NMC
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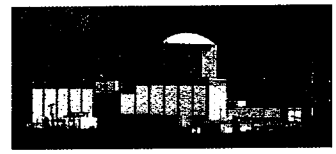
Kewaunee Nuclear Power Plant
N490, State Highway 42
Kewaunee, WI 54216-9511
920-388-2560

Operated by
Nuclear Management Company, LLC



January 18, 2001

10 CFR 50.90

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Docket 50-305
Operating License DPR-43
Kewaunee Nuclear Power Plant
Proposed Amendment 172 to Kewaunee Nuclear Power Plant Technical Specifications

References:
1) NRC Letter from William O. Long to M. L. Marchi, "AMENDMENT NO. 142 TO FACILITY OPERATING LICENSE NO. DPR-43-KEWAUNEE NUCLEAR POWER PLANT (TAC NO. MA1557)," dated December 2, 1998

2) NMC Letter from Kenneth H. Weinhauer to NRC Document Control Desk, "Wisconsin Public Service Corporation Reload Safety Evaluation Methods Topical Report WPSRSEM-NP, Revision 3," dated October 12, 2000

Nuclear Management Company, LLC, (NMC) proposes to amend Kewaunee Nuclear Power Plant (KNPP) Facility Operating License DRP-43 by incorporating the attached changes into Technical Specification (TS) 3.10.m for Reactor Coolant Minimum Flow.

NMC plans to replace existing Westinghouse Model 51 original steam generators (OSG) with Westinghouse Model 54F replacement steam generators (RSG) in the fall of 2001. Plugging and repair of OSG tubes during their operating life has diminished flow-performance. This reduction in performance required amendments to TS 3.10.m from time to time in order to make Reactor Coolant Minimum Flow consistent with diminished flow. Now that the OSGs are being replaced, it is necessary to again amend TS 3.10.m to return the minimum flow value to one appropriate for the new, non-repaired, RSGs. The last amendment (Reference 1) to TS 3.10.m reduced Reactor Coolant Minimum Flow to 85,500 gpm average flow per loop. KNPP is now requesting NRC approval for return of Reactor Coolant Minimum Flow to a value of 93,000 gpm for the RSGs. This value is 440 gpm greater flow than the original value for the OSGs when new of 92,560 gpm; a slight difference in the conservative direction.

Design transient analyses used as the bases for the Reactor Coolant Minimum Flow value proposed in this amendment request were calculated using RETRAN 3D in the 2D mode. NRC permission for KNPP to use this methodology was requested (Reference 2) and discussions contained herein assume approval of that request.

In accordance with 10 CFR 50.90, this letter requests Nuclear Regulatory Commission (NRC) approval to change KNPP TS 3.10.m to the new Reactor Coolant Minimum Flow value. This TS change is needed for operation with the RSGs and for Cycle 25 core design. NMC asks that the NRC approve this amendment by September 1, 2001, for implementation after unit shutdown for steam generator replacement and before commencement of dilution to reactor criticality.

Nothing in this letter should be construed to constitute a commitment or redefine a margin to safety unless specifically so stated in separate correspondence or in safety analyses of record.

In accordance with 10 CFR 50.30(b), a signed and notarized affidavit is included herewith. Additionally, NMC has transmitted a copy of this license amendment request to the State of Wisconsin as required by 10 CFR 50.91(b)(1).

If there are questions regarding this amendment, please contact either Mr. Thomas J. Webb at (920) 388-8537 or me at (920) 755-7627.

Sincerely,

Richard L. Weed

for Mark E. Reddemann
Site Vice President

MTVN

Attachments:
1. Description of Change, Safety Evaluation, Significant Hazards Determination, and Statement of Environmental Considerations
2. Summary of LOCA and Non-LOCA Safety Analyses for KNPP Steam Generator Replacement
3. Current affected page, TS 3.10-10, annotated with the change
4. Technical Specification page TS 3.10-10 as amended

cc - US NRC Region III
US NRC Senior Resident Inspector
Electric Division, PSCW

Subscribed and Sworn to
Before Me This 18 Day
of January 2001.

Notary Public, State of Wisconsin

My Commission Expires:
October 24, 2004

ATTACHMENT 1

Letter from M. E. Reddemann (NMC)

To

Document Control Desk (NRC)

Dated

January 18, 2001

<u>Proposed Amendment 172</u>

Description of Proposed Changes

Safety Evaluation

Significant Hazards Determination

Environmental Consideration

Introduction

Nuclear Management Company, LLC, (NMC) intends to replace the Kewaunee Nuclear Power Plant's (KNPP) Westinghouse Model 51 original steam generators (OSG) with Westinghouse Model 54F replacement steam generators (RSG) in the fall of 2001.

