March 1, 2001

MEMORANDUM TO: George F. Dick, Senior Project Manager
Project Directorate III
Division of Licensing and Project Management
Office of Nuclear Reactor Regulation

FROM: Peter R. Wilson, Acting Chief ***/RA/***
Licensing Section
Probabilistic Safety Assessment Branch
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

SUBJECT: SAFETY EVALUATION FOR PROPOSED POWER UPRATES FOR BYRON STATION, UNITS 1 AND 2 (TAC NOS. MA9426 AND MA9427) AND FOR BRAIDWOOD STATION UNITS 1 AND 2 (TAC NOS. MA9428 AND MA9429)

In response to your request, the Probabilistic Safety Assessment Branch (SPSB) has completed its review of the proposed power uprates for the Byron and Braidwood stations. Our review is limited to the potential increase in the radiological consequences as a result of the proposed power uprate to 3586.6 MWt from 3411MWt (approximately 5.15 percent increase). Our
evaluation is based on a power level of 3658.3 MWt (102 percent of the requested uprated power level of 3586.6 MWt). An input to the power uprate safety evaluation is attached.

On the basis of our review of the licensee’s radiological analyses and our own confirmatory assessment of the radiological consequences of the postulated design basis accidents, we find that the proposed power uprate is acceptable

This review was performed by Jay Lee of SPSB.

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

Attachment: Safety Evaluation

CONTACT: Jay Lee (415-1080)

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
FACILITY OPERATING LICENSE NOS NPF-37, 66 72, AND 77,
COMMONWEALTH EDISON COMPANY
BYRON AND BRAIDWOOD STATIONS, UNIT NOS. 1 AND 2
DOCKET NOS. STN 50- 454, 455, 456, AND 357

3.5 <u>Radiological Analyses</u>

To demonstrate that the Byron and Braidwood engineered safety features (ESFs) designed to mitigate the radiological consequences will remain adequate at uprated power level of 3586.6 MWt, the licensee reevaluated the offsite and control room radiological consequences for the following postulated design basis accidents (DBAs) at a power level of 3658.3 MWt (102 percent of requested uprated power level of 3586.6 MWt):

Main steam line break
Locked reactor coolant pump (RCP) rotor
Locked RCP rotor with power-operated relief valve (PORV) failure
Rod ejection
Small line break
Steam generator tube rupture
Large-break loss-of-coolant accident (LOCA)
Small-break LOCA
Fuel handling accident
Gas decay tank rupture

The licensee submitted the results of its offsite and control room dose calculations. In addition, the licensee provided the major assumptions and parameters used in its dose calculations. As documented In the submittals, the licensee has determined that the existing ESF systems at Byron and Braidwood will still provide assurance that the radiological consequences of the postulated DBAs at the exclusion area boundary (EAB), in the low population zone (LPZ), and in the control room are within the radiation dose acceptance criteria specified in the Standard Review Plan (SRP) and the dose guidelines proved in 10 CFR 100.

The staff has reviewed the licensee’s analysis and has performed an independent confirmatory radiological consequence dose calculation for the following 6 bounding DBAs:

Large-break loss-of-coolant accident (LOCA)
Main steam line break

Attachment

Steam generator tube rupture
Fuel handling accident
Locked RCP rotor with a steam generator PORV failure
Rod ejection

The results of the staff's independent radiological consequence calculations are given in Tables 1 and 2 for Byron and Braidwood stations, respectively. The major parameters and assumptions used by the staff are listed in Tables 3 through 14. The staff did not perform independent dose calculations for the small-break LOCA and the small-line break accident since the radiological consequences of these accidents at Byron and Braidwood stations are bounded by that of the large-break LOCA. The radiological consequences of the locked RCP rotor accident is also bounded by that of the accident with a steam generator PROV failure. The staff also did not perform an independent dose calculation for gas decay tank rupture because the quantity of radioactivity in each gas decay tank is limited by Byron and Braidwood Technical Specification (TS) 5.12 and the licensee did not request to change the limits for this TS.

In addition, the licensee requested to amend the definition of Dose Equivalent Iodine-131 in the Byron/Braidwood Technical Specification Section 1.1, "Definition." The current definition defines Dose Equivalent Iodine-131 as follows:

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

The requested amendment would add two following references to this definition; (1) Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Release of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, 1977, and (2) ICRP 30, "Limits for Intakes of Radionuclides by Workers," Supplement to Part 1, page 192-212, Table titled," Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."

