Wisconsin Electric
POWER COMPANY

Point Beach Nuclear Plant
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NPL 97-0144

10 CFR 50.4
10 CFR 50.90

April 2, 1997

Document Control Desk
US NUCLEAR REGULATORY COMMISSION
Mail Station P1-137
Washington, DC 20555-0001

Ladies/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; December 2, 1996; January 16, 1997; and March 20, 1997. 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 letter also provides supplemental information for Technical Specifications Change Requests 188 and 189. Specifically, Attachment 1 is additional information for the steam generator tube rupture analysis, Attachment 2 is additional and corrected information for the rod ejection analysis, Attachment 3 is additional information regarding control room habitability, and Attachment 4 is other revised and corrected information.

We have determined that these changes 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,

Douglas F. Johnson
Manager-Regulatory Services
and Licensing

1/1
A001

Subscribed and sworn to before me
on this 2nd day of April 1997.

Notary Public, State of Wisconsin

My commission expires 10/21/2000

080132

CAC
Attachment

cc: NRC Resident Inspector, NRC Regional Administrator, PSCW

ATTACHMENT 1

Steam Generator Tube Rupture

Prior to the steam generator tube rupture (SGTR) accident it is assumed that the plant has been operating with simultaneous fuel defects and SG tube leakage for a period of time sufficient to establish equilibrium levels of activity in primary and secondary coolant. The offsite and control room doses following a SGTR are analyzed considering both pre-accident and accident initiated iodine spikes. For the pre-accident iodine spike, it is assumed that a reactor transient has occurred prior to SGTR and has raised the RCS iodine concentration to the allowed Tech Spec value of 50 μCi/g as shown in Figure 15.3.1-5. 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 of 500 times greater than the normal equilibrium rate corresponding to the initial RCS iodine activity. For both of these iodine spike cases, the SGTR radiological analysis includes three primary sources of activity; initial secondary side iodine activity, RCS coolant activity released via primary to secondary SG tube leakage in the intact SG, and RCS coolant activity carried over from the primary coolant via the ruptured SG tube.

The model for the activity available for release to the atmosphere from the ruptured and intact steam generators assumes that the release consists of the activity in the secondary coolant prior to the accident plus that activity leaking from the primary coolant through the SG tubes following the accident. The primary coolant activity after the accident is assumed to be composed of the pre-accident iodine spike activity or accident initiated iodine spike activity, plus the noble gases released due to 1% fuel defects. The leakage of primary coolant to the secondary side of the SG is assumed to continue at its initial rate of 0.35 gpm in the intact SG for the duration of the accident. A coincident loss of offsite power is assumed resulting in the loss of the condensers and the release of activity to the atmosphere through the main steam safety valves and the atmospheric steam relief valve from the intact steam generator. Eight hours after the accident, the Residual Heat Removal (RHR) System begins operation to cool down the plant. No further steam or activity is released to the environment.

A separate thermal hydraulic analysis was performed to determine the amount of reactor coolant transferred to the secondary side of the ruptured steam generator and the amount of steam released from the ruptured and intact steam generators to the atmosphere. This analysis was performed to support a power uprate program for Point Beach Units 1 & 2 and is used to conservatively bound the replacement steam generator program. A specific thermal and hydraulic analysis was performed for the replacement steam generator program. The values for primary to secondary break flow and steam released to the atmosphere are bounded by those calculated at the uprated power conditions. Per this analysis the break flow through the ruptured steam generator will deliver 123,600 lbm of reactor coolant to the secondary side of the steam generator. None of the break flow is assumed to flash in the steam generator resulting in a direct release to the environment. The primary to secondary break flow is assumed to persist until 30 minutes after the initiation of the SGTR, at which time it is assumed that the operators have completed the actions necessary to terminate the break flow and the steam release from the ruptured steam generator. The amount of steam released from the ruptured steam generator during the 30 minute time period is calculated to be 74,000 lbm. A partition factor of 0.01, as defined in SRP 15.6.3, is applied to this steam release. No credit is taken for additional partitioning in the condenser prior to reactor trip. Both the breakflow and steam releases are averaged over the 30 minute time interval.

