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Dominion®

March 28, 2014

U.S. Nuclear Regulatory Commission		Serial No	14-036
Attention: Document Control Desk		NSSL/MLC	R0
Washington, DC 20555		Docket No.	50-336
		License No.	DPR-65

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2
LICENSE AMENDMENT REQUEST FOR ADMINISTRATIVE CHANGES AND CORRECTIONS TO THE TECHNICAL SPECIFICATIONS

In accordance with the provisions of 10 CFR 50.90, Dominion Nuclear Connecticut, Inc. (DNC) is submitting a license amendment request to revise the Millstone Power Station Unit 2 (MPS2) Technical Specifications (TSs). The proposed changes delete the TS Index and make administrative changes and corrections to the TSs.

Attachment 1 to this letter provides a description and evaluation of the proposed changes. Attachment 2 contains the TS page mark-ups of the proposed changes. The proposed changes do not involve a Significant Hazards Consideration under the standards set forth in 10 CFR 50.92. The Facility Safety Review Committee has reviewed and concurred with the determinations herein.

In accordance with 10 CFR 50.91(b), a copy of this license amendment request is being provided to the State of Connecticut.

Should you have any questions in regard to this submittal, please contact Wanda Craft at (804) 273-4687.

Sincerely,

Mark D. Sartain
Vice President – Nuclear Engineering

VICKI L. HULL
Notary Public
Commonwealth of Virginia
140542
My Commission Expires May 31, 2014

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Mark D. Sartain, who is Vice President - Nuclear Engineering of Dominion Nuclear Connecticut, Inc. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 28TH day of March, 2014.

My Commission Expires: May 31, 2014.

Vicki L. Hull
Notary Public

A001
NRR

Attachments:

1. Evaluation of Proposed License Amendment
2. Marked-Up Technical Specification Pages

Commitments made in this letter: None

cc: U.S. Nuclear Regulatory Commission
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ATTACHMENT 1

EVALUATION OF PROPOSED LICENSE AMENDMENT

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2

EVALUATION OF PROPOSED LICENSE AMENDMENT

1.0 SUMMARY

In accordance with the provisions of 10 CFR 50.90, Dominion Nuclear Connecticut, Inc. (DNC) is submitting a license amendment request (LAR) to revise the Millstone Power Station Unit 2 (MPS2) Technical Specifications (TSs). The proposed changes delete the TS Index and make administrative changes and corrections to the TSs.

2.0 PROPOSED CHANGES

DNC proposes to make the following changes to the MPS2 TSs:

a) Delete TS index pages I through XVIII.

b) Replace figure number "2-2-2" on TS Page 2-7 with "2.2-2."

c) Replace "Y_1" (Y sub 1) in two locations in Figure 2-2-2 (to be relabeled as 2.2-2 as noted in (b) above) and in four locations in Figure 2.2-3 with "Y_I" (Y sub I).

d) Replace "3.4.3.a and 3.4.3.b" in TS Table 3.3-11, ACTION 3, with "3.4.3.b and 3.4.3.c."

e) Replace the word "begin" in TS Table 3.3-11, ACTION 5 with the words "be in."

f) Revise TS 6.3.2, Facility Staff Qualifications, from:

"If the operations manager does not hold a senior reactor operator license for Millstone Unit No. 2, then the operations manager shall have held a senior reactor operator license at a Pressurized Water Reactor and an individual serving in the capacity of the assistant operations manager shall hold a senior reactor operator license for Millstone Unit No. 2."

to:

"The operations manager or at least one operations middle manager shall hold a senior reactor operator license for Millstone Unit No. 2."

g) Insert the word "UNRODDED" into the term "Total Integrated Radial Peaking Factor - F^T_r" which is contained in TS 6.9.1.8 a. In addition, this term is a TS definition and therefore will be replaced with capitalized type. The new term will read as: TOTAL UNRODDED INTEGRATED RADIAL PEAKING FACTOR - F^T_r.

h) Replace "EFN" in Reference 5 on Page 6-18a with "EMF."

i) Insert “78” into Reference 7 on Page 6-18a to read: XN-NF-78-44(NP)(A). In addition, the first letter in the words rod, water and reactors in the title of this reference will be replaced with capitalized type.

j) Replace “2130” in Reference 14 on Page 6-19 with “2310.”

k) Replace the letter on the last line of Page 6-20, which currently appears as sub-letter “a” of TS 6.9.2, Special Reports, with sub-letter “g”.

l) Replace the term “SORC” in paragraph b of the “Licensee initiated changes to the REMODCM,” described in TS 6.15 with the term “FSRC.”

