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Gentlemen:

DOCKETS 50-266 AND 50-301
SUPPLEMENT TO TECHNICAL SPECIFICATIONS
CHANGE REQUESTS 188 AND 189
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

This letter provides additional information in support of Technical Specifications Change Requests (TSCRs) 188 and 189. TSCRs 188 and 189 were submitted in letters dated June 4, 1996. Supplements to the TSCRs have been submitted in letters dated August 5, 1996, September 26, 1996, October 21, 1996, November 13, 1996, November 20, 1996, and December 2, 1996. These requests propose amendments to the Point Beach Technical Specifications that were identified by analyses performed in support of Unit 2 operations following replacement of steam generators this fall.

We are providing additional information regarding the analyses of radiological consequences for the Steam Generator Tube Rupture, Rupture of a Steam Pipe, Locked Rotor, and Rod Ejection accidents as attachments to this letter. It was also determined that an inconsistency between these new radiological analyses and the Technical Specifications had been introduced. This inconsistency involves the use of two different standards for determining dose equivalent I-131. The thyroid dose conversion factors used in the new analyses are based on Table 2.1 of Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," September 1988. The Point Beach Technical Specifications thyroid dose conversion factors are based on Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." To correct the inconsistency, we propose to change the Technical Specifications to use Table 2.1 of Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," September 1988. This also requires that the current Technical Specification limits on dose equivalent (DE) I-131 be reduced by approximately 20%. The changes associated with this correction are provided as an attachment to this letter.

Two additional corrections include removal of the nominal pressure setting for high-pressure trip from Technical Specifications basis 15.2.2 and removal of unused references in Technical Specifications section 15.5.3. The pressure setting for high-pressure trip is provided in the applicable Technical Specifications section 15.2.3 and this redundant listing is not necessary. The unused references should have been removed during previous Technical Specification changes that eliminated the need for these references.

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We have determined that the additional information and corrections do not involve a significant hazards consideration, authorize a significant change in the types or total amounts of any effluent release, or result in any significant increase in individual or cumulative occupational exposure. Therefore, we conclude that the proposed amendments meet the requirements of 10 CFR 51.22(c)(9) and that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared. The original "No Significant Hazards" determinations for operation under the proposed Technical Specifications remain applicable.

If you require additional information, please contact us.

Sincerely,

Gary M. Krieser
Manager, Strategic Issues
Nuclear Power

cc: NRC Resident Inspector
NRC Regional Administrator
PSCW

Subscribed and sworn before me on
this 16th day of January, 1997.

Jacqueline Gouch
Notary Public, State of Wisconsin

My commission expires 10/26/2000.

ATTACHMENT 1
ADDITIONAL INFORMATION
TECHNICAL SPECIFICATIONS CHANGE REQUESTS 188 AND 189

Introduction

It has been determined that an inconsistency between new radiological analyses performed in support of these TSCRs and the Technical Specifications has been introduced. This inconsistency involves the use of two different standards for determining dose equivalent (DE) I-131. The thyroid dose conversion factors used in the new analyses are based on Table 2.1 of Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," September 1988. The Point Beach Technical Specifications thyroid dose conversion factors are based on Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." To correct the inconsistency, we propose to change the Technical Specifications to use Table 2.1 of Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," September 1988. This also requires that the current Technical Specification limits on DE I-131 be reduced by approximately 20%.

Two additional corrections include removal of the nominal pressure settings from Technical Specifications basis 15.2.2 and removal of unused references in Technical Specifications section 15.5.3. The high-pressure trip setting is provided in the applicable Technical Specifications and this redundant listing is not necessary. The unused references should have been removed during previous Technical Specification changes that eliminated the need for these references.

The edited Technical Specifications pages associated with these additional changes are provided in attachment 3.

Description of Changes

1. Change TS 15.1.o to reference Table 2.1 of Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," September 1988 in place of Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." This change is necessary to correct an inconsistency between the new radiological analyses as previously submitted for this Technical Specifications change request.

2. Change the basis of TS 15.2.2 to remove the nominal pressure setting for reactor high-pressure trip,. The pressure setting for high-pressure trip is provided in the applicable Technical Specifications section 15.2.3.

3. Change 1.0 microcuries per gram in TS 15.3.1.C.1, 15.3.1.C.1.a, and 15.3.1.C.1.b to 0.8 microcuries per gram. This is necessary because the thyroid dose conversion factors used in the new analyses are based on Table 2.1 of Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," September 1988. The Point Beach Technical Specifications thyroid dose conversion factors are based on Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," as described in change 1. above. This change causes the DE I-131 limit to be reduced by approximately 20%, based on the differences in dose conversion factors between these two standards.

4. Change 1.0 microcuries per gram in the basis of TS 15.3.1 to 0.8 microcuries per gram. This is necessary because the thyroid dose conversion factors used in the new analyses are based on Table 2.1 of Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," September 1988. The Point Beach Technical Specifications thyroid dose conversion factors are based on Table III of

TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," as described in change 1. above. This change causes the DE I-131 limit to be reduced by approximately 20%, based on the differences in dose conversion factors between these two standards.

5. Change Figure 15.3.1-5 such that the limit is reduced by approximately 20%. This is necessary because the thyroid dose conversion factors used in the new analyses are based on Table 2.1 of Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," September 1988. The Point Beach Technical Specifications thyroid dose conversion factors are based on Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," as described in change 1. above. This change causes the DE I-131 limit to be reduced by approximately 20%, based on the differences in dose conversion factors between these two standards.

6. Change TS 15.3.4.B to limit the secondary coolant activity to 1.0 μCi/g of dose equivalent I-131. The change in thyroid dose conversion factors causes the DE I-131 limit to be reduced by approximately 20%, based on the differences in dose conversion factors between the two standards. The change in the units from μCi/cc to μCi/g is also necessary to provide consistency between the new analyses and the Technical Specifications.

7. Change the basis for Technical Specifications section 15.3.4 to establish consistency between the new analyses and this basis.

8. Change 1.0μCi/gram to 0.8μCi/gram in Technical Specifications Table 15.4.1-1 item 1. This is necessary because the thyroid dose conversion factors used in the new analyses are based on Table 2.1 of Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," September 1988. The Point Beach Technical Specifications thyroid dose conversion factors are based on Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," as described in change 1. above. This change causes the DE I-131 limit to be reduced by approximately 20%, based on the differences in dose conversion factors between these two standards.

9. Change 1.2μCi/gram to 1.0μCi/gram in Technical Specifications Table 15.4.1-1 item 8. This is necessary because the thyroid dose conversion factors used in the new analyses are based on Table 2.1 of Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," September 1988. The Point Beach Technical Specifications thyroid dose conversion factors are based on Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," as described in change 1. above. This change causes the DE I-131 limit to be reduced by approximately 20%, based on the differences in dose conversion factors between these two standards.

10. Delete references (2), (3), and (5) from Technical Specifications section 15.5.3. The unused references should have been removed during previous Technical Specification changes that eliminated the need for these references.

11. Change 1.0 microcuries per gram to 0.8 microcuries per gram in TS 15.6.9.B.2.e. This is necessary because the thyroid dose conversion factors used in the new analyses are based on Table 2.1 of Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," September 1988. The Point Beach Technical Specifications thyroid dose conversion factors are based on Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," as described in change 1. above. This change causes the DE I-131 limit to be reduced by approximately 20%, based on the differences in dose conversion factors between these two standards.

Radiological Analyses

The summary and results of the analyses that support these Technical Specifications change requests were provided as an attachment to our letter dated December 2, 1996. These analyses have been revised to account for the change in thyroid dose conversion factors and the revised analyses are provided as a separate attachment. It was also determined that the power level used to determine the source term was actually based on the uprated power level of 1650 MWt, not 102% of 1650 MWt as stated in the previous submittal of this information. This has also been corrected.