As a part of the KNPP Steam Generator In-Service-Inspection (ISI) Program, tubes in the KNPP steam generator were plugged or repaired during their operating life. Tube plugging and repair caused diminished flow, which made it necessary to amend the Technical Specification (TS) 3.10.m for Reactor Coolant Minimum Flow to represent the lower flow. Most recently, the NRC approved Amendment No. 142 (Reference 1) to decrease the TS value to 85,500 gallons per minute (gpm) average flow per loop.[1] Now that new steam generators are being installed, it is again necessary to amend the TS to raise the value to one compatible with new steam generators.

The Model 54F steam generators have been designed to retain the major design functions, thermal performance, and overall physical size of the Series 51. This limits the effect of RSGs on the KNPP design and licensing basis and permits the replacement of steam generators to be conducted within the limitations provided by Title 10 of the Code of Federal Regulations, Section 50.59 (10 CFR 50.59), "Changes, Tests and Experiments."

The most significant change in Model 54F steam generators, with respect to the Series 51 steam generators, is in the tube bundle. Alloy 690 was selected for manufacture of Model 54F tubes because of its superior resistance to corrosion. Since Alloy 690 has a lower heat transfer coefficient than the Alloy 600 used in the Model 51, the replacement bundle has been sized to provide a larger heat transfer area than the original steam generators. This larger tube bundle can produce the same steam pressure as the original tube bundle at a lower operating temperature or can provide more tube-plugging margin under the same operating conditions. It also provides the added benefit of increased reactor coolant flow.

The original TS 3.10.m Reactor Coolant Minimum Flow value of 92,560 gpm for the OSGs was diminished to the current value of 85,500 gpm as a result of tube plugging and repair. Since the RSGs offer greater total flow, the TS Reactor Coolant Minimum Flow value for the new machines was rounded to 93,000 gpm. This value is slightly more conservative.

Operation with the Model 54F steam generator installed as a component of the KNPP nuclear steam supply system was evaluated and analyzed with respect to all affected design basis transients and the results are contained in the Licensing Report (Reference 2). Since the reactor coolant flow characteristics for the RSG closely approximate those of the Series 51 OSG, no changes to accommodate flow differences are required. Results of the evaluation and analysis of design basis accidents and transients demonstrate that the 93,000 gpm value for TS 3.10.m Reactor Coolant Minimum Flow proposed in this amendment request remains bounded and causes no adverse affect on safety.

NRC approval of this TS change is requested by September 1, 2001, to support steam generator replacement and Cycle 25 reload changes to ensure consistency in reload design, safety analyses, and technical specification operating limits.

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[1] All values of TS 3.10.m "Reactor Coolant Minimum Flow" are in gallons per minute average flow per loop.

Description of Change to TS 3.10.m, "Reactor Coolant Flow"

TS Section 3.10.m is being revised as follows:

> Paragraph 3.10.m.1 is revised to say "...reactor coolant flow rate shall be ≥ 93,000 gallons per minute average per loop. If reactor coolant flow rate is < 93,000 gallons per minute per loop, ..."

Safety Evaluation for Proposed Change to TS 3.10.m

The RSG has been designed to retain the major design functions, thermal performance, and overall physical size of the OSG. This limits the effect of the RSG on KNPP design and licensing basis and permits replacement of steam generators to be accomplished within the limitations provided by Title 10 of the Code of Federal Regulations, Section 50.59 (10 CFR 50.59), "Changes, Tests and Experiments."

The change will not adversely affect plant equipment important to safety. Equipment important to safety will continue to perform its design function. Thus, the change does not involve an unreviewed safety question and current design basis analyses (Reference 2) bound its effect. Attachment 3 contains a summary of results and a list of LOCA Non-LOCA safety evaluations.

The original TS 3.10.m Reactor Coolant Minimum Flow value for the OSG was 92,560 gallons per minute average per loop, but has been reduced over time to the current value of 85,500 gpm due to plugging and repair. This amendment to TS 3.10.m increases the specification for Reactor Coolant Minimum Flow from 85,500 gpm to 93,000 gpm for the RSGs, setting it to a value similar to that originally licensed for the OSGs. All current KNPP analyses of record are performed assuming a Reactor Coolant Thermal Design Flow (TDF) lower than and less conservative than the TS value of 93,000 gpm. The results of these analyses bound results at the proposed TS Reactor Coolant Minimum Flow value.

