OCCUPATIONAL RADIATION EXPOSURE DATA

The occupational radiation exposure record system is based on Regulatory Guide 8.7, Revision 2.

REGULATORY GUIDE 8.8

INFORMATION RELEVANT TO ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES AT NUCLEAR POWER STATIONS WILL BE AS LOW AS IS REASONABLY ACHIEVABLE

Regulatory Guide 8.8 describes information that is relevant to meeting the criterion that exposures of station personnel to radiation during routine operations of the station will be as low is reasonably achievable (ALARA).

Maintaining occupational radiation doses ALARA is a function of the health physics program (12.5), station design (12.3), and administrative policies (12.1.1.3). The Licensee has also used operating experience during the design phases and is utilizing supervisory personnel who have several years of operating experience working in its licensed stations.

The health physics program includes the radiation protection program, training, and instruction. The administrative policies maintain occupational exposure ALARA and establish station organization, responsibilities, and procedures. The station design includes access control, shielding, facility design, equipment design, airborne control, crud control, radiation monitoring waste treatment, and modifications (based on operating experience).

The guidance provided by Regulatory Guide 8.8 (all issues) and by WCAP-8872, "Design, Inspection, Operation, and Maintenance Aspects of the Westinghouse NSSS to Maintain Occupational Radiation Exposures as Low As Reasonably Achievable," has been used as an aid for the radiation protection design.

Regulatory Guide 8.8, Revision 3, Sections C.1, C.3, and C.4 are used as a basis for developing the ALARA and radiation protection programs with the following exceptions: C1B page 8.8-6 - qualifications for radiation protection manager (RPM) job. The station does not commit to requiring the RPM to take any type of certification exam.

REGULATORY GUIDE 8.9

ACCEPTABLE CONCEPTS, MODELS, EQUATIONS, AND ASSUMPTIONS FOR A BIOASSAY PROGRAM

The bioassay program will be in compliance with Revision 1 of Regulatory Guide 8.9, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program." Bioassay services are performed by either a contracted vendor or by personnel. When a vendor is contracted to perform bioassay, a requirement of the contract is in compliance with the appropriate regulatory requirements. Exelon Generation Company's bioassay program is in compliance with the applicable regulatory requirements.

REGULATORY GUIDE 8.10

OPERATING PHILOSOPHY FOR MAINTAINING OCCUPATIONAL RADIATION EXPOSURES AS LOW AS IS REASONABLY ACHIEVABLE

The operating philosophy for maintaining occupational exposures ALARA is based on Revision 1 of Regulatory Guide 8.10 to the degree considered reasonable by the respective stations.

REGULATORY GUIDE 8.12

Revision 0, December 1974

CRITICALITY ACCIDENT ALARM SYSTEMS

Area monitors are provided near the spent fuel storage pool to alarm locally and in the main control room. These monitors will respond to radiation in the event of a criticality accident in the new fuel storage area. These monitors meet the intent of Regulatory Guide 8.12 and are further described in Subsection 11.5.2.2.6.

REGULATORY GUIDE 8.14

PERSONNEL NEUTRON DOSIMETERS

The Nuclear Regulatory Commission has withdrawn Regulatory Guide 8.14, "Personnel Neutron Dosimeters." Revision 1 of Regulatory Guide 8.14, published in August 1977, endorsed ANSI N319-1976, "American National Standard for Personnel Neutron Dosimeters (Neutron Energies Less Than 20 MeV)," which has been replaced by ANSI N13.52-1999, "Personnel Neutron Dosimeters." Regulatory Guide 8.14 does not need to be revised because regulations are in place that require licensees to have an adequate dosimetry program.

Licensees are required by 10CFR20.1501 to use dosimetry processors accredited through the National Voluntary Laboratory Accreditation Program (NVLAP). NVLAP requires processors to use new standards for personnel dosimetry, ANSI N13.52-1999 and ANSI N13.11-1993, "Personnel Dosimetry Performance-Criteria for Testing," to maintain an appropriate quality for dosimetry processing.

REGULATORY GUIDE 8.15

ACCEPTABLE PROGRAMS FOR RESPIRATORY PROTECTION

Due to changes in 10 CFR 20, Revision 0 of Regulatory Guide 8.15 is no longer applicable. Regulatory Guide 8.15, Revision 1, exceeds both the requirements of 10 CFR 20 and the recommendations of Regulatory Guide 8.15, Revision 0. The respiratory protection program will be maintained in accordance with 10 CFR 20 and Regulatory Guide 8.15, Revision 1, except in the following areas where Guide recommendations exceed those in Revision 0:

1. Written procedures do not specify training and minimum qualifications of respirator program supervisors and implementation personnel. (Reference: Section 3.2)

2. Where 10 CFR 20 does not require internal dose monitoring, respirator users will not have internal exposure or internal dose documented, since it is not required by 10 CFR 20. (Reference: Section 3.3.4)

3. The respirator program will not document responsibilities of each person in the program, minimum training and retraining requirements, or minimum qualification. (Reference: Section 3.5)

4. Inspection frequencies for respiratory equipment will be determined in accordance with applicable regulations. Inspection frequencies will be documented in station procedures. (Reference: Section 4.3)

5. SCBA cylinders will be tested and marked in accordance with any applicable regulations. (Reference: Section 4.3)

6. Respirator cartridges that are re-used will be tested before re-use in accordance with any applicable regulations. (Reference: Section 4.9)

7. Medical evaluations for respirator use are determined in accordance with a program established by a Company physician. This program is developed to comply with applicable regulatory requirements. (Reference: Section 5.1)

8. The respirator training program does not involve "hands-on" training for all respirator types. Training is conducted to meet applicable regulatory requirements. (Reference: Section 5.2)

9. Expectations are set for re-testing for a respirator fit test when an individual has a significant weight change, but not at the point when a weight change is exactly 10% or more. (Reference: Section 5.3.5)

10. Standby rescue persons are not provided for workers wearing supplied air hoods. (Reference: Section 6.1)

11. Testing frequencies and test methods for breathing air quality will be conducted in accordance with applicable regulations. Inspection frequencies will be documented in station procedures. (Reference: Section 6.5)

12. Wipe samples will be taken at air connection points at the discretion of the licensee as determined necessary. (Reference: Section 6.5.6)

13. Specific time limits have not been set regarding length of time individuals are required to work while using respirators. (Reference: Section 6.7)

REGULATORY GUIDE 8.19

OCCUPATIONAL RADIATION DOSE ASSESSMENT IN LIGHT-WATER REACTOR POWER PLANTS DESIGN STAGE MAN-REM ESTIMATES

The dose assessment objectives in Revision 0 of Regulatory Guide 8.19 have been included in the UFSAR as indicated below. Revision 0 was the current revision of the regulatory guide when the operating license application was docketed.

Item C.(1)

The occupational radiation exposure estimates are in Subsection 12.4.4.

Item C.(2)

B/B's radiation exposure assessment bases are described in Subsections 12.5.3, 12.1.2.7, and 12.4.

Item C.(3)

Design changes which have resulted from Commonwealth Edison's dose assessment process are included in Subsections 12.1.2.3, 12.1.2.7, and 12.4.

REGULATORY GUIDE 8.25

CALIBRATION AND ERROR LIMITS OF AIR SAMPLING INSTRUMENTS FOR TOTAL VOLUME OF AIR SAMPLED

Revision 1 of Regulatory Guide 8.25 does not apply to nuclear power plants. The air sample calibration program requirements are described in UFSAR section 12.3.4.2.