In a conference call on December 9, 1996, Wisconsin Electric was informed by NRC radiological review branch personnel that the use of potassium iodide (KI) for control room dose reduction is not in accordance with current NRC expectations. The use of KI for control room dose reduction at PBNP was previously accepted by the NRC in a safety evaluation transmitted to Wisconsin Electric via letter dated August 10, 1982.

The radiological analyses for control room habitability provided in this submittal continue to account for the use of KI to provide control room dose reduction. The revised radiological analyses and additional information supporting the radiological analyses are provided in Attachment 2.

We are in the process of identifying alternatives that can be used to reduce the control room thyroid dose to ≤30 rem without the use of KI. We are also in the process of reanalyzing the environmental consequences of the loss of coolant accident for Technical Specifications change request 192. We expect to submit the results of loss of coolant radiological analysis and propose a plan for resolving the control room KI issue as a supplement to Technical Specifications change request 192.

ATTACHMENT 2
REVISED RADIOLOGICAL ANALYSES AND ADDITIONAL INFORMATION
TECHNICAL SPECIFICATIONS CHANGE REQUESTS 188 AND 189

A request for additional information for Technical Specification Change requests 188 and 189 was transmitted to Wisconsin Electric in a letter dated November 13, 1996. Part B of the request states the following:

> Analyze the following accidents: (1) steam generator tube rupture, (2) control rod ejection, (3) locked rotor, and (4) main steamline break. Submit a copy of your accident analysis report that contains the calculated control room, EAB and LPZ doses and a comparison of the results to the 10 CFR 50 and 100 acceptance criteria. Also provide sufficient documentation of your analyses to support an independent evaluation, such as code inputs, the references for code inputs (e.g. UFSAR or new docketed analyses), and the dose calculations.

The requested information was provided in a letter dated December 2, 1996. This information is revised as follows to account for the use of thyroid dose conversion factors consistent with ICRP 30 per Table 2.1 of Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," September 1988:

Radiological Consequences of a Locked Rotor Accident

Introduction

An instantaneous seizure of a reactor coolant pump rotor is assumed to occur which rapidly reduces flow through the affected reactor coolant loop. Fuel clad damage is assumed to occur as a result of this accident. Due to the pressure differential between the primary and secondary systems, and assumed SG tube leaks, fission products are discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through either the atmospheric dump valves (ADV) or safety valves (MSSVs). In addition, some of the iodine activity contained in the secondary coolant prior to the accident is released to atmosphere as a result of steaming of the SGs following the accident. This section describes the assumptions and analyses performed to determine the amount of radioactivity released and the offsite and control room doses resulting from this release.

Input Parameters and Assumptions

The analysis of the locked rotor event radiological consequences uses the analytical methods and assumptions outlined in the Standard Review Plan (Reference 1).

The uprated power level of 1650 MWt is used in the analysis. For the pre-accident iodine spike it is assumed that a reactor transient has occurred prior to the locked rotor and has raised the RCS iodine concentration to 50 μCi/gm of dose equivalent (DE) I-131. Since fuel failure is assumed for this accident, it is not necessary to also assume an accident initiated spike, as is the case for events without fuel failure such as a SGTR or a MSLB. The noble gas activity concentration in the RCS at the time the accident occurs is based on a fuel defect level of 1.0%. This is approximately equal to the Technical Specification value of 100/E bar μCi/gm for gross radioactivity. The iodine activity concentration of the secondary coolant at the time the locked rotor occurs is assumed to be equivalent to the proposed Technical Specification limit of 1.0 μCi/gm of DE I-131.

In determining the offsite and control room doses following the locked rotor, it is conservatively assumed that 100% of the fuel rods in the core suffer sufficient damage that all of their gap activity is released to the RCS. Ten percent of the total core activity for both iodines and noble gases is assumed to be in the fuel-cladding gap (Reference 2).

The total primary to secondary SG tube leak rate used in the analysis is the Technical Specification limit of 0.35 gpm per steam generator or 0.70 gpm total. No credit for iodine removal is taken for any steam released to the condenser prior to reactor trip and concurrent loss of offsite power. An iodine partition factor in the SGs of 0.01 (curies I /gm steam) / (curies I /gm water) is used (Reference 3). All noble gas activity carried over to the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere.

At 8 hours after the accident the RHR System is assumed to be placed into service for heat removal, and there are no further steam releases to atmosphere from the secondary system.

The thyroid dose conversion factors, breathing rates, and atmospheric dispersion factors used in the dose calculations are given in Table 1. The core and coolant activities used in the radiological calculations are given in Table 2. The parameters associated with the control room HVAC modes are summarized in Table 3. The remaining major assumptions and parameters used specifically in the locked rotor analysis are itemized in Table 4.

Control Room Model

The Point Beach control room HVAC system operates in one of four modes. Mode 1 is the normal HVAC mode, in which 5% of the air flow is outside air and 95% is recirculated air. Mode 2, which consists of 100% recirculated unfiltered air within the control room, is initiated either by a containment isolation signal or manually from the control room. Mode 3 is initiated manually by operator action and allows for filtered recirculated airflow. Mode 4 is initiated either by a control room radioactivity signal or manually by operator action. In this mode, 25% of the available flow is made up with filtered outside air while the remaining 75% air flow is unfiltered recirculation. The parameters associated with the control room HVAC modes are summarized in Table 3. These parameters have been taken from Reference 4. In addition, a factor of 10 reduction to the thyroid dose is allowed with the use of Potassium Iodide pills by the control room operators.

For the locked rotor accident it is assumed that the HVAC system begins in Mode 1. A containment isolation signal is never received so Mode 2 is not modeled for this accident scenario. The dose rates in the control room trip the control room monitors within 30 minutes, switching the system to Mode 4 where it remains throughout the event. The control room doses are calculated over a period of twenty-four hours to ensure that the largest doses to the control room operator are calculated since the ventilation system will continue to operate in the specified mode for several hours following the termination of the steam releases.

Description of Analyses

The analysis of the locked rotor event radiological consequences uses the analytical methods and assumptions outlined in the Standard Review Plan (Reference 1). Because fuel failure is assumed, only a pre-accident iodine spike is assumed, rather than both pre-accident and accident initiated spikes, as is the case for events without fuel failure.

Acceptance Criteria

The dose limits for a locked rotor are a "small fraction of" the 10 CFR 100 guidelines values. A "small fraction of" is considered 10% of 10 CFR 100 guideline values, or 30 rem thyroid and 2.5 rem γ-body. The criteria defined in SRP Section 6.4 (Reference 5) are used for the control room dose limits: 30 rem thyroid, 5 rem whole body and 30 rem beta skin.

Results

The offsite and control room thyroid, γ-body, and beta skin doses due to the locked rotor event are given in Table 5.

Conclusions

The offsite thyroid and gamma body doses due to the locked rotor accident are within the acceptance criteria. The control room gamma body and beta skin doses are within the current NRC acceptance criteria for the control room. The 24 hour control room thyroid dose exceeds the 30 rem limit; however, assuming that the operators would be instructed to take the potassium iodide pills, the control room thyroid dose would be reduced to approximately 6.5 rem which is within the 30 rem limit.

Areas of Conservatism with respect to the Steam Generator Replacement Project

This analysis was performed to support both the Steam Generator Replacement Project and the fuel upgrade/uprating program for Point Beach Units 1 and 2. As such, there are several areas in which a more conservative or bounding value was used to support the fuel upgrade or uprating which would not be necessary to support the replacement steam generators alone. This section describes the conservatisms incorporated into the locked rotor accident radiological analysis with respect to the Steam Generator Replacement Project.