The amended definition would then read as follows:

> "DOSE EQUIVALENT I-131 shall be that concentration of I-131(microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133 I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," *those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, 1977, or ICRP 30, Supplement to Part 1, page 192-212,*

Table titled," Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."

The International Commission on Radiation Protection Publication 30 (ICRP 30) incorporates the considerable advances in the state of knowledge of radionuclide dosimetry and biological transport in humans achieved in the past few decades and the NRC embraced it and adopted its values into the revision of Part 20, "Standards for Protection Against Radiation," in 1994. Therefore, the staff finds that this requested amendment to the definition is acceptable.

3.5.1 Loss-of-Coolant Accident

The current radiological consequence analysis for the postulated LOCA using Technical Information Document (TID)-14844 source term is provided in Byron/Braidwood Updated Final Safety Analysis Report (UFSAR) Section 15.6.5. The licensee reevaluated the offsite and control room radiological consequences of the postulated LOCA at an uprated power level of 3658.3 MWt. The staff has reviewed the licensee's analysis and performed an independent confirmatory dose calculation for the following two potential fission product release pathways after the postulated LOCA:

(1) containment leakage
(2) post-LOCA leakage from ESF systems outside containment

The current maximum allowable primary containment leakage rate (L_a), is 0.1 percent of containment air weight per day. The staff used this leak rate for the first 24 hours into the accident and 0.05 percent of containment air weight per day for the remaining duration of the accident (30 days). Only fission product removal in the containment atmosphere is achieved by the containment spray system (CSS) other than initial plate out in the containment assumed in the source term. The CSS is an ESF system and is designed to provide containment cooling and fission product removal in the containment following the postulated LOCA. The CSS consists of two redundant and independent loops. Each loop has a design spray water flow capacity of 2950 gpm.

The licensee calculated elemental iodine removal rate by the CSS using the methodologies provided in SRP 6.5.2 and determined that elemental iodine removal rate to be well above the upper limit specified in the SRP and therefore, the licensee used an elemental iodine removal rate of 20 per hour specified in the SRP as an upper limit. The licensee also calculated removal rate of iodine in particulate form using the methodologies provided in SRP 6.5.2 and determined the rate to be 6 per hour. The staff finds these iodine removal rates determined by the licensee are acceptable. The licensee assumed removal of elemental iodine from the containment atmosphere only during spray injection period (from 0.025 to 0.373 hours following the accident) and determined that the decontamination factors (DFs) 100 and 50 referenced in the

SRP for elemental iodine and iodine in particulate form respectively, are not reached during this spray injection period.

The licensee modeled the containment atmosphere as two discrete nodes representing sprayed and unsprayed regions and assumed these nodes are mixed by the reactor containment fan cooler (RCFC) system fans. The RCFC system is an ESF system and is designed to remove energy released in the containment following a postulated LOCA (along with the emergency core cooling system and the containment spray system). The RCFC system is a redundant system consisting two 100 percent trains. Each train is powered from a separate redundant essential bus and has a capacity of 1.18E+6 cfm air flow. The staff assumed only one RCFC system train will be operational with a total air mixing flow rate of 1.06E+6 cfm (90 percent of fan capacity) in the containment following the postulated LOCA. This represents a mixing rate of approximately 12 unsprayed volumes per hour between the sprayed and unsprayed portions of the containment atmosphere.

Any leakage water from ESF components located outside the primary containment releases fission products during the recirculating phase of long-term core cooling following a postulated LOCA. The licensee assumed this leakage to be less than 7820 cc/hour, which is twice the leakage value of 3910 cc/hour assumed in the Byron/Braidwood UFSAR, and that this leakage to begin at the time of the postulated LOCA throughout the entire duration of the accident (30 days). The staff finds the leakage value assumed by the licensee to be acceptable. The licensee further assumed that ten percent of all forms of iodine contained in the leakage is released (consistent with guidelines provided in SRP Section 15.6.5) to the environment through Auxiliary Building Filtration System (ABFS) which is designed as an ESF system. The staff assumed 1 percent of the ABFS flow will bypass the charcoal adsorber in the ABFS.

