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VPNPD-88-035
NRC-88-007

10 CFR 50.71(e)

January 19, 1988

U. S. NUCLEAR REGULATORY COMMISSION
Document Control Desk
Washington, D. C. 20555

Gentlemen:

DOCKETS 50-266 AND 50-301
PERIODIC UPDATING OF FINAL SAFETY ANALYSIS REPORT
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

In accordance with 10 CFR 50.71(e), Wisconsin Electric Power Company, Licensee for Point Beach Nuclear Plant, Units 1 and 2, submits the Fall 1987 revision to the Point Beach Nuclear Plant Final Safety Analysis Report (FSAR). As required by 10 CFR 50.4, one original and ten copies of these revised pages are enclosed. Each copy contains revised FSAR pages that are to be inserted in accordance with the accompanying instructions.

The information contained within the updated FSAR pages represents those changes made since our Spring 1987 submittal which were necessary to reflect information and analyses submitted to the Commission or prepared pursuant to Commission requirements. The attachment to this letter provides a description of those changes. All changes made under the provisions of 10 CFR 50.59 which are included in this update are noted on the attachment and will be identified to the Commission in the 1987 Annual Results and Data Report. As delineated in the Point Beach Nuclear Plant Technical Specifications (TS 15.6.9.1.B.1), this report will be submitted to the Commission prior to March 1, 1988.

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Copies of this update to the Point Beach FSAR will be submitted to the following state and local organizations:

1. Public Service Commission of Wisconsin

2. State of Wisconsin Department of Natural Resources

3. State of Wisconsin Department of Health and Social Services

4. Town Chairman, Town of Two Creeks

Very truly yours,

C W Fay

C. W. Fay
Vice President
Nuclear Power

Copies to NRC Regional Administrator, Region III
NRC Resident Inspector

Subscribed and sworn to before me
this 25th day of January 1988.

Delores B. Gryczkowski
Notary Public, State of Wisconsin

My Commission expires 5-27-90 .

ATTACHMENT

DESCRIPTION OF CHANGES

FALL 1987 REVISION

January 15, 1988

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*Asterisked items denote changes in the facility description resulting from modifications made under the provisions of 10 CFR 50.59 but not previously submitted to the Commission.

1.) <u>Grammatical, typographical and editorial errors have been corrected, or editorial improvements have been made on the following pages</u>:

Table 2.6-3 (1 of 5)
(2 of 5)
(3 of 5)
(4 of 5)
(5 of 5)

Table 2.6-4 (1 of 4)
(2 of 4)
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(4 of 4)

Table 3.2.2-1 (1 of 3)
(2 of 3)
(3 of 3)

Table 3.2.3-1 (1 of 3)
(2 of 3)
(3 of 3)

Table 3.3.2-1 (1 of 2)
(2 of 2)

Table 4.1-4 (1 of 2)
(2 of 2)

Table 4.1-10 (1 of 2)
(2 of 2)

Table 4.2-5 (1 of 3)
(2 of 3)
(3 of 3)

Explanatory notes for
Table 5.2-1 (1 of 2)
(2 of 2)

Table 5.3-1 (1 of 2)
(2 of 2)

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Table 6.2-2 (1 of 2)
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Table 6.2-3 (1 of 3)
(2 of 3)
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Table 6.2-8(a) (1 of 2)
(2 of 2)

Table 6.2-9 (1 of 2)
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Table 6.4-7 (1 of 2)
(2 of 2)

Table 7.4-1 (1 of 2)
(2 of 2)

Table 7.4-3 (1 of 3)
(2 of 3)
(3 of 3)

Table 7.5-1 (1 of 3)
(2 of 3)
(3 of 3)

Table 9.2-6 (1 of 2)
(2 of 2)

Table 9.3-1 (1 of 2)
(2 of 2)

Table 9.3-2 (1 of 2)
(2 of 2)

Table 9.3-3 (1 of 3)
(2 of 3)
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Table 9.4-2 (1 of 2)
(2 of 2)

Table 10.1-1 (1 of 2)
(2 of 2)

Table 12.4-1 (1 of 2)
(2 of 2)

Table 13.2.2-1 (1 of 6) (4 of 6)
(2 of 6) (5 of 6)
(3 of 6) (6 of 6)