The steam releases for the locked rotor accident were calculated using the increased power level of 1650 MWt, a higher Tavg of 580°F, and an RCS pressure of 2250 psi. The steam releases calculated with these parameters bound the steam releases which would correspond to a power level of 1520 MWt, a Tavg of 573.9°F and an RCS pressure of 2000 or 2250 psi. Further, the source term calculations were performed to incorporate the increased core thermal power level and fuel upgrade parameters. The upgraded fuel includes an increase in the mass of the fuel and enrichment which results in an increase to several isotopes in the core and coolant activities. The number of fuel rods assumed to suffer sufficient damage to release all of their gap fraction was increased from 86% to 100%. For the current fuel cycle, only 86% of rods are calculated to enter DNB. Additionally, the locked rotor accident was analyzed using the analytical methods and assumptions outlined in the Standard Review Plan (Reference 1). Specifically, this means the accident also incorporated a pre-accident iodine spike equivalent to 50 μCi/gm of DE I-131 in the coolant activities.

References

1. NUREG-0800, Standard Review Plan 15.3.3, 15.3.4, Revision 2, "Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break", July 1981.

2. US AEC Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors", May 1974.

3. NUREG-0800, Standard Review Plan 15.6.3, "Radiological Consequences of a Steam Generator Tube Rupture (PWR)", Rev. 2, July 1981.

4. Wisconsin Electric letter to NRC, VPNPD-96-099, "Supplement to Technical Specifications Change Requests 188 and 189 Point Beach Nuclear Plant Units 1 and 2," Bob Link, November 20, 1996.

5. NRC SRP Section 6.4, "Control Room Habitability System", Rev 2, July 1981, NUREG-0800.

Radiological Consequences of a Rod Ejection Accident

Introduction

It is assumed that a mechanical failure of a control rod mechanism pressure housing has occurred, resulting in the ejection of a rod cluster control assembly and drive shaft. As a result of the accident fuel clad damage and a small amount of fuel melt are assumed to occur. Due to the pressure differential between the primary and secondary systems, radioactive reactor coolant is discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through either the main condenser or the atmospheric dump valves (ADV)/ safety valves (MSSVs). Some of the iodine activity contained in the secondary coolant prior to the accident is released to atmosphere as a result of steaming of the SGs following the accident. Additionally, radioactive reactor coolant is discharged to the containment via the spill from the reactor vessel head. A portion of this radioactivity is released through containment leakage to the environment. This section describes the assumptions and analyses performed to determine the amount of radioactivity released and the offsite and control room doses resulting from these releases.

Input Parameters and Assumptions

The offsite and control room doses following a rod ejection accident are determined using present-day NRC regulatory requirements. This includes taking into account a pre-accident iodine spike. For the pre-accident iodine spike it is assumed that a reactor transient has occurred prior to the rod ejection and has raised the RCS iodine concentration to 50 μCi/gm of dose equivalent (DE) I-131. The noble gas activity concentration in the RCS at the time the accident occurs is based on a fuel defect level of 1.0%. This is approximately equal to the Technical Specification value of 100/E bar μCi/gm for gross radioactivity. The iodine activity concentration of the secondary coolant at the time the rod ejection accident occurs is assumed to be equivalent to the Technical Specification limit of 1.0 μCi/gm of DE I-131.

As a result of the rod ejection accident less than 10% of the fuel rods in the core undergo cladding damage. In determining the offsite and control room doses following rod ejection accident, it is conservatively assumed that 10% of the fuel rods in the core suffer sufficient damage that all of their gap activity is released to the RCS. Ten percent of the total core activity for both iodines and noble gases is assumed to be in the fuel-cladding gap (Reference 1).

A small fraction (i.e., 0.25%) of the fuel in the core is assumed to melt as a result of the rod ejection accident. One-half of the iodine activity in the melted fuel is released to the RCS, while all of the noble gas activity in the melted fuel is released to the RCS.

Conservatively, all the iodine and noble gas activity (from prior to the accident and resulting from the accident) is assumed to be in the RCS when determining offsite and control room doses due to the primary to secondary SG tube leakage, and all of the iodine and noble gas activity is assumed to be in the containment when determining offsite and control room doses due to containment leakage. However, 50% of the iodine activity released to the containment is assumed to instantaneously plate out on containment surfaces.

The total primary to secondary SG tube leak rate used in the analysis is the Technical Specification limit of 0.35 gpm per steam generator or 0.70 gpm total. Primary and secondary system pressure are equalized after 1500 seconds, thus terminating primary to secondary leakage in the SGs. No credit for iodine removal is taken for any steam released to the condenser prior to reactor trip and concurrent loss of offsite power. An iodine partition factor in the SGs of 0.01 curies/gm steam/curies/gm water is used (Reference 2). All noble gas activity carried over to the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere.

Steam release of the initial secondary coolant activity from the SGs following the rod ejection accident is based on the maximum relief rate of 6.664E6 lb/hr through the main steam safety valves and a steam release duration of 86 seconds. This results in a steam release of 158,200 lb.

The Technical Specification design basis containment leak rate of 0.4% by weight of containment air is used for the initial 24 hours. Thereafter the containment leak rate is assumed to be one-half the design value, or 0.2%/day (Reference 1). The thyroid dose conversion factors, breathing rates, and atmospheric dispersion factors used in the dose calculations are given in Table 1. The core and coolant activities used in the dose calculations are given in Table 2. The parameters associated with the control room HVAC modes are summarized in Table 3. The remaining major assumptions and parameters used specifically in the rod ejection analysis are itemized in Table 6.

Control Room Model

The Point Beach control room HVAC system operates in one of four modes. Mode 1 is the normal HVAC mode, in which 5% of the air flow is outside air and 95% is recirculated air. Mode 2, which consists of 100% recirculated unfiltered air within the control room, is initiated either by a containment isolation signal or manually from the control room. Mode 3 is initiated manually by operator action and allows for filtered recirculated airflow. Mode 4 is initiated either by a control room radioactivity signal or manually by operator action. In this mode, 25% of the available flow is made up with filtered outside air while the remaining 75% air flow is unfiltered recirculation. The parameters associated with the control room HVAC modes are summarized in Table 3. These parameters have been taken from Reference 3. In addition, a factor of 10 reduction to the thyroid dose is allowed with the use of Potassium Iodide pills by the control room operators.

For the rod ejection accident it is assumed that the HVAC system begins in Mode 1. On containment isolation, the system is automatically shifted to Mode 2 which would occur within 5 minutes of event initiation. When the dose rates in the control room exceed the high radiation alarm setpoint, the system is automatically shifted to Mode 4 where it remains throughout the event. The switch to Mode 4 will occur within 30 minutes. For simplicity, the analysis models a shift from Mode 1 to Mode 4 at 30 minutes without switching to Mode 2.

Description of Analyses Performed

The analysis of the rod ejection event radiological consequences uses the analytical methods and assumptions outlined in Regulatory Guide 1.77 (Reference 1). Because fuel failure is assumed, only a pre-accident iodine spike is assumed, rather than both pre-accident and accident initiated spikes, as is the case for events without fuel failure.

Acceptance Criteria

The offsite dose limits for a rod ejection accident are "well within" the 10 CFR 100 guideline values, or 75 rem thyroid and 6 rem whole body (Reference 4). The criteria defined in SRP Section 6.4 (Reference 5) will be used for the control room dose limits: 30 rem thyroid, 5 rem whole body and 30 rem beta skin.

Results

The offsite and control room thyroid, γ-body, and beta skin doses due to the rod ejection accident are given in Table 7.

Conclusions

The offsite thyroid and whole body doses and the control room whole body and beta skin doses are within the current NRC acceptance criteria for a rod ejection accident. The control room thyroid dose exceeds the 30 rem limit; however, assuming that the operators would be instructed to take the potassium iodide pills, the control room thyroid dose would be reduced to approximately 12 rem which is within the 30 rem limit.

Areas of Conservatism with respect to the Steam Generator Replacement Project

This analysis was performed to support both the Steam Generator Replacement Project and the fuel upgrade/uprating program for Point Beach Units 1 and 2. As such, there are several areas in which a more conservative or bounding value was used to support the fuel upgrade or uprating which would not be necessary to support the replacement steam generators alone. This section describes the conservatisms incorporated into the rod ejection accident radiological analysis with respect to the Steam Generator Replacement Project.