Mark-ups of the affected TS pages for the proposed changes described above are provided in Attachment 2.

3.0 TECHNICAL EVALUATION

The technical justification for each of the changes proposed in Section 2.0 of this LAR is provided below:

a) This LAR proposes to delete the index from the MPS2 TSs. Past LAR submittals to the NRC have included the TS index to support requested TS changes, and a revised index has been included in issued amendments. This change will eliminate the need to include TS index pages as part of LAR submittals for future TS changes and to issue revised index pages with an amendment. Although the index would no longer be considered part of the MPS2 TSs, controlled copy holders would continue to receive updates to the TS index for future TS and TS Bases changes.

   The TS index identifies the contents of the TSs and where the specific TS sections can be found throughout the TSs but does not contain any technical information required by 10 CFR 50.36. Therefore, this is an administrative change and does not involve any physical changes to structures, systems, or components (SSCs) in the plant, or the way SSCs are operated or controlled.

   Similar amendments to remove the TS index were approved for Seabrook in June 2013 (Reference 6.1), Waterford Unit 3 in May 2005 (Reference 6.2), and for Arkansas Nuclear One, Unit 2 in June 2005 (Reference 6.3).

b) The figure number on TS Page 2-7 for Figure 2-2-2, Local Power Density – High Trip Setpoint Part 2 (QR_2 Versus Y_1), is incorrect. The correct figure number is 2.2-2. This is a typographical error as the correct reference to this figure appears twice in TS Table 2.2-1.

   This proposed change is editorial in nature and does not involve any physical changes to SSCs in the plant, or the way SSCs are operated or controlled.

c) There are two locations in Figure 2-2-2 (to be relabeled as 2.2-2 as noted in (b) above) and four locations in Figure 2.2-3 where AXIAL SHAPE INDEX for the trip signals in the reactor protection system, is incorrectly labeled as "Y_1" (Y sub 1). As shown in the Definitions section of the MPS2 TSs, the symbol for AXIAL SHAPE INDEX for the trip signals in the reactor protection system, is "Y_I" (Y sub I).

   This proposed change is editorial in nature and does not involve any physical changes to SSCs in the plant, or the way SSCs are operated or controlled.

d) TS Table 3.3-11, ACTION 3, incorrectly references Specification "3.4.3.a or 3.4.3.b" rather than "3.4.3.b or 3.4.3.c."

   On February 15, 1995, Amendment 185 to the MPS2 TSs (Reference 6.4) was approved by the NRC. This amendment added two new ACTIONS to TS 3.4.3, Reactor Coolant System Relief Valves, to address the power-operated relief valve (PORV) and block valve reliability concerns identified in Generic Letter 90-06 (Reference 6.5). As part of this change, the reference in Surveillance Requirement (SR) 4.4.3.2 was changed from Specification "3.4.3 a or b" to Specification "3.4.3 b or c" to coincide with the changes to TS 3.4.3 for when a PORV block valve is closed and power is removed.

   At the time of the submittal for Amendment 185 (Reference 6.6), the reference in TS Table 3.3-11, ACTION 3, should have also been revised (similar to SR 4.4.3.2 above) to coincide with the changes made to the ACTIONS in TS 3.4.3. TS Table 3.3-11, ACTION 3 currently states:

   > *This ACTION is not required if the PORV block valve is closed with power removed in accordance with Specification 3.4.3.a or 3.4.3.b.*

   This statement is contradictory since Specification 3.4.3.a is for when power is maintained to the PORV block valve, not removed from the PORV block valve. Specifications 3.4.3.b and 3.4.3.c are the correct reference when a block valve(s) is closed and power is removed. This was an administrative oversight during the preparation of the LAR for Amendment 185.