The staff has reviewed the licensee's analysis and finds that the calculational methods used for the radiological consequence assessment are acceptable and that the radiological consequences calculated by the licensee meet the relevant dose acceptance criteria. The resulting radiological consequence analyses performed by the staff for the EAB, the LPZ, and for the control room are provided in Tables 1 and 2 for Byron and Braidwood stations, respectively. The major parameters and assumptions used for the postulated LOCA dose calculations by the staff are provided in Table 3. The radiological consequences calculated by the staff are consistent with those calculated by the licensee. Therefore, the staff concludes that the Byron and Braidwood stations operating at the uprated power level of 3658.3 MWt will still provide reasonable assurance that the radiological consequences of a postulated LOCA will not exceed the dose guidelines provided in 10 CFR 100 and the control room dose acceptance criteria specified in GDC 19.

3.5.2 Main Steamline Break Outside Containment

The licensee has reevaluated the radiological consequences of a postulated main steamline break accident occurring outside containment and upstream of the main steam isolation valves at an uprated power level of 3658.3 MWt. The licensee analyzed this hypothetical accident using 0.5 gpm of primary-to-secondary leakage through the faulted steam generator and 0.218 gpm through each of the intact steam generator. The staff has reviewed the licensee's analysis and finds that the calculational methods used for the radiological consequence assessment are acceptable and that the radiological consequences calculated by the licensee meet the relevant dose acceptance criteria.

The staff performed an independent radiological consequence calculation for 2 cases. For case 1, the staff assumed that a temporary increase in the primary coolant iodine concentration (iodine spike) occurred as a result of the power/pressure transient caused by the main steamline break accident. Before the accident, the Byron and Braidwood reactor were assumed to be operating at its TS equilibrium limit of 1.0 µCi/gm dose equivalent iodine-131 (DEI-131) in the primary coolant. The iodine spike generated during the accident is assumed to increase the release rate of iodine from the fuel by a factor of 500. This increase in the release rate results in an increasing concentration in the primary coolant during the course of the accident. For case 2, the staff assumed that previous reactor operation had resulted in a primary coolant iodine concentration equal to the maximum instantaneous TS limit of 60 µCi/gm DEI-131. For both cases, the staff assumed that all fission products in the entire mass of secondary water in the faulted steam generator (167,000 lbs) is released to the environment directly with no iodine partition.

The resulting radiological consequence analyses for the EAB, the LPZ, and for the control room are provided in Tables 1 and 2 for Byron and Braidwood stations, respectively. The major parameters and assumptions used by the staff for the main steam line break accident are provided in Table 4. The radiological consequences calculated by the staff are consistent with those calculated by the licensee. Therefore, the staff concludes that the Byron and Braidwood stations operating at the uprated power level of 3658.3 MWt will still provide reasonable assurance that the radiological consequences of a postulated main steamline break accident occurring outside containment will not exceed the dose acceptance criteria specified in SRP Section 15.1.5 and dose guidelines set forth in 10 CFR 100, and the control room dose acceptance criteria specified in GDC 19.

3.5.3 Steam Generator Tube Rupture Accident

The licensee has reevaluated the radiological consequences of a postulated steam generator tube rupture accident at an uprated power level of 3658.3 MWt and provided a radiological consequence analysis. The staff has reviewed the licensee's analysis and finds that the calculational methods used for the radiological consequence assessment are acceptable and that the radiological consequences calculated by the licensee meet the relevant dose acceptance criteria.

To verify the Westinghouse assessments, the staff performed independent radiological consequence calculations for two scenarios for the steam generator tube rupture accident as the staff did for the steamline break accident above. For case 1, the staff assumed that a temporary increase in the primary coolant iodine spike occurred as a result of the power/pressure transient caused by the steam generator tube rupture. Before the postulated accident, the Byron and Braidwood stations were assumed to be operating at its TS equilibrium iodine concentration limit of 1.0 µCi/gm DEI-131 in the primary coolant. The iodine spike generated during the accident is assumed to increase the release rate of iodine from the fuel by a factor of 500. This increase in the release rate results in an increasing iodine concentration in the primary coolant during the course of the accident. For case 2, the staff assumed that previous reactor operation had resulted in a primary coolant concentration equal to the maximum instantaneous concentration limit of 60 µCi/gm DEI-131 specified in the Byron and Braidwood TSs.