The source term calculations were performed to incorporate the increased core thermal power level and fuel upgrade parameters. The upgraded fuel includes an increase in the mass of the fuel and enrichment which results in an increase to many isotopes in the core and coolant activities. Additionally, the rod ejection accident was analyzed using the analytical methods and assumptions outlined in Reg Guide 1.77 (Reference 1). Specifically, this means the accident also incorporated a pre-accident iodine spike equivalent to 50 μCi/gm of DE I-131 in the coolant activities.

References

1. US AEC Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors", May 1974.

2. NUREG-0800, Standard Review Plan 15.6.3, "Radiological Consequences of a Steam Generator Tube Rupture (PWR)", Rev. 2, July 1981.

3. Wisconsin Electric letter to NRC, VPNPD-96-099, "Supplement to Technical Specifications Change Requests 188 and 189 Point Beach Nuclear Plant Units 1 and 2," Bob Link, November 20, 1996.

4. NUREG-0800, Standard Review Plan 15.4.8, Revision 2, "Spectrum of Rod Ejection Accidents (PWR)", July 1981.

5. NRC SRP Section 6.4, "Control Room Habitability System", Rev 2, July 1981, NUREG-0800.

Radiological Consequences of a Steam Generator Tube Rupture Accident

Introduction

For the SGTR event, the complete severance of a single steam generator tube is assumed to occur. Due to the pressure differential between the primary and secondary systems, radioactive reactor coolant is discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through either the main condenser, the atmospheric dump valves (ADV) or safety valves (MSSVs). In addition, some of the iodine activity contained in the secondary coolant prior to the accident is released to atmosphere as a result of steaming of the SGs following the accident. This section

describes the assumptions and analyses performed to determine the amount of radioactivity released and the offsite and control room doses resulting from this release.

Input Parameters and Assumptions

The analysis of the steam generator tube rupture (SGTR) radiological consequences uses the analytical methods and assumptions outlined in the Standard Review Plan (Reference 1).

The uprated power level of 1650 MWt is used in the analysis. For the pre-accident iodine spike it is assumed that a reactor transient has occurred prior to the SGTR and has raised the RCS iodine concentration to 50 μCi/gm of dose equivalent (DE) I-131. For the accident initiated iodine spike, the reactor trip associated with the SGTR creates an iodine spike in the RCS which increases the iodine release rate from the fuel to the RCS to a value 500 times greater than the release rate corresponding to the maximum proposed equilibrium RCS Technical Specification concentration of 0.8 μCi/gm of DE I-131. The duration of the accident initiated iodine spike is 1.6 hours.

The noble gas activity concentration in the RCS at the time the accident occurs is based on a fuel defect level of 1.0%. This is approximately equal to the Technical Specification value of 100/E bar μCi/gm for gross radioactivity. The iodine activity concentration of the secondary coolant at the time the SGTR occurs is assumed to be equivalent to the proposed Technical Specification limit of 1.0 μCi/gm of DE I-131. The amount of primary to secondary SG tube leakage in the intact SG is assumed to be equal to the Technical Specification limit for a single SG of 0.35 gpm. No credit for iodine removal is taken for any steam released to the condenser prior to reactor trip and concurrent loss of offsite power. An iodine partition factor in the SGs of 0.01 (curies I/gm steam) / (curies I/gm water) is used (Reference 1). All noble gas activity carried over to the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere.

Flow through the ruptured SG tube is assumed to be terminated at 30 minutes following accident initiation due to operator action. Eight hours after the accident the RHR System is assumed to be placed into service for heat removal, and there are no further steam releases to atmosphere from the secondary system.

The thyroid dose conversion factors, breathing rates, and atmospheric dispersion factors used in the dose calculations are given in Table 1. The core and coolant activities used in the dose calculations are given in Table 2. The parameters associated with the control room HVAC modes are summarized in Table 3. The remaining major assumptions and parameters used specifically in this analysis are itemized in Table 8.

Control Room Model

The Point Beach control room HVAC system operates in one of four modes. Mode 1 is the normal HVAC mode, in which 5% of the air flow is outside air and 95% is recirculated air. Mode 2, which consists of 100% recirculated unfiltered air within the control room, is initiated either by a containment isolation signal or manually from the control room. Mode 3 is initiated manually by operator action and allows for filtered recirculated airflow. Mode 4 is initiated either by a control room radioactivity signal or manually by operator action. In this mode, 25% of the available flow is made up with filtered outside air while the remaining 75% air flow is unfiltered recirculation. The parameters associated with the control room HVAC modes are summarized in Table 3. These parameters have been taken from Reference 2. In addition, a factor of 10 reduction to the thyroid dose is allowed with the use of Potassium Iodide pills by the control room operators.

For the steam generator tube rupture accident it is assumed that the HVAC system begins in Mode 1. On containment isolation, which is conservatively assumed to begin 10 minutes after event initiation, the system is automatically shifted to Mode 2. When the dose rates in the control room exceed one of the

radiation alarm setpoints, the system is automatically shifted to Mode 4 and remains there until the radiation release has ended. The shift to Mode 4 will occur within 30 minutes. In order to bound the total dose received by the operators, the radiological consequences were calculated for both Mode 2 and Mode 4 separately. The first case has a shift from Mode 1 to Mode 2 after 10 minutes, and remaining in Mode 2 throughout the event. The second case has a shift from Mode 1 to Mode 4 after 30 minutes, then remaining in Mode 4 for the rest of the event. The control room doses are calculated over a period of twenty-four hours to ensure that the largest doses to the control room operator are calculated since the ventilation system will continue to operate in the specified modes for several hours following the termination of the steam releases.

Description of Analyses

The analysis of the steam generator tube rupture (SGTR) radiological consequences uses the analytical methods and assumptions outlined in the Standard Review Plan (Reference 1) to calculate the activity release to atmosphere from the ruptured and intact SGs and the resulting offsite and control room doses. Both the pre-accident iodine spike and accident initiated iodine spike models are analyzed for these release paths.

Acceptance Criteria

The offsite dose limits for a SGTR with a pre-accident iodine spike are the guideline values of 10 CFR 100 (Reference 1). These guideline values are 300 rem thyroid and 25 rem γ-body. For a SGTR with an accident initiated iodine spike, the acceptance criteria are a "small fraction of" the 10 CFR 100 guideline values, or 30 rem thyroid and 2.5 rem γ-body. The criteria defined in SRP Section 6.4 (Reference 3) will be used for the control room dose limits: 30 rem thyroid, 5 rem whole body and 30 rem beta skin.

Results

The offsite and control room thyroid, γ-body, and beta skin doses due to the SGTR accident are given in Table 9. The results of both control room models are included.

Conclusions

The offsite and control room doses due to the SGTR are within the acceptance criteria.

Areas of Conservatism with respect to the Steam Generator Replacement Project

This analysis was performed to support both the Steam Generator Replacement Project and the fuel upgrade/uprating program for Point Beach Units 1 and 2. A similar analysis was also performed to support the Steam Generator Replacement Project; however, since additional calculations are necessary to address control room habitability issues the fuel upgrade/uprate SGTR analysis which is the most recent analysis is being used to support the Steam Generator Replacement Project at this time. As such, there are several areas in which a more conservative or bounding value was used to support the fuel upgrade or uprating which would not be necessary to support the replacement steam generators alone. This section describes the conservatisms incorporated into the steam generator tube rupture accident radiological analysis with respect to the Steam Generator Replacement Project.

The thermal and hydraulic SGTR calculations were performed for both the replacement steam generator and the fuel upgrade/uprate programs. The primary to secondary break flow and the steam releases to the atmosphere for the fuel upgrade uprate are approximately 7% higher than those calculated for the Steam Generator Replacement Project. These higher break flow and steam releases were used to calculate the offsite and control room doses for the fuel upgrade/uprate program. In addition, the source term calculations were performed to incorporate the increased core thermal power level and fuel upgrade

parameters. The upgraded fuel includes an increase in the mass of the fuel and enrichment which results in an increase to many isotopes in the core and coolant activities. Additionally, the steam generator tube rupture accident was analyzed using the analytical methods and assumptions outlined in the Standard Review Plan (Reference 1). Specifically, this means the accident incorporated both an accident initiated and a pre-accident iodine spike in the coolant activities; however, as consistent with the Standard Review Plan, the steam generator partition coefficient was reduced from 0.1 presented in the Point Beach FSAR to 0.01.