The TS Reactor Coolant Minimum Flow is a low flow limit that allows plant operation at values of Reactor Coolant flow greater than the limit. Because the proposed 93,000 gpm flow is greater than the analyzed value used as input for the safety analyses, the proposed Reactor Coolant Minimum Flow value is bounded by the safety analyses. For instance, margins to departure-from-nucleate-boiling-ratio (DNBR) limit and fuel peak-centerline-temperature (PCT) safety limit during transients and accidents are increased. Thus, the proposed value for TS 3.10.m is conservative, conforms to existing design bases, and does not alter the result of any bounding safety analysis. Since the design basis safety analyses for Large and Small Break LOCA accidents and non-LOCA accidents have been analyzed or evaluated for the RSGs (Reference 2) at TDF and bound the proposed TS 3.10.m value for Reactor Coolant Minimum Flow, all safety analysis acceptance criteria are satisfied at the proposed TS 3.10.m Reactor Coolant Minimum Flow. Thus, the proposed change to the value of Reactor Coolant Minimum Flow does not involve an unreviewed safety question.

Significant Hazards Determination for Proposed Change to TS 3.10.m

Nuclear Management Company, LLC, (NMC) reviewed the proposed change in accordance with provisions of 10 CFR 50.92 and determined that it involves no significant hazards consideration (Reference 2). The proposed change does not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The change in Reactor Coolant Minimum Flow value for TS 3.10.m proposed in this amendment request is needed to reflect operating characteristics of the new RSGs. Accident analyses affected by the RSGs have each been evaluated to establish that there is no significant change in the documented results (Attachment 3). These evaluations have shown that the proposed value for Reactor Coolant Minimum Flow is bounded by the Thermal Design Flow value used in the analyses and provides greater margin to safety analysis acceptance criteria (e.g., DNB). All safety analysis acceptance criteria are satisfied. Since Reactor Coolant flow values for the RSG conform to the design bases and are bounded by the existing safety analyses, changing the technical specification within limits of the bounding accident analyses will not cause an increase in the probability or consequences of an accident previously evaluated.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change is fully consistent with current plant design bases and does not adversely affect any fission product barrier, nor does it alter the safety function of safety related systems, structures, and components depended upon for accident prevention or mitigation. Thus, it does not create the possibility of a new or different kind of accident.

3) Involve a significant reduction in the margin of safety.

   The proposed change does not alter the manner in which Safety Limits, Limiting Safety System Setpoints, or Limiting Conditions for Operation are determined. It returns TS 3.10.m for Reactor Coolant Minimum Flow to a value slightly higher, thus more conservative, than the value specified for the OSG when new. It conforms to plant design bases, is consistent with current safety analyses, and limits actual plant operation. Analysis of the effect of the proposed Reactor Coolant Minimum Flow limitation on LOCA and non-LOCA transients determined that all safety analysis acceptance criteria are satisfied at a TDF that bounds the revised Reactor Coolant Minimum Flow and all KNPP safety requirements continue to be met. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Environmental Considerations

This proposed amendment involves a change to the Technical Specifications. It does not modify any facility components located within the restricted area, as defined in 10 CFR 20. NMC has determined that the proposed amendment involves no significant hazards considerations and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in the individual or cumulative occupational radiation exposure. This proposed amendment accordingly meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with this proposed amendment.

References:

1. NRC Letter from William O. Long to M. L. Marchi, "AMENDMENT NO. 142 TO FACILITY OPERATING LICENSE NO. DPR-43-KEWAUNEE NUCLEAR POWER PLANT (TAC NO. MA1557), dated December 2, 1998
2. Kewaunee Nuclear Power Plant Steam Generator Replacement and T_{avg} Operating Window Program Licensing Report, November 2000, by Westinghouse Electric Company, LLC

ATTACHMENT 2

Letter from M. E. Reddemann (NMC)

To

Document Control Desk (NRC)

Dated

January 18, 2001

Proposed Amendment 172

Summary of LOCA and Non-LOCA Safety Analyses for
KNPP Steam Generator Replacement

Kewaunee Nuclear Power Plant
Replacement of Steam Generator Lower Assemblies
and Modification of Steam Domes