The major parameters and assumptions used by the staff are provided in Table 5, and the resulting radiological consequence analyses for the EAB and the LPZ and for the control room are provided in Tables 1 and 2 for Byron and Braidwood stations, respectively. The radiological consequences calculated by the staff are consistent with those calculated by the licensee. Therefore, the staff concludes that the Byron and Braidwood will still provide reasonable assurance that the radiological consequences of a postulated steam generator tube rupture accident will not exceed the dose acceptance criteria specified in SRP Section 15.1.5 and dose guidelines set forth in 10 CFR 100, and the control room dose acceptance criteria specified in GDC 19.

3.5.4 Fuel Handling Accident

The licensee has reevaluated the radiological consequences of a postulated fuel handling accident (FHA) at an uprated power level of 3658.3 MWt and provided a radiological consequence analysis. The staff has reviewed the licensee's analysis and finds that the calculational methods used for the radiological consequence assessment are acceptable and that the radiological consequences calculated by the licensee meet the relevant dose acceptance criteria. A FHA can be postulated to occur either inside or outside of the containment. If the FHA occurs in the containment, the release of fission products can be terminated by closure of the containment based on the detection of high airborne radioactivity. For the postulated FHA occurring outside the containment, the licensee assumed that fission products are released to the environment within two hour period through the Fuel Handling Building Exhaust System (FHBES). The FHBES is an ESF system and designed to operate continuously during plant normal operation bypassing charcoal adsorbers. Upon receiving of high radiation signal, the effluent from fuel handling building is routed through the charcoal adsorbers.

The staff performed the radiological consequences analyses of a FHA assuming a single fuel assembly dropped onto the irradiated fuel stored in the spent fuel pool. The kinetic energy of the falling fuel assembly will be assumed to break open the maximum possible number of fuel rods using perfect mechanical efficiency. Instantaneous release of noble gases and radioiodine vapor from the gaps of the broken rods (10 percent of noble gases other than krypton-85, 30 percent krypton-85, and 12 percent iodine) will be assumed to occur, with the released gases bubbling up through the fuel pool water. The staff assumed an overall effective fuel pool decontamination factor of 100 for iodine and of 1 for noble gases. The staff also provided a 90 percent iodine removal efficiency for the FHBES filter and assumed 1 percent of flow will bypass the filter.

The major parameters and assumptions used by the staff are provided in Table 6, and the resulting radiological consequence analyses are provided in Tables 1 and 2 for Byron and Braidwood stations, respectively. The radiological consequences calculated by the staff are consistent with those calculated by the licensee. Therefore, the staff concludes that the Byron and Braidwood design will still provide reasonable assurance that the radiological consequences of a postulated fuel handling accident will be well within the dose criteria specified in SRP 15.7.4 and the control room dose acceptance criteria specified in GDC 19.

3.5.5 Locked Rotor Accident with a steam generator PORV Failure

The reactor primary coolant pump locked rotor accident is caused by an instantaneous seizure of a reactor coolant pump rotor rapidly reducing the primary coolant flow through the affected reactor coolant loop leading to a reactor trip on a low-flow signal. The licensee analyzed this hypothetical accident assuming 2 percent of the fuel elements will experience cladding failure, releasing the entire fission product inventory in the fuel gap (10 percent of the core activity) to the reactor coolant. The licensee assumed the primary-to-secondary steam generator tube leak rate is 0.5 gpm for the faulted steam generator and 0.218 gpm for each of the intact steam generators. A steam generator PORV is assumed to fail-open resulting in an uncontrolled blowdown of steam from the steam generators directly to the environment for 20 minutes. In addition, radioactivity is assumed to be released to the environment by way of primary-to-secondary leakage and steaming from the secondary side to the environment. The staff finds these assumptions to be conservative and therefore, acceptable.

The staff has reviewed the licensee's analysis and performed an independent confirmatory dose calculation. The results of the staff's independent radiological consequence calculation are provided in Tables 1 and 2 for Byron and Braidwood stations, respectively. The major parameters and assumptions used by the staff in the radiological consequence calculations are listed in Table 8. The radiological consequences calculated by the staff are consistent with those calculated by the licensee.

The staff concludes that the Byron and Braidwood stations operating at uprated power level of 3658.3 MWt will still provide reasonable assurance that the radiological consequences of a postulated LOCA will not exceed a small fraction the dose guidelines set forth in 10 CFR 100 (30 rem to the thyroid and 2.5 rem to the whole body) and the control room dose acceptance criteria specified in GDC 19.