References

1. NUREG-0800, Standard Review Plan 15.6.3, "Radiological Consequences of a Steam Generator Tube Rupture (PWR)", Rev. 2, July 1981.

2. Wisconsin Electric letter to NRC, VPNPD-96-099, "Supplement to Technical Specifications Change Requests 188 and 189 Point Beach Nuclear Plant Units 1 and 2," Bob Link, November 20, 1996.

3. NRC SRP Section 6.4, "Control Room Habitability System", Rev 2, July 1981, NUREG-0800.

Radiological Consequences of a Steamline Break Accident

Introduction

The complete severance of a main steamline outside containment is assumed to occur. The affected SG will rapidly depressurize and release radioiodines initially contained in the secondary coolant and primary coolant activity, transferred via SG tube leaks, directly to the outside atmosphere. A portion of the iodine activity initially contained in the intact SGs and noble gas activity due to tube leakage is released to atmosphere through either the atmospheric dump valves (ADV) or the safety valves (MSSVs). This section describes the assumptions and analyses performed to determine the amount of radioactivity released and the offsite and control room doses resulting from this release.

Input Parameters and Assumptions

The analysis of the steam line break (SLB) radiological consequences uses the analytical methods and assumptions contained in the Standard Review Plan (Reference 1).

The uprated power level of 1650 MWt is used in the analysis. For the pre-accident iodine spike it is assumed that a reactor transient has occurred prior to the SLB and has raised the RCS iodine concentration to 50 μCi/gm of dose equivalent (DE) I-131. For the accident initiated iodine spike the reactor trip associated with the SLB creates an iodine spike in the RCS which increases the iodine release rate from the fuel to the RCS to a value 500 times greater than the release rate corresponding to the maximum proposed equilibrium RCS Technical Specification concentration of 0.8 μCi/gm of DE I-131. The duration of the accident initiated iodine spike is 1.6 hours.

The noble gas activity concentration in the RCS at the time the accident occurs is based on a fuel defect level of 1.0%. This is approximately equal to the Technical Specification value of 100/E bar μCi/gm for gross radioactivity. The iodine activity concentration of the secondary coolant at the time the SLB occurs is assumed to be equivalent to the proposed Technical Specification limit of 1.0 μCi/gm of DE I-131. The amount of primary to secondary SG tube leakage in each of the two SGs is assumed to be equal to the Technical Specification limit for a single SG of 0.35 gpm. No credit for iodine removal is taken for any steam released to the condenser prior to reactor trip and concurrent loss of offsite power.

The SG connected to the broken steamline is assumed to boil dry within the initial half hour following the SLB. The entire liquid inventory of this SG is assumed to be steamed off and all of the iodine initially in this SG is released to the environment. Also, iodine carried over to the faulted SG by SG tube leaks is assumed to be released directly to the environment with no credit taken for iodine retention in the SG.

An iodine partition factor in the intact SG of 0.01 (curies/gm steam)/(curies/gm water) is used (Reference 1). All noble gas activity carried over to the secondary through SG tube leakage is assumed to be immediately released to the outside atmosphere.

Eight hours after the accident, the RHR System is assumed to be placed into service for heat removal, and there are no further steam releases to atmosphere from the secondary system.

The thyroid dose conversion factors, breathing rates, and atmospheric dispersion factors used in the dose calculations are given in Table 1. The core and coolant activities used in the dose calculations are given in Table 2. The parameters associated with the control room HVAC modes are summarized in Table 3. The remaining major assumptions and parameters used specifically in this analysis are itemized in Table 10.

Control Room Model

The Point Beach control room HVAC system operates in one of four modes. Mode 1 is the normal HVAC mode, in which 5% of the air flow is outside air and 95% is recirculated air. Mode 2, which consists of 100% recirculated unfiltered air within the control room, is initiated either by a containment isolation signal or manually from the control room. Mode 3 is initiated manually by operator action and allows for filtered recirculated airflow. Mode 4 is initiated either by a control room radioactivity signal or manually by operator action. In this mode, 25% of the available flow is made up with filtered outside air while the remaining 75% air flow is unfiltered recirculation. The parameters associated with the control room HVAC modes are summarized in Table 3. These parameters have been taken from Reference 2. In addition, a factor of 10 reduction to the thyroid dose is allowed with the use of Potassium Iodide pills by the control room operators.

For the steam line break accident it is assumed that the HVAC system begins in Mode 1. On containment isolation, which is conservatively assumed to begin 5 minutes after event initiation, the system is automatically shifted to Mode 2. When the dose rates in the control room exceed one of the radiation alarm setpoints, the system is automatically shifted to Mode 4 and remains there until the radiation release has ended. The shift to Mode 4 will occur within 30 minutes. In order to bound the total dose received by the operators, the radiological consequences were calculated for both Mode 2 and Mode 4 separately. The first case has a shift from Mode 1 to Mode 2 after 5 minutes, and remaining in Mode 2 throughout the event. The second case has a shift from Mode 1 to Mode 4 after 30 minutes, then remaining in Mode 4 for the rest of the event. The control room doses are calculated over a period of twenty-four hours to ensure that the largest doses to the control room operator are calculated since the ventilation system will continue to operate in the specified modes for several hours following the termination of the steam releases.

Description of Analyses Performed

The analysis of the steam line break (SLB) radiological consequences uses the analytical methods and assumptions outlined in the Standard Review Plan (Reference 1). Both the pre-accident iodine spike and accident initiated iodine spike models are analyzed for these release paths.

Acceptance Criteria

The offsite dose limits for a SLB with a pre-accident iodine spike are the guideline values of 10 CFR 100. These guideline values are 300 rem thyroid and 25 rem γ-body. For a SLB with an accident initiated iodine spike the acceptance criteria are a "small fraction of" the 10 CFR 100 guideline values, or 30 rem thyroid and 2.5 rem γ-body. The criteria defined in SRP Section 6.4 (Reference 3) will be used for the control room dose limits: 30 rem thyroid, 5 rem whole body and 30 rem beta skin.

Results

The offsite and control room thyroid, γ-body, and beta skin doses due to the SLB are given in Table 11. The results of both control room models have been included.

Conclusions

The offsite thyroid and whole body doses are within the current NRC acceptance criteria for a steamline break accident. The control room whole body and beta skin doses are within the current NRC acceptance criteria for the control room. The control room thyroid dose exceeds the 30 rem limit for the cases in which the control room HVAC system continues to operate in Mode 2; however, assuming that the operators would be instructed to take the potassium iodide pills, the control room thyroid dose would be reduced to approximately 14 rem which is within the 30 rem limit. For the cases in which the control room HVAC system switches to Mode 4 within 30 minutes, the control room thyroid doses are within the NRC acceptance criteria.

Areas of Conservatism with respect to the Steam Generator Replacement Project

This analysis was performed to support both the Steam Generator Replacement Project and the fuel upgrade/uprating program for Point Beach Units 1 and 2. As such, there are several areas in which a more conservative or bounding value was used to support the fuel upgrade or uprating which would not be necessary to support the replacement steam generators alone. This section describes the conservatisms incorporated into the steamline break accident radiological analysis with respect to the Steam Generator Replacement Project.