The Westinghouse analysis for SGTR is comprised of 5 separate computer runs; a nominal RCS iodine activity case, a pre-accident iodine spike case, an accident initiated iodine spike case, an initial secondary coolant iodine case and a noble gas case. Each of the iodine cases model the releases to the environment from both the intact and ruptured steam generator using a partition factor of 0.01 on the steam releases. For each of these cases, except the initial secondary coolant iodine case, a transfer is modeled from the

RCS to the steam generators based on primary to secondary leakage to the intact SG and the breakflow through the ruptured SG.

The following table specifies the activities in the steam generators and those released to the environment at the end of the 30 minute and 8 hour time intervals. After 30 minutes no further activity release from the ruptured SG is assumed. Primary to secondary leakage to the intact steam generator is terminated 8 hours after the accident. The reduction in the concentration of the ruptured steam generator between 30 minutes and 8 hours is due to radioactive decay.

	Intact SG Concentration µCi/g	Ruptured SG Concentration µCi/g	Total Activity Released Ci
Normal RCS Iodine Activity Case			
0.5 hours	1.777E-3	3.402	0.636
8.0 hours	1.306E-2	2.066	0.671
Initial Secondary Iodine Activity Case			
0.5 hours	3.197	3.338	7.543
8.0 hours	1.582	2.027	26.267
Pre-Accident Iodine Case			
0.5 hours	0.1066	204.0	38.168
8.0 hours	0.7839	123.9	40.237
Accident Initiated Iodine Case			
0.5 hours	0.04903	93.05	11.140
8.0 hours	1.777	32.66	15.062

The steam releases calculated for the intact steam generator were calculated for the power uprate program over a twenty-four hour period which corresponds to the time required to reach plant conditions to allow the RHR system to be placed into operation for the uprated plant conditions. Replacement steam generators would not create changes that would impact the time for the RHR system to be placed into operation. The FSAR analysis specifies a total release time of 6 hours until RHR cut-in is reached. This has been increased to 8 hours for conservatism and to be more consistent with typical analysis values. The total steam releases from the intact and ruptured SGs are summarized below.

	Rate of Steam Release (gm/min)	Mass of Steam Released (lbm)
Ruptured Steam Generator		
0 - 0.5	1.12E6	74,000
Intact Steam Generator		
0 - 2 hours	6.28E6[1]	1.66E6[1]
2 -8 hours	4.72E5	3.743E5

[1] It should be noted that the 0 - 2 hour steam release was calculated in error in the conservative direction. The revised mass released during this interval was recalculated to be 232,600 lbm. The radiological analysis conservatively used the larger value.

The noble gas case models a release directly from the RCS to the environment based on primary to secondary leakage to the intact SG and the breakflow through the ruptured SG assuming an RCS coolant activity corresponding to a fuel defect level of 1 percent.

ATTACHMENT 2

Rod Ejection Accident

Prior to the accident, the plant is assumed to have been operating with one percent fuel defects for determing noble gas activity and a primary system event has caused an iodine spike which has raised the primary system I-131 dose equivalent iodine concentration to the allowed Technical Specification value of 50 μCi/g as shown in Figure 15.3.1-5. Following the accident, two release paths contribute to the total radiological consequences of the accident. The first is the leakage of radioactivity from the containment atmosphere to the environment and the second is the leakage of radioactivity from the secondary system through the steam generator relief valves. The radioactivity in the containment atmosphere is due to the radioactivity in the primary system coolant that has spilled out of the primary system into the containment through the hole in the reactor head created by the rod ejection. The radioactivity in the secondary system is due to the radioactivity in the primary system coolant that has leaked into the secondary system prior to the accident and also to the radioactivity that is transported to the secondary system by the primary system coolant that leaks through the steam generator tubes during the accident. Steam is released from the steam generator for heat removal purposes because condenser cooling is lost due to the assumed coincident loss of offsite power during the accident.