Safety Evaluation Summary: **Non - LOCA Accidents**

References:

1. SGR DCR2858Mod1, Folder III, Safety Evaluation Reports (SERs)

2. SGR Licensing Report

3. NRC letter 00-082 from K.H.Weinhauer to document control desk submitting WPSC KNPP topical report WPSRSEM-NP, Rev. 3, entitled, "Reload Safety Evaluation Methods for Application to Kewaunee," dated October 12, 2000

Introduction: Operation of Kewaunee Nuclear Power Plant (KNPP) Nuclear Steam Supply System (NSSS) with replacement steam generators (RSG) was evaluated according to the criteria of 10CFR50.59. The effect of RSGs on KNPP design basis Non- Loss of Coolant Accident (Non- LOCA) safety analyses is summarized below. These safety analyses include the design basis accidents described in chapter 14 of the KNPP Updated Safety Analysis Report (USAR) and the high energy line break (HELB) outside containment accident.

KNPP will replace the original Westinghouse model 51 SGs with Westinghouse model 54F SGs. Descriptions and safety evaluations of the changes in design features, thermal hydraulic performance, and materials associated with the RSGs are provided in reference 1. Operation of the NSSS with the model 54F SGs will affect the steady state conditions and transient response of the KNPP NSSS. Safety analyses for Non-LOCA accidents appropriately incorporate all physical and operational changes associated with the RSGs.

Design and safety aspects of operating with the RSGs are described in the SGR Licensing Report (reference 2). Section 6.8 documents the Non-LOCA safety analyses and provides the technical basis for the conclusions of this safety evaluation.

Description of Non – LOCA Safety Analysis Changes

Due to the scope of the DCR 2858 Mod 1 change, Kewaunee plant has reanalyzed all Non-LOCA design basis accidents affected by the RSG using assumptions consistent with RSG design and operating characteristics.

The following areas of transient analysis were evaluated to determine the effect of on design basis transients of operating with the RSGs:

- USAR Chapter 14 Non–LOCA design basis accidents
- high energy line break and auxiliary building compartment thermal hydraulic analysis
- containment integrity analysis for main steam line break accident
- Anticipated Transients Without Scram (ATWS) analysis.

This represents all KNPP non-LOCA design basis accidents.

Westinghouse plant performance capability (PCWG) analyses (reference 2, section 2) provide the bases for RSG plant input assumptions.

Safeguard systems' design and function, plant setpoints, fuel design, bounding core physics safety parameters, reactor power, and reactor Tave program are not changing in the year of RSG implementation. This is to reduce the number of changes due to DCR 2858 Mod1 and thereby simplify licensing changes. The overall strategy for RSG analyses is designed to support both first year RSG plant operation with minimal license and operational changes as well as potential future plant changes that are possible because of the RSG design.

Non LOCA Safety Analysis Methodology

The RSG Non LOCA safety analyses apply the analysis methods described in WPS Topical Report WPS-RSEM-NP revision 3 (reference 3). Changes in Analysis Methods for SGR are the subject of a Safety Evaluation Report (reference 1).

Evaluation Summary

A description of the KNPP design basis accidents by condition category, including design requirements, acceptance criteria, and the applicable design basis transient events, is provided in the Licensing Report for:

Condition II, Incidents of Moderate Frequency

Transient Events Reanalyzed for SGR

- Uncontrolled rod cluster control assembly (RCCA) withdrawal from sub-critical
- Uncontrolled RCCA withdrawal at power
- RCCA misalignment (dropped/static)
- Chemical and volume control system malfunction
- Startup of inactive reactor coolant loop
- Feedwater system malfunction
- Excessive load increase

- Partial loss of reactor coolant flow
- Loss of external load
- Loss of normal feedwater
- Loss of AC power to plant auxiliaries

Condition III, Infrequent Incidents

Transient Events Reanalyzed for SGR

- Small Break LOCA
- Small steam line break
- Complete loss of reactor coolant flow
- Volume control tank rupture

Condition IV, Limiting Faults

Transient Events Reanalysed for SGR

- Large Break LOCA
- Steam generator tube rupture
- Main steam line break (MSLB)
- Locked rotor
- RCCA ejection
- Fuel handling accident

KNPP performed safety analyses of design basis Non LOCA accidents using assumptions appropriate for the RSG design and operating characteristics. Results of the safety analyses satisfy all applicable acceptance criteria.