The most significant impact to the steamline break accident is the use of the analytical methods and assumptions outlined in the Standard Review Plan (Reference 1). Specifically, this means the accident was modeled to include both a pre-accident and an accident initiated iodine spike in the coolant activities where the prior FSAR analysis was based on an equilibrium RCS activity equivalent to 1% fuel defects. The steam releases for the steamline break accident were calculated using the increased power level of 1650 MWt, a higher Tavg of 580°F, and an RCS pressure of 2250 psi. The steam releases calculated with these parameters bound the steam releases which would correspond to a power level of 1520 MWt, a Tavg of 573.9°F and an RCS pressure of 2000 or 2250 psi. In addition, the source term calculations were performed to incorporate the increased core thermal power level and fuel upgrade parameters. The upgraded fuel includes an increase in the mass of the fuel and enrichment which results in an increase to many isotopes in the core and coolant activities.

References

1. NUREG-0800, Standard Review Plan 15.1.5, Appendix, A, "Radiological Consequences of Main Steam Line Failures Outside of a PWR, Rev. 2, July 1981.

2. Wisconsin Electric letter to NRC, VPNPD-96-099, "Supplement to Technical Specifications Change Requests 188 and 189 Point Beach Nuclear Plant Units 1 and 2," Bob Link, November 20, 1996.

3. NRC SRP Section 6.4, "Control Room Habitability System", Rev 2, July 1981, NUREG-0800.

ADDITIONAL INFORMATION

Release Point Descriptions

Control Rod Ejection Accident: Two release points are assumed for this accident. For the source term in the reactor containment building, releases are assumed to occur through the reactor containment structure at its design leak rate. For the source term in the reactor coolant system, releases are assumed to occur through the tubes in the steam generator with subsequent release out through the safety relief valves or the atmospheric steam dump valve. The release point on the containment structure is assumed to be the point on the structure that is closest to the control room ventilation intake. This point was taken as the point on the containment structure that lies on the line that is drawn though the center of the containment structure and the control room ventilation intake. The release point for the safety relief valves or the atmospheric steam dump valve is assumed to be the vents for the valves of the steam generator that are closest to the control room ventilation intake. These valves exhaust through the top of the facade to the atmosphere at elevation 170 feet. The containment structure and the atmospheric steam dump and safety relief valve locations are shown on Figure 1.2-9, "Equipment Location - Plan," of the Point Beach Nuclear Plant (PBNP) Final Safety Analysis Report (FSAR). The atmospheric steam dump and the safety relief valves are located at elevation 88 feet inside the facade which is a sheet metal structure with side lengths of 126 feet by 132 feet.

Steam Generator Tube Rupture Accident: Two release points are assumed for this accident. For the source term in the reactor coolant system, releases are assumed to occur through the tubes in the steam generator with subsequent release out through the safety relief valves or the atmospheric steam dump valve for both the intact and the faulted steam generators. The release point for the safety relief valves or the atmospheric steam dump valve is assumed to be the vents for the valves. These valves exhaust through the top of the facade to the atmosphere at elevation 170 feet. For both release points, atmospheric dispersion factors were calculated. The most limiting factor was then used for both release points to evaluate the dose consequences of the releases. The containment structure and the atmospheric steam dump and safety relief valve locations are shown on Figure 1.2-9, "Equipment Location - Plan," of the Point Beach Nuclear Plant (PBNP) Final Safety Analysis Report (FSAR). The atmospheric steam dump and the safety relief valves are located at elevation 88 feet inside the facade which is a sheet metal structure with side lengths of 126 feet by 132 feet.

Loss of Reactor Coolant Flow Accident: Two release points are assumed for this accident. For the source term in the reactor coolant system, releases are assumed to occur through the tubes in the steam generator with subsequent release out through the safety relief valves or the atmospheric steam dump valve for both the intact and the faulted steam generators. The release point for the safety relief valves or the atmospheric steam dump valve is assumed to be the vents for the valves. These valves exhaust through the top of the facade to the atmosphere at elevation 170 feet. For both release points, atmospheric dispersion factors were calculated. The most limiting factor was then used for both release points to evaluate the dose consequences of the releases. The containment structure and the atmospheric steam dump and safety relief valve locations are shown on Figure 1.2-9, "Equipment Location - Plan," of the Point Beach Nuclear Plant (PBNP) Final Safety Analysis Report (FSAR). The atmospheric steam dump and the safety relief valves are located at elevation 88 feet inside the facade which is a sheet metal structure with side lengths of 126 feet by 132 feet.

Main Steam Line Break Accident: The steam pipe is assumed to rupture inside the facade which is considered a release direct to the environment rather than inside a building with an active ventilation system which could possibly delay the release to the atmosphere. The release point is assumed to be from the steam pipe at the facade and occurs in the steam pipe that lies closest to the control room ventilation intake. The steam pipe locations are shown on Figure 1.2-9, "Equipment Location - Plan," of the Point

Beach Nuclear Plant (PBNP) Final Safety Analysis Report (FSAR). The steam pipes are located at elevation 88 feet inside the facade which is a sheet metal structure with side lengths of 126 feet by 132 feet.

Distance Descriptions

The horizontal straight line distances to the control room ventilation intake from the release points are listed below.

Release Point	Straight Line Distance (m)	Notes
U2 Containment	31.1	The distance is from the point on the outside of the containment structure that lies closest to the ventilation intake. U2 containment structure is closer than U1 containment.
U1 Containment	46.0	Ibid.
U1 A Safeties	44.8	The distance is from the midpoint of the manifold pipe that contains the safeties and the atmosphere relief valve to the ventilation intake.
U1 B Safeties	54.3	Ibid.
U2 A Safeties	34.3	Ibid.
U2 B Safeties	35.5	Ibid.

Building Dimensions

The dimensions of the major building structures located around the location of the control room ventilation intake are shown on the attached drawing, PBC-231, Plot Plan. Note: Some distances on the drawing have been corrected after the distances were checked against drawings contained in Chapter 2 of the PBNP FSAR.

The containment structures are located inside the facade which is a sheet metal structure with side lengths of 126 feet by 132 feet. The smallest plane cross-sectional area presented by the containment building is 1640 square meters.

Control Room Emergency Filtration

The following is a summary of Technical Specifications requirements, test requirements, and calculation assumptions concerning control room emergency filtration.

Technical Specification Requirements [References 1 & 2]

The results of in-place cold DOP and halogenated hydrocarbon tests on HEPA filter and charcoal adsorber banks shall show a minimum of 99% DOP removal and 99% halogenated hydrocarbon removal.

The results of laboratory charcoal tests shall show a minimum of 90% removal of methyl iodide.

Point Beach Test Requirements [Reference 3]

If the methyl iodide efficiency falls below 95%, the charcoal adsorbers will be replaced with fresh ones and be retested in accordance with applicable Technical Specifications.

DOP and Freon tests must show >99% DOP and Freon removal.

Control Room Habitability Calculation Assumptions [Reference 4]

For the purposes of calculating particulate, elemental and methyl iodine removal in the control room habitability calculations, the following assumptions were made.

Filter Bank Iodine Removal Efficiencies

	Filter Removal Efficiency
Elemental	90%
Methyl	90%
Particulate	99%

References

1. Technical Specification 15.3.12 "Control Room Emergency Filtration."
2. Technical Specification 15.4.11 "Control Room Emergency Filtration."
3. Health Physics Implementing Procedure 11.54 "Control Room F-16 Filter Testing."
4. Wisconsin Electric Letter to the US NRC, VPNPD-96-099, dated November 20, 1996.

TABLE 1

DOSE CONVERSION FACTORS, BREATHING RATES AND ATMOSPHERIC DISPERSION FACTORS

Isotope	Thyroid Dose Conversion Factors [1] (rem/curie)	
I-131	1.07 E6	
I-132	6.29 E3	
I-133	1.81 E5	
I-134	1.07 E3	
I-135	3.14 E4	
Time Period	**Breathing Rate [2] (m^3/sec)**	
0-8 hr	3.47 E-4	
8-24 hr	1.75 E-4	
24-720 hr	2.32 E-4	
Site Boundary	**Atmospheric Dispersion Factors [3] (sec/m^3)**	
0-2 hr	5.0 E-4	
Low Population Zone		
0-8 hr	3.0 E-5	
8-24 hr	1.6 E-5	
24-96 hr	4.2 E-6	
96-720 hr	8.6 E-7	
Control Room	**Release from Containment [4]**	**Release from Safety Valves [4]**
0-8 hr	2.1 E-3	1.9 E-3
8-24 hr	1.3 E-3	1.3 E-3
24-96 hr	8.3 E-4	7.6 E-4
96-720 hr	3.3 E-4	2.9 E-4

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[1] ICRP Publication 30

[2] Regulatory Guide 1.4

[3] Wisconsin Electric letter VPNPD-96-099

[4] The rod ejection and MSLB release is from containment, the SGTR and locked rotor release is from the safety valves.