Page 14-5

Page 14.1.1-2

Table 14.1.9-1 (1 of 2)
(2 of 2)

Page 14.2.4-1

Table 14.3.2-3 (1 of 3)
(2 of 3)
(3 of 3)

Page 14.3.4-4

Page 14.3.4-8

Page 14.3.4-10

Appendix B Table 1 (1 of 7)
(2 of 7)
(3 of 7)
(4 of 7)
(5 of 7)
(6 of 7)
(7 of 7)

2.) <u>The listed drawings have been revised due to the completion of design changes or minor drawing discrepancies</u>:

<u>Section 1</u>

Figure 1.2-3
1.2-5
1.2-10
1.2-11

<u>Section 2</u>

Figure 2.2-4

<u>Section 3</u>

Figure 3.2.3-1
3.2.3-5

<u>Section 4</u>

Figure 4.2-1 Sh. 2
4.2-1a Sh. 1*

<u>Section 7</u>

Figure 7.2-6
7.7-1

<u>Section 8</u>

Figure 8.2-10 Sh. 1

<u>Section 9</u>

Figure 9.2-1
9.2-2
9.2-3
9.3-1*
9.3-2
9.3-3
9.4-1
9.6-1
9.6-2

<u>Section 9 (cont'd)</u>

Figure 9.6-4
9.6-5
9.6-6
9.6-7

<u>Section 10</u>

Figure 10.2-1 Sh. 1
10.2-1 Sh. 3
10.2-1a Sh. 1
10.2-1a Sh. 2
10.2-1a Sh. 3
10.2-2 Sh. 1
10.2-2 Sh. 2
10.2-2a Sh. 1
10.2-2a Sh. 2
10.2-4a Sh. 2
10.2-6
10.2-7a
10.2-8
10.2-8a

<u>Section 11</u>

Figure 11.1-1 Sh. 1
11.1-1 Sh. 2
11.1-2 Sh. 1
11.1-3
11.1-4 Sh. 1
11.1-5
11.1-6

<u>Section 14</u>

Figure 14.3.4-1
14.3.4-9
14.3.4-10
14.3.4-11
14.3.4-13

<u>Appendix E</u>

Figure E-6

3.) The following pages are revised to reflect technical changes to the FSAR:

Section 1

Page 1.2-7, Table 1.2-1

Description of Changes

Page 1.2-7 has been revised to reflect the addition of a fourth component cooling water (CCW) heat exchanger previously described in the Spring 1987 FSAR update. The addition of a fourth heat exchanger allows 2 of 4 heat exchangers to be available as "swing" or shared equipment, vice the previous 1 of 3.

Changes to Table 1.2-1, Partial Equipment List, reflect additional equipment which has been recently installed at PBNP. These changes are primarily associated with the DC electrical system; i.e., the additions of new station batteries, new battery chargers, and new DC-AC inverters used to power the instrument buses and the replacement of the Plant Process Computer System (PPCS). Changes elsewhere in the text associated with these additions have been made in recent FSAR update submittals.

Additionally, numerous changes were made to the table to eliminate or clarify abbreviations not explained in the Symbol Key at the end of the table.

Pages 1.8-1, 1.8-2, 1.8-3, 1.8-5, 1.8-7, 1.8-12, 1.8-17, 1.8-18, 1.8-19, 1.8-20, 1.8-21, 1.8-22, Table 1.8-1 (Items 9-14)

Description of Changes

Changes to these pages result from our annual update of the Wisconsin Electric Nuclear Power Department Quality Assurance Program Description in Section 1.8. This update is required by 10 CFR 50.54(a)(3), and an item by item description of the changes was submitted to the NRC by letter dated June 25, 1987.

Section 3

Table 3.2.2-1

Description of Change

The Total Heat Output value has been increased slightly to reflect a total Primary Heat Output of 1524 MWt (summing Total Core Heat (1518.5 MWt) and Reactor Coolant Pump heat (5.5 MWt)) in terms of BTU/hr., instead of 1518.5 MWt in terms of BTU/hr.