   This proposed change is administrative in nature and does not involve any physical changes to SSCs in the plant, or the way SSCs are operated or controlled.

e) A typographical error exists in TS Table 3.3-11, ACTION 5. Specifically, the sentence should read:

   *"....or **be in** at least HOT SHUTDOWN within the next 12 hours."*

   rather than,

   *"....or **begin** at least HOT SHUTDOWN within the next 12 hours."*

This change is consistent with the wording used in other sections of the TSs. This proposed change is editorial in nature and does not involve any physical changes to SSCs in the plant, or the way SSCs are operated or controlled.

f) DNC proposes to revise TS 6.3.2, Facility Staff Qualifications, from:

"If the operations manager does not hold a senior reactor operator license for Millstone Unit No. 2, then the operations manager shall have held a senior reactor operator license at a Pressurized Water Reactor and an individual serving in the capacity of the assistant operations manager shall hold a senior reactor operator license for Millstone Unit No. 2."

to:

"The operations manager or at least one operations middle manager shall hold a senior reactor operator license for Millstone Unit No. 2."

The format of TS 6.3.2 is being revised to more closely match the format used in NUREG-1432, Rev. 4, TS 5.2.2d, for Combustion Engineering standard TS plants.

The proposed change also replaces the position of "assistant operations manager" with "at least one operations middle manager" and would require that either the operations manager or at least one operations middle manager (e.g., an assistant operations manager or the supervisor in charge of the operations shift crews) have a senior reactor operator (SRO) license for MPS2. As discussed in the regulatory analysis section of this LAR (Section 4.1.2), the proposed change to TS 6.3.2 is administrative in nature and has no technical implications on TS requirements. With the proposed change, operation, maintenance and testing will continue to assure safe operation of MPS2 and to protect the health and safety of the public. The TSs will continue to provide administrative controls relating to the organization and management of MPS2 and to ensure compliance with regulations.

g) In TS 6.9.1.8 a, the word UNRODDED is missing from the term "Total Integrated Radial Peaking Factor - F^T_r." Additionally, since this term is a TS definition, it should be shown in capitalized type. The replacement text will read as: TOTAL UNRODDED INTEGRATED RADIAL PEAKING FACTOR - F^T_r.

This proposed change is editorial in nature and does not involve any physical changes to SSCs in the plant, or the way SSCs are operated or controlled.

h) Reference 5 on Page 6-18a is incorrect. The "EFN" in Reference 5 should be "EMF." Therefore, the reference should read as follows:

EMF-2328(P)(A), "PWR Small Break LOCA Evaluation Model S-RELAP5 Based," Framatome ANP.

This proposed change is editorial in nature and does not involve any physical changes to SSCs in the plant, or the way SSCs are operated or controlled.

i) Reference 7 on Page 6-18a is incorrect. The number "78" is missing from the document number. Also, the first letter in the words rod, water and reactors in the document title should be in capitalized type. Reference 7 currently reads:

> *XN-NF-44(NP)(A), "A Generic Analysis of the Control rod Ejection Transient for Pressurized water reactors," Exxon Nuclear Company.*

Reference 7 should read:

> *XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," Exxon Nuclear Company.*

This proposed change is editorial in nature and does not involve any physical changes to SSCs in the plant, or the way SSCs are operated or controlled.

j) Reference 14 on Page 6-19 is incorrect. The "2130" in the document number should be "2310." Therefore, the reference should read as follows:

> *EMF-2310(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," Framatome ANP.*

This proposed change is editorial in nature and does not involve any physical changes to SSCs in the plant, or the way SSCs are operated or controlled.

k) A typographical error exists in TS 6.9.2, Special Reports. Specifically, the last line of Page 6-20 currently appears as sub-letter "a". The correct lettering should be sub-letter "g".

This proposed change is editorial in nature and does not involve any physical changes to SSCs in the plant, or the way SSCs are operated or controlled.

l) The term "SORC" which appears in the licensee initiated changes described in TS 6.15 is no longer used at MPS. This acronym, which stood for Site Operations Review Committee, has been replaced with FSRC or Facility Safety Review Committee. Consequently, the term "SORC" needs to be replaced with "FSRC."

This proposed change is administrative in nature and does not involve any physical changes to SSCs in the plant, or the way SSCs are operated or controlled.

4.0 REGULATORY ANALYSIS

4.1 Applicable Regulatory Requirements/Criteria

4.1.1 The NRC's regulatory requirements related to the content of the TSs are set forth in Title 10 of the *Code* of *Federal Regulations* (10 CFR) Section 50.36, "Technical specifications." This regulation requires that the TSs include items in the following five specific categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. However, the regulation does not specify the particular requirements to be included in a plant's TSs.