TABLE 2

CORE AND COOLANT ACTIVITIES [1]

Nuclide	Total Core Activity at Shutdown (Ci)	Maximum Coolant Activity (based on 1% fuel defects) (μCi/gm)
I-131	4.4 E7	2.4 E0
I-132	6.3 E7	2.4 E0
I-133	9.0 E7	3.8 E0
I-134	9.9 E7	5.3 E-1
I-135	8.4 E7	1.9 E0
Kr-85	5.4 E5	6.9 E0
Kr-85m	1.2 E7	1.4 E0
Kr-87	2.3 E7	9.7 E-1
Kr-88	3.2 E7	2.7 E0
Xe-131m	4.7 E5	2.5 E0
Xe-133	8.9 E7	2.3 E2
Xe-133m	2.8 E6	4.2 E0
Xe-135	2.3 E7	7.4 E0
Xe-135m	1.7 E7	4.0 E-1
Xe-138	7.5 E7	5.9 E-1

[1] These core and coolant activities were specifically recalculated for the Point Beach fuel upgrade/uprating program.

TABLE 3

CONTROL ROOM PARAMETERS

Volume	65,243 ft^3
Unfiltered Inleakage	
Mode 1	65.2 cfm
Mode 2	65.2 cfm
Mode 4	10.0 cfm
Normal unfiltered CR HVAC (Mode 1)	1000 cfm
Total Flow Rate	19800 cfm
Filtered Makeup	
Mode 2	0 cfm
Mode 4	4950 cfm
Filtered Recirculation	
Mode 2	0 cfm
Mode 4	0 cfm
Filter Efficiency	
Elemental	90%
Organic	90%
Particulate	99%
Occupancy Factors	
0-1 day	1.0
1-4 days	0.6
4-30 days	0.4

TABLE 4

ASSUMPTIONS USED FOR LOCKED ROTOR DOSE ANALYSIS

Power	1650 MWt
Reactor Coolant Noble Gas Activity Prior to Accident	1.0% Fuel Defect Level
Reactor Coolant Iodine Activity Prior to Accident	60 μCi/gm of DE I-131
Activity Released to Reactor Coolant from Failed Fuel (Noble Gas & Iodine)	100% of Core Gap Activity
Fraction of Core Activity in Gap (Noble Gas & Iodine)	0.10
Secondary Coolant Activity Prior to Accident	1.2 μCi/gm of DE I-131
Total SG Tube Leak Rate During Accident	0.7 gpm
SG Iodine Partition Factor	0.01
Duration of Activity Release from Secondary System	8 hours
Offsite Power	Lost [1]
Steam Release from SGs to Environment	206,000 lb (0-2 hr) 434,000 lb (2-8 hr)

[1] Assumption of a loss of offsite power is conservative for the locked rotor dose analysis.

TABLE 5

LOCKED ROTOR DOSES

Site Boundary (0-2 hr)	
Thyroid	15.6 rem
γ-body	1.8 rem
Low Population Zone (0-8 hr)	
Thyroid	10.0 rem
γ-body	0.2 rem
Control Room (0-24 hr)	
Thyroid	65.3 rem [1]
γ-body	0.4 rem
Beta skin	11.0 rem

[1] This calculated dose exceeds the 30 rem thyroid limit; however, assuming that the operators would be instructed to take the potassium iodide pills, this control room thyroid dose would be reduced to approximately 6.5 rem which is within the limit.

TABLE 6

ASSUMPTIONS USED FOR ROD EJECTION ACCIDENT DOSE ANALYSIS

Power	1650 MWt
Reactor Coolant Noble Gas Activity Prior to Accident	1.0% Fuel Defect Level
Reactor Coolant Iodine Activity Prior to Accident	60 μCi/gm of DE I-131
Activity Released to Reactor Coolant AND Containment from Failed Fuel (Noble Gas & Iodine)	10.0% of Core Gap Activity
Fraction of Core Activity in Gap (Noble Gas & Iodine)	0.10
Activity Released to Reactor Coolant AND Containment from Melted Fuel	
Iodine	0.125% of Core Activity
Noble Gas	0.25% of Core Activity
Iodine Removal in Containment	
Instantaneous Iodine Plateout	50%
Secondary Coolant Activity Prior to Accident	1.2 μCi/gm of DE I-131
Total SG Tube Leak Rate During Accident	0.35 gpm per SG
Iodine Partition Factor in SGs	0.01
Containment Free Volume	1.065×10^6 ft^3
Containment Leak Rate	
0-24 hr	0.4% / day
> 24 hr	0.2% / day
Steam Release from SGs	158,200 lb
Duration of Steam Release	
Primary to secondary leakage	1500 seconds
Initial secondary activity	86 seconds
Offsite Power	Lost

TABLE 7

ROD EJECTION OFFSITE & CONTROL ROOM DOSES

Site Boundary (0-2 hr)	
Thyroid	21.9 rem
γ-body	0.2 rem
Low Population Zone (0-8 hr)	
Thyroid	9.4 rem
γ-body	0.03 rem
Control Room (0-24 hr)	
Thyroid	122 rem [1]
γ-body	0.02 rem
Beta skin	0.4 rem

[1] This calculated dose exceeds the 30 rem thyroid limit; however, assuming that the operators would be instructed to take the potassium iodide pills, this control room thyroid dose would be reduced to approximately 12 rem which is within the limit.

TABLE 8

ASSUMPTIONS FOR SGTR DOSE ANALYSIS

Power	1650 MWt
Reactor Coolant Noble Gas Activity Prior to Accident	1.0% Fuel Defect Level
Reactor Coolant Iodine Activity Prior to Accident Pre-Accident Spike Accident Initiated Spike	 60 μCi/gm of DE I-131 1.0 μCi/gm of DE I-131
Reactor Coolant Iodine Activity Increase Due to Accident Initiated Spike	500 times equilibrium release rate from fuel for initial 1.6 hours after SGTR
Secondary Coolant Activity Prior to Accident	1.2 μCi/gm of DE I-131
SG Tube Leak Rate for Intact SG During Accident	0.35 gpm
Break Flow to Ruptured SG	123,600 lb (0-30 min)
SG Iodine Partition Factor	0.01
Duration of Activity Release from Secondary System	8 hours
Offsite Power	Lost
Steam Release from SGs to Environment Ruptured SG Intact SG	 74,000 lb (0-30 min) 1,660,000 lb (0-2 hr) [1] 1,373,000 lb (2-24 hr)

[1] The actual steam release for 0-2 hours is much lower (232,600 lb); however, this larger value was used in the radiological analysis and is conservative.