**Kewaunee Nuclear Power Plant
Replacement of Steam Generator Lower Assemblies
and Modification of Steam Domes**

Safety Evaluation Summary: **Loss of Coolant Accidents Analysis**

References:

1. DCR 2858, Modification Package 1, Design Description
2. USAR Sections - See Section 3.2 of the Design Description in DCR 2858, Modification Package 1
3. Technical Specifications – See Section 3.2 of the Design Description in DCR 2858, Modification Package 1
4. Westinghouse Design Specification 414A03, Rev. 2 "Model 54F Replacement Steam Generator – Complete Lower Assembly and Modified Upper Assembly"
5. Westinghouse Certified Design Report WCAP-15318,15319,15324,15374
6. Westinghouse Final Thermal Hydraulic Report WNEP-9902
7. SGR Licensing Report (Westinghouse Submittal Letter WPS-99-125)

Introduction: Operation of the Kewaunee Nuclear Power Plant (KNPP) with the proposed steam generator modifications has been evaluated using the guidance of Title 10 of the Code of Federal Regulations (CFR), Section 50.59 (10 CFR 50.59) and has been determined to not represent an unreviewed safety question. The following assessment is specific to the Small Break Loss of Coolant Accident (SBLOCA) and Large Break Loss of Coolant Accident (LBLOCA).

The SBLOCA and LBLOCA analyses and evaluations for the Kewaunee (WPS) replacement steam generator (RSG) and T_{ave} operating window program were performed using essentially the same analysis input assumptions as the existing analyses/evaluations in the USAR. The main exceptions are the steam generator type, the incorporation of a T_{ave} operating window; revised pressurizer and accumulator line data, and revised containment heat sink data. The Model 54F steam generator, the T_{ave} operating window, and revised pressurizer and accumulator line data were explicitly analyzed in the SBLOCA and LBLOCA analyses. The revised containment heat sink data was explicitly analyzed in the LBLOCA analysis. The SBLOCA analysis does not model a containment pressure transient; therefore, the revised containment heat sink data was not incorporated in the SBLOCA analysis.

The prior analyses for SBLOCA and LBLOCA modeled a Model 51 steam generator. The prior LBLOCA analysis explicitly analyzed a steam generator tube plugging (SGTP) of 30-percent. This same SGTP had been evaluated for SBLOCA. The new analyses for LBLOCA and SBLOCA have explicitly modeled a Model 54F steam generator at an SGTP of 24-percent. This SGTP was determined to be the maximum possible SGTP consistent with the desired thermal design flow (TDF) of 89,000 gpm/loop.

Previously, the SBLOCA and LBLOCA analyses modeled a single reactor coolant system (RCS) T_{ave} of 562.0°F with a measurement uncertainty of ± 4°F. The new analyses explicitly modeled a T_{ave} operating window of $554.1°F \leq T_{ave} \leq 575.3°F$ with a measurement uncertainty of ± 4°F. The new analyses for SBLOCA and LBLOCA have also incorporated revised pressurizer surge line and accumulator line data pursuant to NSAL-98-004. The revised piping data has been explicitly modeled in both the
SBLOCA and LBLOCA analyses for the RSG and T_{ave} operating window program.

Nuclear Management Company, LLC (NMC), supplied revised containment heat sink data in order to ensure that the minimum containment pressure response was modeled in the LBLOCA analysis.

Additionally, the SBLOCA and LBLOCA analyses reanalyzed two situations previously evaluated and presented in the USAR. The SBLOCA and LBLOCA analyses both analyzed the impact of the transition from Siemens 14x14 Standard fuel to Siemens 14x14 Heavy fuel and determined there was no peak cladding temperature (PCT) penalty associated with the transition cycles. The LBLOCA analysis also reanalyzed the impact of increasing the low power region power factor (P_{Low}) from 0.50 to 0.60 after 1500 MWD/MTU of cycle burnup and again determined that there was no PCT penalty associated with the P_{Low} increase.

The SBLOCA and LBLOCA analyses and evaluations performed for the RSG and T_{ave} operating window program yield results well within all established acceptance criteria.