The TS index, which this LAR proposes to remove from the TSs, is not a required component of the TSs in 10 CFR 50.36. Similar to the TS Bases, the index provides information about the TSs but is not part of the TSs. Because the index does not provide technical information required by 10 CFR 50.36, the proposed administrative change to remove the index from the TSs is consistent with regulatory requirements.

4.1.2 TS 6.3.1 currently requires that each member of the facility staff shall meet or exceed the minimum qualifications of American National Standards Institute (ANSI) N18.1-1971, "Selection and Training of Nuclear Power Plant Personnel," for comparable positions (note that as of November 1, 2001, applicants for reactor operator and senior reactor operator qualification shall meet or exceed the education and experience guidelines of Regulatory Guide 1.8, Revision 3, May 2000). ANSI N18.1-1971 specifies that the operations manager hold an SRO license for the unit. In 1987, this guidance was revised in ANSI/ANS-3.1-1987 such that if the operations manager does not hold an SRO license, then the operations middle manager shall hold an SRO license.

The existing language used in TS 6.3.2, Facility Staff Qualifications, reflects a limited exception to ANSI N18.1-1971. Specifically, the exception allows the assistant operations manager (functionally equivalent to the operations middle manager referenced in ANSI/ANS-3.1-1987), if one is designated, to hold an SRO license if the operations manager does not hold an SRO license for the unit. This exception was approved for MPS2 under License Amendment (LA) 178 (Reference 6.7). As stated in the NRC safety evaluation report for LA 178, the staff concluded that:

"...the specification that either the Operations Manager or Assistant Operations Manager hold a Millstone Unit 2 SRO license, is consistent with the requirements of 10 CFR 50.54(l) and ensures that a licensed off-shift senior operator is directing the licensed activities of the licensed operators. Requiring an ANSI/ANS-3 .1-1987 qualified and licensed Assistant Operations Manager when the Operations Manager does not hold a valid Millstone Unit 2 SRO license is consistent with the requirements of ANSI N18 .1-1971 and ensures there is site-specific detailed relevant technical and systems knowledge in a senior operations management position."

The proposed administrative change to TS 6.3.2 is consistent with the NRC-approved limited exception to ANSI N18.1-1971. DNC proposes to revise TS 6.3.2 such that the operations manager or at least one operations middle manager (e.g., an assistant operations manager or the supervisor in charge of the operations shift crews) shall hold an SRO license for MPS2. Modifying the title in TS 6.3.2 to allow at least one operations middle manager (i.e., any position between the operations manager and the shift managers) to hold the SRO license on the unit meets the intent of ANSI N18.1-1971. This change will allow the operations manager and other operations middle managers to perform higher level duties such as management, planning, and coordinating of operations activities with more general senior operator knowledge.

This proposed change continues to be consistent with the requirements of 10 CFR 50.54(l) to ensure that a licensed off-shift senior operator is directing the licensed activities of the licensed operators and with the standards set forth in ANSI N18.1-1971 to ensure there is relevant operational experience and knowledge in senior operations management position(s).

Additionally, the format of TS 6.3.2 is being revised to more closely match the format used in NUREG-1432, Rev. 4, TS 5.2.2d, for Combustion Engineering standard TS plants.

4.2 No Significant Hazards Consideration

The NRC has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92(c). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with a proposed amendment would not: 1) involve a significant increase in the probability or consequences of an accident previously evaluated; or 2) create the possibility of a new or different kind of accident from any accident previously evaluated; or 3) involve a significant reduction in a margin of safety.

DNC has evaluated whether or not a significant hazards consideration is involved with the proposed changes. A discussion of these standards as they relate to this change request is provided below.

1) Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No. The proposed changes are administrative in nature. The proposed changes remove the TS Index and make other editorial and administrative corrections to the TSs. These administrative changes are not initiators of any accident previously evaluated, and, consequently, the probability and consequences of an accident previously evaluated is not significantly increased. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2) Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No. The proposed changes are administrative in nature so no new or different accidents result from the proposed changes. The changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed), a change in the method of plant operation, or new operator actions. The changes do not alter assumptions made in the safety analysis. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3) Do the proposed changes involve a significant reduction in the margin of safety?