Core Release Model

The quantity of radioactivity released from the reactor core either to the primary system or to the containment atmosphere during the accident was conservatively calculated using the following assumptions:

1. Ten percent of the fuel rods in the reactor core experience DNB resulting in damage to the clad and all of the noble gases and iodines in the gap of these fuel rods is released. The activity contained in all of the fuel rod gaps consists of 10 percent of the iodines and noble gases accumulated in the reactor core at the end of core life. The Point Beach FSAR analysis (§14.2.6) states that less than 15% of the fuel rods in the core enter DNB. The 10% value used in this analysis is supported by WCAP-7588, Revision 1-A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods."

2. One quarter of one percent (0.25 %) of the fuel in the reactor core suffers fuel melt. This value was determined using the following assumptions.

   a. Fifty percent of the fuel rods experiencing clad damage may also experience fuel melting at the centerline of the fuel rod;

   b. Ten percent of the fuel rods that may experience centerline fuel melting actually melt;

   c. Of those fuel rods actually melting, fifty percent of the axial length of the fuel rod melts due to the power distribution;

Containment Release Pathway

The model for this release pathway assumes that all of the radioactivity initially present in the primary system due to the fuel defects and the pre-accident iodine spike and the radioactivity introduced by the fuel rod cladding failures and the melted fuel is instantaneously and homogeneously mixed throughout the net free volume of the containment atmosphere at the time of the accident. Of the radioiodines released to the containment atmosphere, fifty percent instantaneously plate out on the equipment and the structural members within the containment building. No credit is assumed for the removal of the radioiodine from the containment atmosphere by the containment spray system. The only removal processes considered are radioactive decay and leakage.

The containment leak rate for the first 24 hours is its design leak rate of 0.4 percent per day. Thereafter, the containment leak rate is 0.2 percent per day. The iodine and noble gas activity released during the 0 to 30 days post-accident time period is calculated to cause the offsite and control room doses listed in Table 1. The control room doses are calculated assuming control room HVAC operation in Mode 1 for the first 30 minutes and Mode 4 thereafter.

Table 1
Offsite and Control Room Doses Due to the Radioactivity Released From the Containment Building During the Rod Ejection Accident
(rem)

Location	Thyroid	Whole Body
Site Boundary	21.5	0.12
Low Population Zone	9.4	0.02
Control Room	122.0	0.02

Steam Generator Release Pathway

The model for the steam generator release pathway assumes that the release consists of the activity in the secondary system coolant prior to the accident plus the activity that leaks from the primary system to the secondary system through the steam generator tubes following the accident. The primary system activity is composed of the pre-accident iodine spike activity and all of the noble gases and radioiodines released from the fuel rods that experienced cladding failure and all of the noble gases and radioiodines released from the fuel rods that experienced fuel melt. This primary system activity is instantaneously and homogeneously mixed throughout the volume of the primary system.

The initial leakage rate of primary system coolant to the secondary system is 0.35 gallons per minute per steam generator. This leakage continues at this rate until the primary system and secondary system pressures equalize. Using a 2-inch small break LOCA model, it is calculated that the pressures will equalize in 1500 seconds. This is a conservative assumption because this size of break depressurizes the primary system at a rate that is slower than a break of comparable size in the reactor head. Once the pressures equalize, transfer of coolant from the primary system to the secondary system stops. Steam is released from the steam generator for heat removal purposes because condenser cooling is lost due to the assumed coincident loss of offsite power during the accident. A partition factor of 0.01 is applied to the radioiodine that is released from the secondary side of the steam generators through the steam generator relief valves.