The following summarizes the results of the LOCA evaluations:

Small Break LOCA

The High Tavg, 4-inch equivalent diameter limiting case peak cladding temperature calculated for the KNPP Appendix K SBLOCA analysis for the Model 54F RSG is 843°F. The analysis models a total peaking factor (FQ) of 2.50, a hot rod channel factor of 1.70, a hot assembly factor of 1.514. This result is below the acceptance criteria limit of 2200°F. The maximum local metal-water reaction is <1.0 percent, which is well below the embrittlement Acceptance Criteria limit of 17 percent. The limiting total core metal-water reaction is <1.0 percent, which is below the 1.0 percent limit, in accordance with the Acceptance Criteria. The cladding temperature transient is terminated at a time when the core geometry is still amenable to cooling. As a result, the core temperature will continue to drop and the ability to remove decay heat generated in the fuel for an extended period of time will be provided.

Large Break LOCA

The Low Tavg, Pumps Running limiting case peak cladding temperature calculated for the KNPP Appendix K LBLOCA analysis for the Model 54F RSG is 2038°F. The analysis models a total peaking factor (FQ) of 2.35, a hot rod channel factor of 1.70, a hot assembly factor of 1.514, and a low power/periphery region factor (Plow) of 0.50. The peak cladding temperature occurs during reflood at approximately 124 seconds. This result is below the acceptance criteria limit of 2200°F. The maximum local metal-water reaction is 6.58 percent, which is below the embrittlement Acceptance Criteria limit of 17 percent. The limiting total core metal-water reaction is 0.006 percent, which is much less than the 1.0 percent limit, in accordance with the Acceptance Criteria. The cladding temperature transient is terminated at a time when the core geometry is still amenable to cooling. As a result, the core temperature will continue to drop and the ability to remove decay heat generated in the fuel for an extended period of time will be provided.

Long Term LOCA Containment Response

Results of this analysis show that the containment pressure remains less than the limit of 46 psig.

The evaluation of the proposed modification concludes that it will not result in potential unreviewed safety questions, as defined in 10CF50.59, since it does not increase the probability or occurrence in the consequences of an accident in the Kewaunee USAR. Nor has any mechanism for an accident or malfunction, which has not been previously evaluated in the USAR, been identified. Also, the change does not decrease the margin of safety as identified in the basis for any Technical Specification.

Summary Determination of Unreviewed Safety Question for All KNPP LOCA and Non-LOCA Design Basis Accidents Performed in Conjunction with RSG

Changes to KNPP safety analyses affected by RSG installation were reviewed as required by Title 10 Code of Federal Regulations, §50.59 (10 CFR 50.59) and found to not represent an unreviewed safety question.

ATTACHMENT 3

Letter from M. E. Reddemann (NMC)

To

Document Control Desk (NRC)

Dated

January 18, 2001

Proposed Amendment 172

Strike Out Technical Specification Page

TS 3.10-10

m. Reactor Coolant Flow

1. During steady-state power operation, reactor coolant flow rate shall be ≥ ~~85,500~~93,000 gallons per minute average per loop. If reactor coolant flow rate is < ~~85,500~~93,000 gallons per minute per loop, action shall be taken in accordance with TS 3.10.n.

2. Compliance with this flow requirement shall be demonstrated by verifying the reactor coolant flow during initial power escalation following each REFUELING, between 70% and 95% power with plant parameters as constant as practical.

n. DNBR Parameters

If, during power operation any of the conditions of TS 3.10.k, TS 3.10.l, or TS 3.10.m.1 are not met, restore the parameter in 2 hours or less to within limits or reduce power to < 5% of thermal rated power within an additional 6 hours. Following analysis, thermal power may be raised not to exceed a power level analyzed to maintain a DNBR greater than the minimum DNBR limit.

ATTACHMENT 4

Letter from M. E. Reddemann (NMC)

To

Document Control Desk (NRC)

Dated

January 18, 2001

Proposed Amendment 172

Amended Technical Specification Page

TS 3.10-10

m. Reactor Coolant Flow

1. During steady-state power operation, reactor coolant flow rate shall be ≥ 93,000 gallons per minute average per loop. If reactor coolant flow rate is < 93,000 gallons per minute per loop, action shall be taken in accordance with TS 3.10.n.

2. Compliance with this flow requirement shall be demonstrated by verifying the reactor coolant flow during initial power escalation following each REFUELING, between 70% and 95% power with plant parameters as constant as practical.

n. DNBR Parameters

If, during power operation any of the conditions of TS 3.10.k, TS 3.10.l, or TS 3.10.m.1 are not met, restore the parameter in 2 hours or less to within limits or reduce power to < 5% of thermal rated power within an additional 6 hours. Following analysis, thermal power may be raised not to exceed a power level analyzed to maintain a DNBR greater than the minimum DNBR limit.