Response: No. The proposed administrative changes do not involve a change in the method of plant operation, do not affect any accident analyses, and do not relax any safety system settings. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, DNC concludes that the proposed changes do not present a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

4.3 Conclusion

Based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

DNC has determined that the proposed amendment not would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined by 10 CFR 20, or an inspection or surveillance requirement. The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need to be prepared in connection with the proposed amendment.

6.0 REFERENCES

6.1 NRC Letter "Seabrook Station, Unit No. 1 – Issuance of Amendment Regarding the Administrative Changes and Corrections to the Technical Specifications (TAC No. MF1033)," June 17, 2013 (ADAMS Accession No. ML13074A760).

6.2 NRC Letter "Waterford Steam Electric Station, Unit 3 - Issuance of Amendment Re: Modification of Technical Specification (TS) 5.3.1, Fuel Assemblies, T.S. 5.6.1, Criticality, T.S. 6.9.1.11.1, Core Operating Limits Reports, and Deletion of TS Index (TAC No. MC3584)," May 9, 2005 (ADAMS Accession No. ML051290368).

6.3 NRC Letter "Arkansas Nuclear One, Unit 2 - Issuance of Amendments Re: Deletion of Index Pages from the Technical Specifications (TAC No. MC3246)," June 22, 2005 (ADAMS Accession No. ML051740191).

6.4 NRC Letter "Issuance of Amendment (TAC No. M89380)," February 15, 1995.

6.5 Generic Letter 90-06, "Resolution of Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability" and Generic Issue 94, "Additional Low-Temperature Over Pressure Protection for Light-Water Reactors" Pursuant to 10 CFR 50.54(f)," June 25, 1990.

6.6 Northeast Utilities Letter B14559 "Millstone Nuclear Power Station, Unit No. 2 – Proposed Revision to Technical Specifications – Generic Letter 90-06," April 25, 1994.

6.7 NRC Letter "Issuance of Amendment (TAC No. M89735)," August 11, 1994.

ATTACHMENT 2

MARKED-UP TECHNICAL SPECIFICATION PAGES

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2

~~October 27, 2008~~

~~INDEX~~

~~DEFINITIONS~~

~~INDEX~~

~~DEFINITIONS~~

~~January 1, 1986~~

~~INDEX~~

~~SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS~~

~~BASES~~

~~March 16, 2006~~

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~~LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS~~

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~~September 20, 2004~~

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~~September 25, 2003~~

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~~INDEX~~

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~~May 31, 2007~~

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~~March 16, 2006~~

~~INDEX~~

~~BASES~~

~~SECTION~~		~~PAGE~~
~~3/4.9.9 and 3/4.9.10 DELETED~~		~~B 3/4 9-2~~
~~3/4.9.11 and 3/4.9.12 WATER LEVEL - REACTOR VESSEL AND~~	~~STORAGE POOL WATER LEVEL~~	~~B 3/4 9-2b~~
~~3/4.9.13~~	~~DELETED~~	~~B 3/4 9-3~~
~~3/4.9.14~~	~~DELETED~~	~~B 3/4 9-3~~
~~3/4.9.15~~	~~STORAGE POOL AREA VENTILATION SYSTEM~~	~~B 3/4 9-3~~
~~3/4.9.16~~	~~SHIELDED CASK~~	~~B 3/4 9-3b~~
~~3/4.9.17~~	~~SPENT FUEL POOL BORON CONCENTRATION~~	~~B 3/4 9-3b~~
~~3/4.9.18~~	~~SPENT FUEL POOL - STORAGE~~	~~B 3/4 9-4~~
~~3/4.9.19~~	~~SPENT FUEL POOL - STORAGE PATTERN~~	~~B 3/4 9-4~~
~~3/4.9.20~~	~~SPENT FUEL POOL - CONSOLIDATION~~	~~B 3/4 9-4~~

~~3/4.10 SPECIAL TEST EXCEPTIONS~~

~~3/4.10.1~~	~~SHUTDOWN MARGIN~~	~~B 3/4 10-1~~
~~3/4.10.2~~	~~GROUP HEIGHT AND INSERTION LIMITS~~	~~B 3/4 10-1~~
~~3/4.10.3~~	~~DELETED~~	~~B 3/4 10-1~~
~~3/4.10.4~~	~~DELETED~~	~~B 3/4 10-1~~
~~3/4.10.5~~	~~DELETED~~	~~B 3/4 10-1~~