3.5.6 Rod Ejection Accident

The mechanical failure of a control rod mechanism pressure housing is postulated to result in the ejection of a rod cluster control assembly and drive shaft. Because of the resultant opening in the pressure vessel, primary coolant is released to the containment with concurrent rapid depressurization of the reactor pressure vessel. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

The licensee assumed that 15 percent of the fuel elements will experience cladding failure, releasing the entire fission product inventory in the fuel-cladding gap of these elements. In addition, the licensee assumed that 0.375 percent of the fuel rods will experience fuel melting. The licensee performed its calculations to obtain these parameters using the guidelines provide in Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for PWRs," therefore, the staff finds these assumptions to be acceptable. The staff has reviewed the licensee’s analysis and finds that the calculational methods used for the radiological consequence assessment are acceptable and that the radiological consequences calculated by the licensee meet the relevant dose acceptance criteria.

The licensee assumed that the release of fission products to the environment will occur via either of two pathways. The first pathway involves a release of primary coolant to the containment, which is then assumed to leak to the environment at the design leak rate of the containment. In the second pathway, fission products would reach the secondary coolant via the steam generators with a maximum total allowable primary-to-secondary leak rate of 1 gallon per minute. To verify the licensee’s assessments, the staff performed independent radiological consequence calculations for the same two pathways as described above for the control rod ejection accident. The major parameters and assumptions used by the staff are provided in Table 7, and the resulting radiological consequence analyses are provided in Tables 1 and 2 for Byron and Braidwood stations, respectively. The radiological consequences calculated by the staff are consistent with those calculated by the licensee.

The staff concludes that the Byron and Braidwood stations operating at an uprated power level of 3658.3 MWt will still provide reasonable assurance that the radiological consequences of a postulated rod ejection accident will not exceed a small fraction the dose guidelines set forth in 10 CFR 100 (30 rem to the thyroid and 2.5 rem to the whole body) and the control room dose acceptance criteria specified in GDC 19.

TABLE 1

Radiological Consequences (rem)
Byron Station, Units 1 and 2

Design Basis Accidents	EAB		LPZ		Control Room	
	Thyroid	WB[1]	Thyroid	WB	Thyroid	WB
LOCA	61	3	7	<1	15	<1
MSLB						
Pre-accident	4.6	<1	0.5	<1	29	<1
Accident-initiated	5.0	<1	2.0	<1	13	<1
SGTR						
Pre-accident	11	<1	3.6	<1	3.3	<1
Accident-initiated	9.0	<1	0.3	<1	0.3	<1
FHA	56	3.8	1.7	<1	1.3	<1
Locked rotor	13	<1	1.0	<1	16	<1
Rod ejection	34	1	6	<1	27	<1

[1] Whole body

TABLE 2

Radiological Consequences (rem)
Braidwood Station, Units 1 and 2

Design Basis Accidents	EAB		LPZ		Control Room	
	Thyroid	WB	Thyroid	WB	Thyroid	WB
LOCA	82	4	34	<1	15	<1
MSLB						
Pre-accident	6.2	<1	2.6	<1	13	<1
Accident-initiated	3.6	<1	4.8	<1	29	<1
SGTR						
Pre-accident	14	<1	1.3	<1	1.1	<1
Accident-initiated	12	<1	1.1	<1	1.0	<1
FHA	75	5.1	7.0	<1	1.4	<1
Locked rotor	18	<1	2.0	<1	16	<1
Rod ejection	45	<1	22	<1	26	<1

Table 3
Parameters and Assumptions Used in
Radiological Consequence Calculations
Loss-of-Coolant Accident

Parameter	Value
Reactor power	3658.3 MWt
Containment volume of sprayed region	2.35E+6 ft^3
Containment volume of unsprayed region	5.0E+5 ft^3
Flow rate from sprayed to unsprayed region	1.06E+5 cfm
Flow rate from unsprayed to sprayed region	1.06E+5 cfm
Containment leak rate to environment	
0 - 24 hours	0.1 percent per day
1 - 30 days	0.05 percent per day
Spray removal rates	
Elemental iodine	20 per hour
Time to reach DF(1) of 100	Not reached
Particulate iodine	6 per hour
Time to reach DF(1) of 50	Not reached
Spray operation	
Initiation time	90 seconds
Termination time for injection	22.4 minutes
ECCS leak rate	
0 to 30 days	7820 cc/hr
Iodine partition factor	10 percent
Sump volume	38979 ft^3
Auxiliary building exhaust filter efficiency	90 percent
Auxiliary building exhaust filter bypass	1 percent
Control room isolation time	15 seconds