Page 3.3-1

Description of Change

The change to this page removes a sentence which described the "current" fuel cycle of operation for both Unit 1 and Unit 2. This sentence is removed because it provides superfluous information available through other sources, and because the information is "static" and will become obsolete in the time period between FSAR updates.

Figure 3.2.3-1 and Figure 3.2.3-5

Description of Changes

Changes to these figures resulted from the completion of the Reactor Vessel Upflow Modification. This modification was completed on Unit 2 during the Fall 1986 refueling outage, and was completed on Unit 1 during the Spring 1987 refueling outage.

The purpose of the modification was to effect a reversal of reactor coolant flow in the outer baffled area of the reactor, between the baffle and the core barrel. Coolant flow prior to the modification was in a top to bottom direction, but coolant jetting ("baffle jetting") associated with this flow was believed to have caused damage to fuel assemblies.

Figure 3.2.3-1 was modified to show the addition of eight flow holes in the top former plate. Figure 3.2.3-5 was modified to show the plugging of the flow holes in the core barrel. (MRs 86-058 and 86-059)*

Section 4

Page 4.3-3

Description of Change

A paragraph was deleted which described Westinghouse as providing plant operator training in "reactor vessel design, fabrication and testing as well as present and future precautions necessary for pressure testing and operating modes..."

Continuing operator training is conducted in areas such as heatup and cooldown rates, embrittlement effects, and operating procedures associated with plant evolutions involving heatups and cooldowns. It is conducted by the PBNP Training Group. The existing paragraph was erroneous in view of present operator training, and is deleted.

Section 6

Page 6.2-21, page 6.2-23, and Table 6.2-11

Description of Change

Various wording in these sections was revised/added to reflect a change in packing configurations for valves in the SI system exposed to recirculation flow. This change involves, in some cases, removing or blanking off lantern ring leakoff paths to the waste disposal system, and a reduction in the required number of packing rings for these valves. These changes reflect industry experience in these valving applications, as documented in EPRI Report NP-4255, Project 2233-3 (February 1986). (Spare Parts Equivalency Evaluation Document (SPEED) 87-06)

Page 6.2-22

Description of Change

The specified filler type for spiral wound gaskets (asbestos) has been deleted. The industry is phasing out asbestos in lieu of other filler materials. The gasket design is still spiral wound. (Spare Parts Equivalency Evaluation Document (SPEED) 87-015)

Table 6.2-13

Description of Change

Seven safety related snubbers were deleted from the Unit 1 listing in this table. These snubbers on the main steam system were replaced by five energy absorbers during the spring 1987 refueling outage. This modification also necessitated a change to Appendix A, Table A.1-1, which lists design damping factors for various structures and piping systems in response to the Design and Hypothetical Earthquake. (MR 87-086)*

Section 7

Page 7.2-24, 7.2-25 and Figure 7.2-6

Description of Change

Changes to these pages and Figure replace the existing designations of reactor trip breakers (TB-1, TB-2) and reactor trip bypass breakers (AB-1, AB-2) with more commonly used designations (RTA, RTB; BYB, BYA).

Page 7.2-33

Description of Change

The 1985 FSAR update was extensive in magnitude due to the introduction of optimized fuel assembly (OFA) fuel into the core beginning with the Fall 1984 refueling outage of Unit 2. Prior to this, the design Departure from Nucleate Boiling Ratio (DNBR) for the (standard) fuel was 1.30. As both the Unit 1 and Unit 2 cores now contain both OFA and STD fuel, reference to 1.30 as the design DNBR is improper - rather, we have used the term "limit value" to replace 1.30 as the design DNBR, as different DNBR values exist for the two fuel types.

Changes to the FSAR relative to OFA fuel were recommended by Westinghouse, and were based on analyses contained in the OFA Reload Transition Safety Report. Reference to a DNBR of 1.30 in the last sentence of page 7.2-33 has been removed, as was the reference to a table which was removed with the 1985 FSAR update. This sentence should have been modified with that update, but was missed due to oversight.

Figure 7.7-1

Description of Change

The figure was modified to properly depict the control room access doors opening into the turbine hall, vice the reverse direction.