~~3/4.11 DELETED~~

~~3/4.11.1~~	~~DELETED~~	~~B 3/4 11-1~~
~~3/4.11.2~~	~~DELETED~~	~~B 3/4 11-1~~
~~3/4.11.3~~	~~DELETED~~	~~B 3/4 11-1~~

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+

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~~ADMINISTRATIVE CONTROLS~~

~~May 31, 2007~~

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~~June 28, 2012~~

~~INDEX~~

~~ADMINISTRATIVE CONTROLS~~

~~November 10, 1992~~

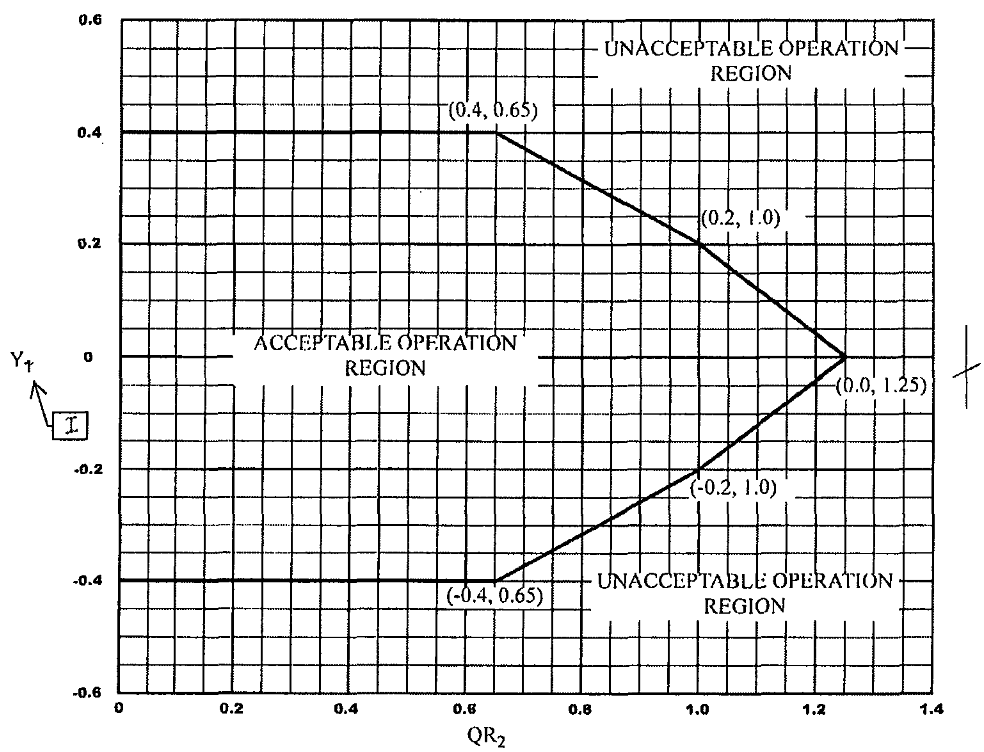

FIGURE 2-2-2 Local Power Density - High Trip Setpoint Part 2 (QR_2 Versus Y_1)