(1) Decontamination factor

Table 4
Parameters and Assumptions
Used in
Radiological Consequence Calculations
Main Steamline Break Accident

Parameter	Value
Reactor power	3658.3 MWt
Primary coolant iodine activity prior to accident	
Pre-existing spike	60 µCi/gm DEI-131
	42.6 µCi/gm I-131
Accident-initiated spike	1.0 µCi/gm DEI-131
	0.77 µCi/gm I-131
Secondary coolant iodine activity	
prior to accident	0.1 µCi/gm DEI-131
	0.077 µCi/gm I-131
Iodine spike (appearance) rate increase	500 times
Duration of accident-initiated spike	6 hours
Steam generator tube leak rates	
Faulted steam generator	0.5 gpm
Intact steam generators	0.654 gpm from 3 steam generators
Steam releases	
Faulted steam generator	1.67E+5 lbs
Intact steam generators	
0 to 2 hours	4.42E+5 lbs
2 to 8 hours	9.77E+5 lbs
8 to 40 hours	2.216E+6 lbs
Duration of activity release	40 hours
Control room isolation time	5 minutes

Table 5
Parameters and Assumptions
Used in
Radiological Consequence Calculations
Steam Generator Tube Rupture Accident

Parameter	Value
Reactor power	3658.3 MWt
Primary coolant iodine activity prior to accident	
Pre-existing spike	60 µCi/gm DEI-131
	42.6 µCi/gm I-131
Accident-initiated spike	1.0 µCi/gm DEI-131
	0.77 µCi/gm I-131
Secondary coolant iodine activity	
prior to accident	0.1 µCi/gm DEI-131
	0.077 µCi/gm I-131
Iodine spike (appearance) rate increase	500 times
Duration of accident-initiated spike	8 hours
Steam generator tube leak rates	
Intact steam generator	1.0 gpm total
Steam releases	
Faulted steam generator	
0 to 2 hours	9.75E+4 lbs
2 to 8 hours	2.69E+4 lbs
Intact steam generators	
0 to 2 hours	6.53E+5 lbs
2 to 8 hours	1.20E+6 lbs
Average primary coolant flashing factor	0.015
Control room isolation time	10 minutes

Table 6
Parameters and Assumptions
Used in
Radiological Consequence Calculations
Fuel Handling Accident

Parameter	Value
Reactor power	3658.3 MWt
Radial peaking factor	1.7
Fission product decay period	48 hours
Number of fuel rods damaged	one assembly
Fuel pool water depth	23 ft
Fuel gap fission product inventory	
Noble gases excluding Kr-85	10 percent
Kr-85	30 percent
I-131	12 percent
Other iodines	10 percent
Fuel pool decontamination factors	
Iodine	100
Noble gases	1
Auxiliary building exhaust system filter efficiency	90 percent
Fuel handling building exhaust system filter bypass	1.0 percent
Duration of accident	2 hours
Control room isolation time	15 seconds

Table 7
Parameters and Assumptions
Used in
Radiological Consequence Calculations
Control Rod Ejection Accident

Parameters	Values
Reactor power	3658.3MWt
Fuel gap release fraction	10 percent
Fraction of Fuel rods failed	15 percent
Fraction of fuel melt	0.375 percent
Primary coolant activity	60 µCi/gm DEI-131
Secondary coolant activity	0.1 µCi/gm DEI-131
Iodine plate out in containment	50 percent
Containment leak rates	
0 to 24 hours	0.1 percent
1 to 30 days	0.05 percent
Primary coolant mass	2.063E+8 gm
Primary-to-secondary leak rate	1.0 gpm
Iodine partition factor	0.01
Duration of primary-to-secondary leak	1.1 hours
Steam release from secondary	2.5E+6 lbs
Control room isolation time	2.5 minutes

Table 8
Parameters and Assumptions
Used in
Radiological Consequence Calculations
Locked Rotor Accident with Power-Operated Relief Valve Failure