Section 9

Page 9.2-31

Description of Changes

Changes to the valve description of the CVCS system reflect the existing packing philosophy for modulating valves. (See the Description of Change for Page 6.2-21) (Spare Parts Equivalency Evaluation Document (SPEED) 87-06)

Section 10

Table 10.2-1

Description of Change

Extensive changes to Table 10.2-1, AVT (All Volatile Treatment) Control, Secondary Chemistry Control Guidelines, were made to reflect PBNP adherence to EPRI Steam Generator Owners Group (SGOG) recommendations for secondary chemistry control.

Section 11

Page 11.1-6, 11.1-10, 11.1-11

Description of Changes

Changes to these pages reflect the removal of the radioactive waste solidification (ATCOR) system. With this removal, waste solidification is now performed by a vendor. The mobile mixer and all services in this capacity are provided by the vendor. (MR 85-199)

Page 11.2-25

Description of Change

The changes to this page removed reference to the Bio-Pak 60 breathing apparatus, and provide a brief description of the Scott Aviation 4.5 breathing apparatus. The substitution of Bio-Pak 60s with the Scott units was at the direction of the Wisconsin Electric Accident Prevention Division.

Page 11.2-26

Description of Changes

Maximum allowable radioactive contamination limits for respirators following cleaning have been revised to achieve consistency with recently revised Health Physics procedures.

Pages 11.3-1, 11.3-2, 11.3-3, 11.3-4, 11.3-5

Description of Changes

Some of the indicated changes to this section reflect a change in organizational structure at PBNP. The Chemistry and Health Physics group is now split into the Chemistry Group and the Health Physics Group, each headed by a Superintendent.

Changes to pages 11.3-2 and 11.3-3 remove item specific equipage listings for the chemistry lab and counting room.

Page 11.3-4 was corrected, in part, to better reflect the actual use and availability of radiation safety instructions, policies and procedures to PBNP personnel.

Changes to page 11.3-5 reflect a removal of a vendor-specific description of the instrument gamma calibrator. Removal of such specific information does not detract from the facility description.

Section 12

Pages 12.2-3, 12.2-4, Figure 12.2-2

Description of Change

These pages are revised to bring the text up-to-date with recent plant staff organizational changes. As well, the PBNP staff organization chart has been included as new Figure 12.2-2.

Page 12.3-1

Description of Change

Reference to Appendix "A" of 10 CFR Part 55 was removed, and reference to 10 CFR Part 55 only was retained. This is to reflect the recent reorganization and reissuance of Part 55, removing Appendix "A", as described in 52FR9453.

Section 14

Page 14.2.5-2 and 14.2.5-7

Description of Changes

The description of main steam isolation valve (MSIV) operation was modified to state that the valves are designed to be fully closed in less than five seconds under low flow vice no flow conditions. This is consistent with the conditions of testing for the MSIVs, as delineated in Technical Specification 15.4.7.

Amendment No. 106 to the Unit 1 operating license, and Amendment No. 109 to the Unit 2 operating license, issued March 23, 1987, allowed a change in the MSIV closure time test (TS 15.4.7) conditions from "no" flow to "low" flow. This was to allow testing of the valves in the "hot" condition, with steam in the main steam system, during system warmup, and prior to reactor criticality following a refueling shutdown. Under such conditions, "no" flow requirements were difficult to achieve. This change to the FSAR maintains consistency between the two documents.

Page 14.3.4-15

Description of Change

Corrections were made to the initial conditions (temperature and pressure) in containment for the containment integrity analysis. Previous conditions were listed as 90°F and 14.7 psig. These values have been changed to 120°F and 14.7 psia.

A review of the energy calculation following this sentence revealed that the actual specified initial containment conditions must be 120°F and 14.7 psia to substantiate the figures in the calculation. An additional review of the Final Facility Description and Safety Analysis Report (FFDSAR), the original PBNP Licensing Document, supported the 120°F figure, but also listed pressure in psig vice psia. It is believed that this error is typographical in both the FSAR and FFDSAR, as nothing in the other provided data or figures supports such a high initial pressure stipulated by 14.7 psig.