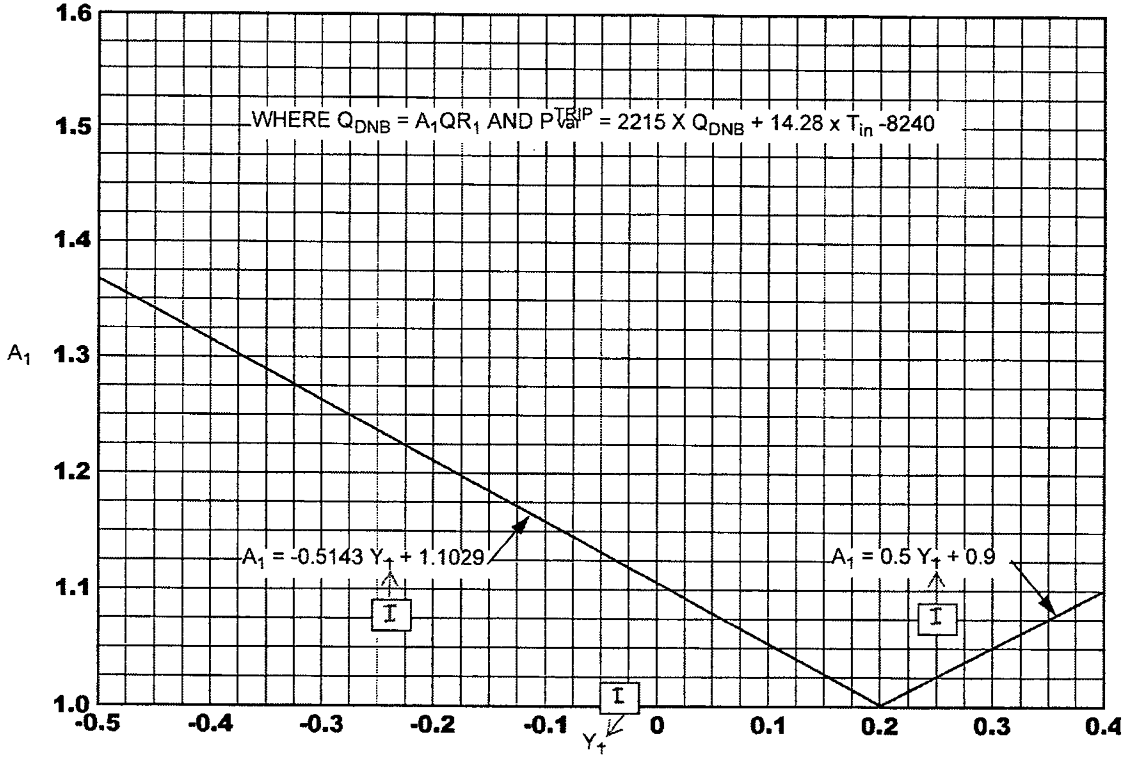

FIGURE 2.2-3 Thermal Margin/Low Pressure Trip Setpoint Part 1 (Y_1 versus A_1)

I

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40

~~May 12, 1979~~

~~March 16, 2006~~

TABLE 3.3-11 (Continued)

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels less than the MINIMUM CHANNELS OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 30 days or be in HOT STANDBY within the next 12 hours.

ACTION 2 - With the number of channels OPERABLE less than the MINIMUM CHANNELS OPERABLE, determine the subcooling margin once per 12 hours.

ACTION 3 - With any individual valve position indicator inoperable, obtain quench tank temperature, level and pressure information, and monitor discharge pipe temperature once per shift to determine valve position. This ACTION is not required if the PORV block valve is closed with power removed in accordance with Specification 3.4.3.~~a~~ b or 3.4.3.~~b.~~ c

ACTION 4 - a. With the number of OPERABLE accident monitoring instrumentation channels less than the total number of channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 7 days, or submit a special report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction, the plans for restoring the channel(s) to OPERABLE status, and any alternate methods in affect for estimating the applicable parameter during the interim.

b. With the number of OPERABLE accident monitoring instrumentation channels less than the MINIMUM CHANNELS OPERABLE requirements of Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 48 hours, or submit a special report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction, the plans for restoring the channel(s) to OPERABLE status, and any alternate methods in affect for estimating the applicable parameter during the interim.

~~September 25, 2003~~

ACTION 5 - With the number of OPERABLE accident monitoring instrumentation channels less than the MINIMUM CHANNELS OPERABLE requirements of Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 48 hours, or ~~begin~~ be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 6 - With any channel of radiation monitoring instrumentation inoperable, portable hand-held radiation detection equipment will be used to assess radiation releases from the atmospheric dump valves and steam generator safeties subsequent to a steam generator tube rupture.

ACTION 7 - Restore the inoperable system to OPERABLE status within 7 days or be in COLD SHUTDOWN within the next 36 hours. (See the ACTION statement in Technical Specification 3.4.6.1.).

ACTION 8 - With the number of OPERABLE Channels one less than the MINIMUM CHANNELS OPERABLE in Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

1. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status; and

2. Restore the system to OPERABLE status at the next scheduled refueling; and

3. Initiate an alternate method of monitoring the Reactor Vessel inventory.

FACILITY STAFF (CONTINUED)

d. A radiation protection technician shall be on site when fuel is in the reactor. (Table 6.2-1)

e. ALL CORE ALTERATIONS after the initial fuel loading shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

f. Deleted

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971* for comparable positions. Exceptions to this requirement are specified in the Quality Assurance Program.