Three separate analyses are performed to assess the amount of radioactivity released to the atmosphere through the steam generator relief valves. For each analysis, the mass of steam released is calculated which is then used to calculate the quantity of radioactivity released to the environment. The first analysis, called the initial activity case, calculates the amount of radioiodine released when the concentration of radioiodine in the secondary coolant is equal to the Technical Specification limit. The second analysis, called the transferred activity case, calculates the quantity of radioiodine released due to the radioactivity in the primary system coolant that leaks to the secondary system. The third analysis, called the cooldown activity case, calculates the amount of radioiodine released during the time it takes to cooldown the primary system to the pressure and temperature conditions needed to initiate RHR. For both the initial activity case and the transferred activity case, the mass of steam released is conservatively assumed to be equal to the total mass of secondary coolant present in both steam generators. This is a conservative assumption because the majority of the primary system depressurization and energy removal will occur through the break in the reactor head with a smaller amount of energy removal occurring through the steam generator relief valves. Since the main feedwater and auxiliary feedwater systems are operational, complete evaporation of the secondary side of the steam generators will not occur. For the cooldown activity case, the mass of steam released is calculated based on the amount of energy that must be removed to reach the conditions which allow operation of the RHR system. The quantity of energy which must be removed depends on the reactor decay heat, the heat content of the primary and secondary coolant mass, and the heat content of the metal mass involved with the primary and secondary systems.

For the initial activity case, the concentration of radioiodine in the secondary system coolant when the accident occurs is equal to the Technical Specification limit of 1.0 μCi/g. It is assumed that the entire mass of water in the secondary sides of both steam generators is released to the atmosphere through all four main steam safety valves at the maximum available relief rate. This is conservative since a 2-inch small break LOCA will typically only open one or possibly two of the main steam safety valves. This maximum relief rate assumption was made to minimize the time required to release the contents of the secondary side of the steam generator to the atmosphere and to limit the amount of radioactive decay of the radioiodine that could occur. The maximum relief rate for one main steam safety valve is 833,000 lbm/hr. Since each steam generator has four main steam safety valves, the total relief rate is 6,664,000 lb/hr. With a steam generator secondary side coolant mass of 158,200 lbs for two steam generators, complete depressurization occurs in 85.46 seconds. Thus, 158,200 lbs of steam is released in 85.5 seconds at an average rate of 6,664,000 lbm/hr. The offsite and control room doses resulting from the release of radioiodine contained within the coolant of the secondary side of the steam generators is listed in Table 2. The control room doses are calculated assuming control room HVAC operation in Mode 1 for the first 30 minutes and Mode 4 for the duration of the accident.

Table 2
Offsite and Control Room Doses Due to the Radioactivity Released From the Steam Generator During the Initial Activity Case - Rod Ejection Accident
(rem)

Location	Thyroid	Whole Body
Site Boundary	0.14	< 0.01
Low Population Zone	0.01	< 0.01
Control Room	0.01	< 0.01

For the transferred activity case, separate analyses were performed to calculate the noble gas and radioiodine releases. It is assumed that the entire coolant mass of the secondary sides of both steam generators is released to the atmosphere through the main steam safety valves. However, the time over which this mass of water is released is based on the maximum calculated time for the primary system and secondary system pressures to equalize when the primary system depressurizes at the rate calculated for a 2-inch small break LOCA. This assumption is made to maximize the time over which the primary system coolant can leak to the secondary system, resulting in the maximum release to the environment. During the depressurization, primary system coolant leaks to the secondary system at the Technical Specification limit of 0.7 gallons per minute or approximately 265 lbm/hour. The secondary system pressure is determined assuming that the pressure begins at the lowest lift setpoint for the main steam safety valves less the allowed setpoint tolerance and the 13% blowdown of the steam generators. This produces the lowest secondary system pressure further maximizing the mass of steam released to the environment. Using the 2-inch small break LOCA model, it is calculated that the primary system and secondary system pressures will equalize in 1500 seconds.