TABLE 9

SGTR OFFSITE & CONTROL ROOM DOSES

Site Boundary (0-2 hr)	
Thyroid: Accident Initiated Spike	1.7 rem
Thyroid: Pre-Accident Spike	3.5 rem
γ-body	0.1 rem
Low Population Zone (0-8 hr)	
Thyroid: Accident Initiated Spike	0.1 rem
Thyroid: Pre-Accident Spike	0.2 rem
γ-body	0.006 rem
Control Room w/Mode 2 (0-24 hr)	
Thyroid: Accident Initiated Spike	10.9 rem
Thyroid: Pre-Accident Spike	25.2 rem
γ-body	0.04 rem
Beta skin	3.6 rem
Control Room w/Mode 4 (0-24 hr)	
Thyroid: Accident Initiated Spike	1.4 rem
Thyroid: Pre-Accident Spike	3.8 rem
γ-body	0.005 rem
Beta skin	0.3 rem

TABLE 10

ASSUMPTIONS USED FOR SLB DOSE ANALYSIS

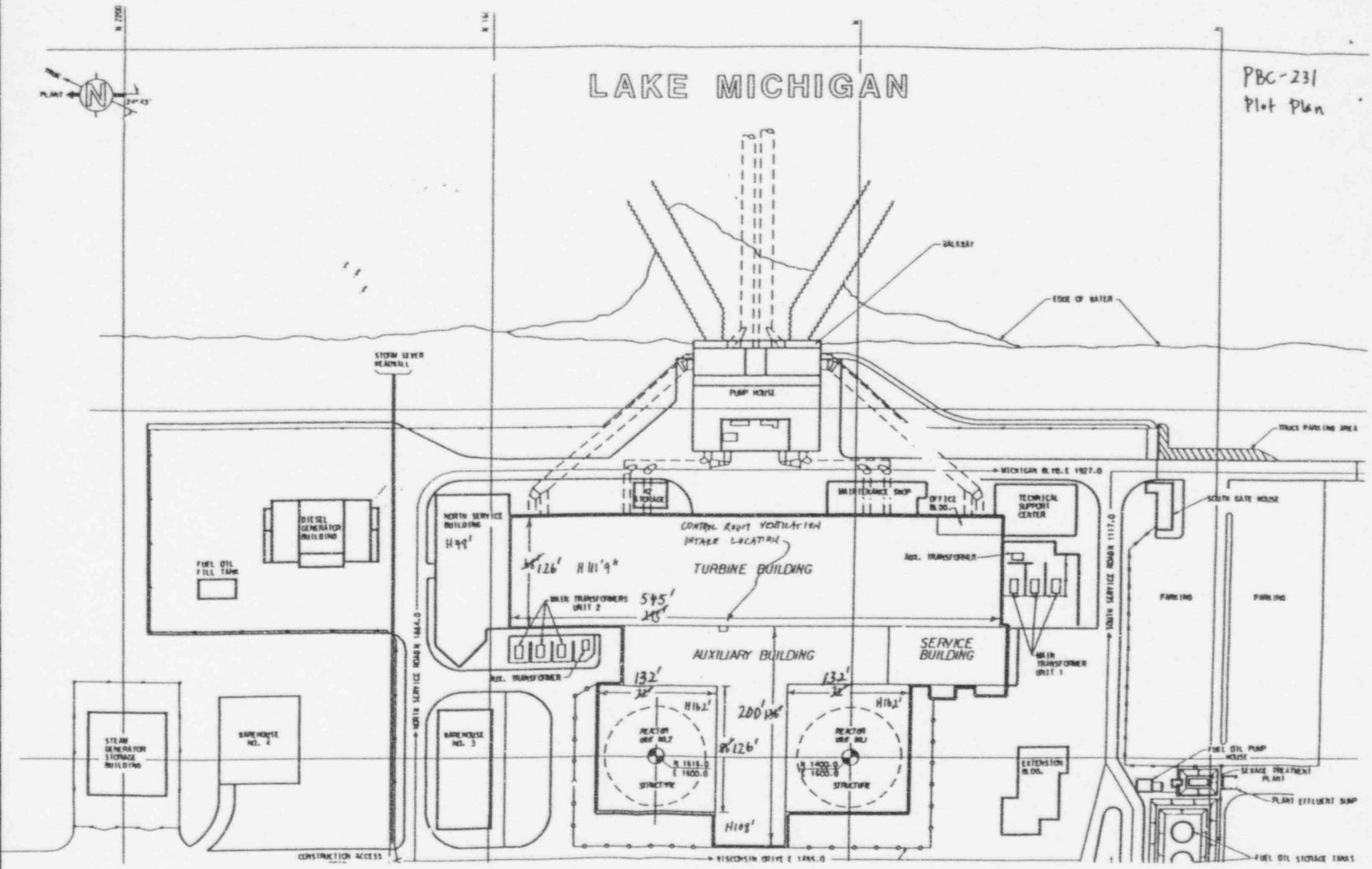
Power	1650 MWt
Reactor Coolant Noble Gas Activity Prior to Accident	1.0% Fuel Defect Level
Reactor Coolant Iodine Activity Prior to Accident	
Pre-Accident Spike	60 μCi/gm of DE I-131
Accident Initiated Spike	1.0 μCi/gm of DE I-131
Reactor Coolant Iodine Activity Increase Due to Accident Initiated Spike	500 times equilibrium release rate from fuel for initial 1.6 hours after SGTR
Secondary Coolant Activity Prior to Accident	1.2 μCi/gm of DE I-131
SG Tube Leak Rate for Intact SG During Accident	0.35 gpm
Iodine Partition Factor	
Faulted SG	1.0 (SG assumed to steam dry)
Intact SG	0.01
Duration of Activity Release from Secondary System	8 hours
Offsite Power	Lost
Steam Release from Intact SG	212,000 lb (0-2 hr) 405,000 lb (2-8 hr)

TABLE 11

SLB DOSES

Site Boundary (0-2 hr)	
Thyroid: Accident Initiated Spike	8.0 rem
Thyroid: Pre-Accident Spike	8.3 rem
γ-body	0.03 rem
Low Population Zone (0-8 hr)	
Thyroid: Accident Initiated Spike	0.7 rem
Thyroid: Pre-Accident Spike	0.7 rem
γ-body	0.002 rem
Control Room w/Mode 2 (0-24 hr)	
Thyroid: Accident Initiated Spike	134 rem [1]
Thyroid: Pre-Accident Spike	136 rem [1]
γ-body	0.006 rem
Beta skin	0.08 rem
Control Room w/Mode 4 (0-24 hr)	
Thyroid: Accident Initiated Spike	15.6 rem
Thyroid: Pre-Accident Spike	15.8 rem
γ-body	0.002 rem
Beta skin	0.03 rem

[1] This calculated dose exceeds the 30 rem thyroid limit; however, assuming that the operators would be instructed to take the potassium iodide pills, this control room thyroid dose would be reduced to approximately 14 rem which is within the limit.


PBC-231
Plot Plan
LAKE MICHIGAN
PLANT
N
GALLERY
EDGE OF WATER
STORM SEWER HEADWALL
PUMP HOUSE
TRUCK PARKING AREA
MICHIGAN BLVD. E 1927.0
ICE STORAGE
MAINTENANCE SHOP
OFFICE BLDG.
TECHNICAL SUPPORT CENTER
SOUTH GATE HOUSE
DIESEL GENERATOR BUILDING
FUEL OIL FILL TANK
NORTH SERVICE BUILDING
H99'
CONTROL ROOM VENTILATION INTAKE LOCATION
TURBINE BUILDING
126'
H 111'9"
545'
245'
MAIN TRANSFORMERS UNIT 2
AUX. TRANSFORMER
AUX. TRANSFORMER
MAIN TRANSFORMER UNIT 1
AUXILIARY BUILDING
SERVICE BUILDING
NORTH SERVICE ROADWAY 1864.0
SOUTH SERVICE ROADWAY 1117.0
PARKING
PARKING
132'
32'
H162'
200'
126'
REACTOR UNIT NO.2
N 1515.0
E 1600.0
STRUCTURE
REACTOR UNIT NO.1
N 1400.0
E 1600.0
STRUCTURE
H109'
EXTENSION BLDG.
FUEL OIL PUMP HOUSE
SEWAGE TREATMENT PLANT
PLANT EFFLUENT SUMP
FUEL OIL STORAGE TANKS
STEAM GENERATOR STORAGE BUILDING
WAREHOUSE NO. 4
WAREHOUSE NO. 3
CONSTRUCTION ACCESS
WISCONSIN DRIVE E 1866.0