Parameter	Value
Reactor power	3658.3 MWt
Primary coolant iodine activity	60 µCi/gm DEI-131
Secondary coolant iodine activity	0.1 µCi/gm DEI-131
Steam generator tube leak rates	
Faulted steam generator	0.5 gpm
Intact steam generator	0.218 gpm each
Fraction of fuel rods failed	2 percent
Fraction of fission product in gap	10 percent
Iodine partition factors	
steam generators	0.01
PORV release	1
Primary coolant mass	2.063E+8 gm
Duration of PORV release	20 minutes
Duration of steam release	40 hours
Steam release through PORV	
0 to 20 minutes	3.788E+6 gm
After 20 minutes	0
Iodine partition factor used	1
Steam release through steam generators	
0 to 2 hours	2.72E+6 gm
2 to 8 hours	1.40E+6 gm
8 to 40 hours	5/30E+5 gm
Iodine partition factor used	0.01
Control room isolation time	5 minutes

Table 9

Control Room

Parameter	Value
Volume	70,275 ft^3
Emergency ventilation system flow rates	
Filtered makeup air flow	54,000 cfm
Recirculation flow	35,000 cfm
Unfiltered inleakage	100 cfm
Filter efficiencies for intake flow	
Elemental iodine	99 percent
Organic iodine	99 percent
Particulate iodine	99 percent
Filter efficiencies for Recirculation flow	
Elemental iodine	90 percent
Organic iodine	90 percent
Particulate iodine	80 percent
Delay to switch over from normal to emergency operation	15 seconds

Table 10

Meteorological Data

Byron Station

Exclusion Area Boundary

Time (hr)	X/Q (sec/m^3)
0-2	5.7E-4

Low Population Zone Distance

Time (hr)	X/Q (sec/m^3)
0-8	1.7E-5
8-24	2.4E-6
24-96	1.1E-6
96-720	7.6E-7

Table 11

Meteorological Data

Braidwood Station

Exclusion Area Boundary

Time (hr)	X/Q (sec/m^3)
0-2	7.7E-4

Low Population Zone Distance

Time (hr)	X/Q (sec/m^3)
0-8	7.1E-5
8-24	1.4E-5
24-96	7.1E-6
96-720	4.1E-6

Table 12

Meteorological Data
Control Room χ/Q (sec/m^3)
for
LOCA - Containment Leak (CL)
LOCA - ECCS Leak
MSLB - Faulted Steam Generator (FSG)
Fuel Handling Accident

Byron Station

Time (hr)	LOCA/CL	LOCA/ECCS and FHA	MSLB/FSG
0-2	6.10E-3	2.28E-3	1.70E-2
2-8	5.30E-3	1.91E-3	1.46E-2
8-24	2.68E-3	8.88E-4	7.24E-3
24-96	2.30E-3	5.97E-4	4.89E-3
96-720	1.53E-3	4.77E-4	3.58E-3

Braidwood Station

Time (hr)	LOCA/CL	LOCA/ECCS and FHA	MSLB/FSG
0-2	6.20E-3	2.48E-3	1.68E-2
2-8	5.37E-3	1.87E-3	1.44E-2
8-24	2.79E-3	8.11E-4	6.53E-3
24-96	1.82E-3	5.04E-4	4.47E-3
96-720	1.32E-3	3.91E-4	2.96E-3

Table 13

Meteorological Data
Control Room χ/Q (sec/m^3)

MSLB/Intact steam generator
SGTR
Locked rotor accident with failed PORV

Time (hr)	Byron	Braidwood
0 to 0.083	8.79E-3	8.71E-3
0.083 to 2	3.98E-3	4.08E-3
2 to 8	3.48E-3	3.43E-3
8 to 24	1.64E-3	1.69E-3
24-96	1.04E-3	9.78E-4
96-720	8.96E-4	6.56E-4

Table 14

Meteorological Data
Control Room χ/Q (sec/m^3)

Rod Ejection Accident - Containment leak (CL)
Rod Ejection Accident -Stem release (SL)

	Byron		Braidwood	
Time (hr)	REA/CL	REA/SL	REA/CL	REA/SL
0 to 0.0417	9.82E-2	8.79E-3	9.53E-2	8.71E-3
0.0417 to 2	6.10E-3	3.98E-3	6.20E-3	4.08E-3
2 to 8	5.30E-3	3.48E-3	5.37E-3	3.43E-3
8 to 24	2.68E-3	1.64E-3	2.79E-3	1.69E-3
24-96	2.00E-3	1.04E-3	1.82E-3	9.78E-4
96-720	1.53E-3	8.96E-4	1.32E-3	6.56E-4