When the pressures equalize, primary system leakage to the secondary system stops and steam releases to the environment stop. Thus, for the transferred activity case, primary system coolant leaks to the secondary system at a constant rate of approximately 265 lbm/hr for 1500 seconds. During this time, 158,200 lbm of steam is released to the environment at an average rate of 379,700 lb/hr. For the radioactive noble gas, the activity is released directly from the primary system to the environment at the primary system to secondary system leak rate until the primary and secondary system pressures are equalize. Thus radioactive noble gas is released directly to the environment at a rate equivalent to approximately 265 lbm of primary system coolant per hour for 1500 seconds. The offsite and control room doses resulting from the release of the radioiodine and the radioactive noble gas contained within the primary system coolant transferred to the of the secondary side of the steam generators is listed in Table 3. The control room doses are calculated assuming control room HVAC operation in Mode 1 for the first 30 minutes and Mode 4 for the duration of the accident.

Table 3
Offsite and Control Room Doses Due to the Radioactivity Released From the Steam Generator During the Transferred Activity Case - Rod Ejection Accident
(rem)

Location	Thyroid	Whole Body
Site Boundary	0.33	0.11
Low Population Zone	0.02	0.01
Control Room	0.17	<0.01

For the cooldown activity case, the radioiodine present at the start of the cooldown is determined from the activity remaining in the secondary coolant following the initial activity and transferred activity cases discussed above. These activities are summarized in Table 4.

Table 4
Radioiodine Present in the Secondary System Following the Initial and Transferred Activity Cases
Rod Ejection Accident
(curies)

Nuclide	Radioiodine From the Initial Activity Case	Radioiodine From the Transferred Activity Case	Total Radioiodine
I-131	55.7	250.4	306.1
I-132	55.3	316.0	371.3
I-133	88.2	504.4	592.7
I-134	11.8	403.5	415.3
I-135	44.5	454.9	499.4

This case begins at 1500 seconds post-accident when the primary and secondary system pressures equalize and leakage of primary system coolant to the secondary system has stopped. It is assumed that during the time period 1500 seconds to two hours decay heat from the core is removed through the break in the reactor head and by the water supplied by the safety injection system. Since noble gases are not retained in the secondary system coolant and no new primary system coolant is being introduced into the secondary system, only radioiodine will be available for release. At 2 hours post-accident, cooldown to RHR initiation conditions begins. Cooldown will continue for six hours at which time the conditions for RHR operation are met and RHR operation can begin. At the start of the cooldown, the primary system pressure is assumed equal to the secondary system pressure and radioiodine is released to the environment for the next six hours. This time interval is consistent with that used for the steam generator tube rupture and the main steam line break accidents and is conservative considering the depressurization characteristics of a rod ejection accident. The mass of steam released to the environment during the six hours is conservatively calculated at 428,000 lbm which is approximately three times the release for the initial and transferred activity cases discussed previously. This release, averaged over the six hour interval, yields a release rate of 71,333 lbm/hour. The offsite and control room doses resulting from the release of the radioiodine during the cooldown is listed in Table 5. The control room doses are calculated assuming control room HVAC operation in Mode 1 for the first 30 minutes and Mode 4 for the duration of the accident. Since the steam release for this case does not begin until two hours post-accident, there are no additional doses calculated at the site boundary location.

Table 5
Offsite and Control Room Doses Due to the Radioactivity Released From the Steam Generator During the Cooldown Activity Case - Rod Ejection Accident
(rem)

Location	Thyroid	Whole Body
Site Boundary	0.0	0.0
Low Population Zone	0.12	< 0.01
Control Room	0.84	< 0.01

The total offsite and control room doses following a rod ejection accident are listed in Table 6. The control room doses are calculated assuming control room HVAC operation in Mode 1 for the first 30 minutes and Mode 4 for the duration of the accident.

Table 6
Offsite and Control Room Doses Due to the Radioactivity Released During the Rod Ejection Accident
(rem)

Location	Thyroid	Whole Body
Site Boundary	22	0.23
Low Population Zone	10	0.03
Control Room	123	0.02

ATTACHMENT 3

Control Room Habitability

Point Beach currently maintains compliance with the dose limits of GDC 19 by the use of potassium iodide to block the radioactive iodine from the thyroid. The use of potassium iodide is directed by Point Beach Nuclear Plant Emergency Plan Implementing Procedure, EPIP 5.2, "Radioiodine Blocking and Thyroid Dose Accounting." A copy of this procedure is attached.