6.3.2 ~~If the operations manager does not hold a senior reactor operator license for Millstone Unit No. 2, then the operations manager shall have held a senior reactor operator license at a Pressurized Water Reactor and an individual serving in the capacity of the assistant operations manager shall hold a senior reactor operator license for Millstone Unit No. 2.~~

The operations manager or at least one operations middle manager shall hold a senior reactor operator license for Millstone Unit No. 2.

* As of November 1, 2001, applicants for reactor operator and senior reactor operator qualification shall meet or exceed the education and experience guidelines of Regulatory Guide 1.8, Revision 3, May 2000.

CORE OPERATING LIMITS REPORT

6.9.1.8 a. Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle.

3/4.1.1.1	SHUTDOWN MARGIN (SDM)
3/4.1.1.4	Moderator Temperature Coefficient
3/4.1.3.6	Regulating CEA Insertion Limits
3/4.2.1	Linear Heat Rate
3/4.2.3	~~Total Integrated Radial Peaking Factor~~ TOTAL UNRODDED INTEGRATED RADIAL PEAKING FACTOR - F^T_r
3/4.2.6	DNB Margin

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1) EMF-96-029(P)(A) Volumes 1 and 2, "Reactor Analysis System for PWRs Volume 1 - Methodology Description, Volume 2 -Benchmarking Results," Siemens Power Corporation.

2) ANF-84-73 Appendix B (P)(A), "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," Advanced Nuclear Fuels.

3) XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company.

4) XN-75-32(P)(A) Supplements 1 through 4, "Computational Procedure for Evaluating Fuel Rod Bowing," Exxon Nuclear Company.

5) E~~FN~~ MF-2328(P)(A), "PWR Small Break LOCA Evaluation Model S-RELAP5 Based," Framatome ANP.

6) EMF-2087(P)(A), "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications," Siemens Power Corporation.

7) XN-N~~F~~ 78-44(NP)(A), "A Generic Analysis of the Control ~~r~~ Rod Ejection Transient for Pressurized ~~w~~ Water ~~r~~ Reactors," Exxon Nuclear Company.

CORE OPERATING LIMITS REPORT (CONT.)

8) XN-NF-621(P)(A), "Exxon Nuclear DNB Correlation for PWR Fuel Designs," Exxon Nuclear Company.

9) XN-NF-82-06(P)(A), and Supplements 2, 4 and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Exxon Nuclear Company.

10) ANF-88-133(P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels PWR Design Methodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation.

11) XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company.

12) ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation.

13) EMF-1961 (P)(A), "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors," Siemens Power Corporation.

14) EMF-2 31 ~~13~~0(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," Framatome ANP.

15) EMF-92-153(P)(A) and Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Power Corporation.

16) EMF-92-116(P)(A) Revision 0, "Generic Mechanical Design Criteria for PWR Fuel Designs," Siemens Power Corporation.

c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as SHUTDOWN MARGIN, and transient and accident analysis limits) of the safety analysis are met.

d. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

STEAM GENERATOR TUBE INSPECTION REPORT

6.9.1.9 A report shall be submitted within 180 days after initial entry into MODE 4 following completion of an inspection performed in accordance with TS 6.26, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each degradation mechanism,

f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator.

g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, one copy to the Regional Administrator, Region I, and one copy to the NRC Resident Inspector within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Deleted

b. Deleted

c. Deleted

d. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

e. Deleted

f. Deleted

g ~~a.~~ RCS Overpressure Mitigation, Specification 3.4.9.3.

6.15 RADIOLOGICAL EFFLUENT MONITORING AND OFFSITE DOSE CALCULATION MANUAL (REMODCM)

a. The REMODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and

b. The REMODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release, reports required by Specification 6.9.1.6a and specification 6.9.1.6b.

Licensee initiated changes to the REMODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

   2. a determination that the change(s) will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I of 10 CFR 50, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after review and acceptance by ~~SORC~~ FSRC and the approval of the designated officer; and

c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire REMODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the REMODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.