Figure 14.3.4-1

Description of Change

The ordinate label was modified to provide the correct units, 10^3 BTU/SEC vice 10^6 BTU/SEC. At 60 psig, each containment fan cooler unit is rated at 50E+06 BTU/HR. (This information is derived from the text of Section 14.3.4 and from the component technical manual.) This corresponds to a rate of 13.9E+03 BTU/SEC, a point on the curve. The error also existed in the FFDSAR. This is strictly an error of labelling, and does not affect the analysis.

Section 14 (continued)

Figure 14.3.4-9

Description of Change

The "10" value on the ordinate scale was incorrect, and was replaced by the "0" value. This error is seen through examination of the ordinate scale, and has no bearing on the curve values.

A value check was performed. The "Full Safeguards" energy removal (flat) curve on the figure was verified to be approximately 110E+03 BTU/SEC. Full Safeguards energy removal results from all four containment fan coolers operating at their design point (50E+06 BTU/HR x 4 coolers) plus both containment spray pumps operating (~100E+06 BTU/HR x 2 pumps). The summation of these rates (400E+06 BTU/HR) corresponds to 111E+03 BTU/SEC, and checks with the curve. Therefore, it is established that the previous "0" value was in error, and not the "100" or "200" ordinate values.

Figure 14.3.4-10

Description of Change

The lower curve label was changed from "Head Addition" to the proper "Heat Addition". The ordinate scale was given units of "Energy (10^6 BTU)", similar to Figure 14.3.4-11. These units were taken from the corresponding Figures in the FFDSAR, and were probably truncated inadvertently during the 1982 FSAR printing process. The "200" and "300" values on the ordinate were shifted to their proper location, exactly corresponding to the ordinate of Figure 14.3.4-11.

Figure 14.3.4-11

Description of Change

The ordinate scale was given units of "Energy (10^6 BTU)".

Figure 14.3.4-13

Description of Change

The "1000" second value on the abscissa is now correctly labelled vice the previous "10,000".

Appendices

Appendix A

Table A.1-1 (p.A-4)

Description of Change

A footnote was added to this table of damping factors, reflecting new and different damping factors for the Unit 1 main steam line outside of containment. These new factors resulted from the removal of seven safety related snubbers on the main steam bypass line, and their subsequent replacement with energy absorbers. (See also the Description of Change to Table 6.2-13) (MR 87-086)*

Appendix E

Table of Contents, Page E-10a, Page E-14 (References) and Figure E-6

Description of Change

Changes to these pages reflect completion of a modification which moved an auxiliary steam line to allow for the addition of a fourth component cooling water heat exchanger. (MRs 84-048 and 84-048-01)

Appendix F

Description of Change

A letter from the Atomic Energy Commission, dated February 27, 1974, directed that an analysis be performed to determine possible damage and consequences which may occur in the event of a spent fuel cask drop accident in or near the spent fuel pool. Appendix F to both the Final Facility Description and Safety Analysis Report (FFDSAR) and the Final Safety Analysis Report (FSAR) contained that analysis, addressing the following items of concern:

1. Integrity of the spent fuel storage pool.
2. Spent fuel integrity.
3. Integrity of critical systems and equipment.
4. Description of the handling equipment for the spent fuel cask.

The safety analysis addressed and considered the use of limit switches on the auxiliary building rectilinear crane to prevent load movements over the south half of the spent fuel pool (containing spent fuel), and to allow movement over the north half of the pool (containing no fuel). Restriction of crane movement to the north half of the pool was intrinsic to the development and results of the analysis. These physical crane restrictions were imposed, as, too, were administrative controls in the PBNP Technical Specifications.

The "heavy loads" issue, Generic Technical Activity A-36, was resolved with the publication of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." For PWR plants, a resolution option for the heavy loads issue was installation of a "single-failure-proof" crane for the handling of heavy loads (>1750 pounds) over spent fuel. Such a crane is designed so that no single failure of any crane component will result in a load drop. Wisconsin Electric chose this option, and the auxiliary building crane was so modified in 1985. The NRC Safety Evaluation Report dated September 3, 1985, addressed the installation of the single-failure-proof crane, and removed all restrictions on load movements in the spent fuel pool area through Amendment 96 to the Unit 1 license, and Amendment 100 to the Unit 2 license.

With the modification of the auxiliary building crane, and the subsequent removal of crane movement restrictions, the safety analysis presented in Appendix F is moot and is removed.