We have recently discovered that a modification performed in 1994 established a flow-path from the discharge of the control room recirculation fans (W-13B1 and W-13B2) to the mechanical equipment room where most of the control room HVAC equipment is located. This flow path diverts approximately 700 cfm from the control room recirculation flow. This flow diversion pressurizes the mechanical equipment room with control room envelope air. This mitigates the effect of any inleakage into the system in this room. The assumptions used for Mode 4 operation of the system are unaffected by this modification. Mode 4 uses 4950 cfm of filtered make-up air, which is sufficient to maintain positive pressure within the entire control room envelope and the mechanical equipment room.

Previous information regarding the operation of the control room HVAC system stated that Mode 2 was assumed to have an unfiltered inleakage of 65.2 cfm. Mode 2 should not assume 65.2 cfm unfiltered inleakage based on the diversion of 700 cfm from the system. All previously reported analysis results for Mode 2 should be disregarded. The analysis results provided that assume Mode 1 operation followed by Mode 4 are appropriate because the system will automatically switch to Mode 4 on a high radiation signal. If the system is in Mode 2 prior to the switch to Mode 4, the analyses assuming Mode 1 operation prior to Mode 4 are expected to result in higher doses because Mode 1 unfiltered make-up and inleakage is 1065.2 cfm which is greater than the 765.2 cfm that could be assumed in Mode 2. Therefore, the analysis results provided remain valid for unfiltered inleakage in Mode 2 of up to 1065.2 cfm.

Iodine Protection Factor for Mode 4 Control Room Model

A control room model sensitivity analysis was performed on the containment leakage release pathways for the large LOCA to determine the iodine protection factor inherent in the Westinghouse TITAN5 computer code for Mode 4 operation of the Point Beach control room HVAC system. The iodine protection factor was determined by calculating the control room dose in an unfiltered control room, and dividing this dose by the previously calculated control room dose which took credit for the HVAC filtration. Based on this analysis, the iodine protection factor modeled by the TITAN5 computer code for Mode 4 operation of the Point Beach control room HVAC system is approximately 10.5.

This can readily be justified by the fact that the only filtration available during Mode 4 operation is on the intake flow. Since the filter efficiencies are 90% elemental, 90% organic and 99% particulate it is expected that there would be approximately a factor of 10 reduction in the doses which credit the control room filters because of the comparatively large elemental iodine species fraction. This is less than the factor of 20 allowed in Murphy-Campe paper, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19", presented at the 13th AEC Air Cleaning Conference, for a control room model that only includes filtered intake without any filtered recirculation.

ATTACHMENT 4

Revisions and Corrections to Previous Information

The following revised/corrected information is provided as follows:

1. Revisions and corrections to Tables 1 through 11 previously provided as attachments to letter dated January 16, 1997.

2. Correction of typographical errors in the description of changes provided as an attachment to letter dated January 16, 1997.

3. Revised edited pages of FSAR Section 14.2.4 Steam Generator Tube Rupture.

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

[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	50 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	1.0 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)

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[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	50 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	1.0 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 (Initial) 428,000 lb (Cooldown)
Duration of Steam Release	
Primary to secondary leakage	1500 seconds
Initial secondary activity	86 seconds Cooldown 6 hours
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 (0-30 days)	
Thyroid	10 rem
γ-body	0.03 rem
Control Room (0-30 days)	
Thyroid	123 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	 50 μCi/gm of DE I-131 0.8 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	1.0 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	50 µCi/gm of DE I-131
Accident Initiated Spike	0.8 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	1.0 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.

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~~2 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. cwc 1/21/96

9. Change 1.2μCi/gram to 1.0μCi/gram in Technical Specifications Table 15.4.1-~~1~~2 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. cwc 1/21/96

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.