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SUBJECT: Provides clarifications to 870327 request for Tech Spec changes supporting operation following Spring 1987 reload, Cycle 3.

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April 22, 1987
G02-87-144

Docket No. 50-397

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

Gentlemen:

Subject: NUCLEAR PLANT NO. 2
OPERATING LICENSE NPF-21,
REQUEST FOR AMENDMENT TO TECHNICAL SPECIFICATIONS -
RELOAD LICENSE AMENDMENT (CYCLE 3)
CLARIFICATION

Reference:
1) Letter, G02-87-0105, GC Sorensen (SS) to NRC, same subject, dated March 27, 1987
2) WNP-2 Cycle 3 Plant Transient Analysis, XN-NF-87-24, March 1987
3) WNP-2 Cycle 3 Reload Summary Report, WPPSS-EANF-109, March 1987
4) WNP-2 Cycle 3 Reload Analysis, XN-NF-87-25, March 1987
5) Letter, G02-86-477, GC Sorensen (SS) to NRC, "Request for Amendment to Technical Specifications - Reload License Amendment (Cycle 2), Supplemental Information," dated May 22, 1986

Reference 1) provided a request for technical specification changes in support of operation for WNP-2 following the Spring 1987 reload, cycle 3. Staff review of this submittal has prompted several requests for clarification. Accordingly, the attachments to this letter provide the requested clarifications.

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Attachment 1 is a replacement for Section 3.2.3 of Reference 2) and further replaces the last paragraph, page 13 of Reference 3). This attachment presents the analysis of the Loss of Feedwater Heating (LOFWH) done specifically for WNP-2. Also, the footnote labeled *** at the bottom of page 4 (Table 2.1) of Reference 2) reading "Generic Analysis bounding value, Reference 9" should be changed to read simply, "Bounding analysis". Additionally this information supports the revision of the last sentence on page 6 of Reference 4) to read:

> "The loss of feedwater heating event was analyzed specifically for WNP-2, and the delta CPR value reported bounds the analysis"

Attachment 2, page 3/4 2-1, reflects cycle 2 changes to the technical specification requested by Reference 5) and approved in Amendment 28 to the WNP-2 Technical Specifications. However, by oversight, portions of the changes approved for cycle 2 were not included in Amendment 28. Hence, because attachment 2 is also affected by the cycle 3 requested technical specification changes it is being submitted reflecting both cycle 2 and cycle 3 changes and replaces the page forwarded in attachment 1 to Reference 1).

As requested by Mr. M. McCoy of your staff, attachment 3 provides a comparison of the ANF and GE methodologies in establishing MCPR safety limits and supports the ANF calculation of 1.06 as the MCPR Safety Limit for GE fuel in the Reference 1) reload information.

Attachment 4, again requested by Mr. McCoy, provides a discussion of the use of ASEA-ATOM fuel channels at WNP-2 and by the BWR industry and provides a comparison with GE fuel channels.

Reference 1) identified June 15, 1987 as the date for re-commencing commercial operations. In order to support a June 15, 1987 startup, the date the requested changes will be needed is June 4, 1987. This allows sufficient time for testing and confirming system integrity without impacting commercial operations.

Should you have any questions, please contact Mr. P. L. Powell, Manager, WNP-2 Licensing.

Very truly yours,

G. C. Sorensen, Manager
Regulatory Programs

cc: JB Martin - NRC RV
JO Bradfute - NRC
C Eschels - EFSEC
CE Revell - BPA
NRC Site Inspector
RB Samworth - NRC

ATTACHMENT 1

Section 3.2.3, XN-NF-87-24

WNP-2 LOSS OF FEEDWATER HEATING ANALYSIS

The analysis of the loss of feedwater heating event was performed specifically for the WNP-2 reactor. The analysis was performed for a total of 9 cases wherein the cycle exposure as well as the initial conditions of core power and flow were varied. The range of cycle exposure, core power, and core flow considered in the analysis reflect reactor operation over the proposed operating power versus flow map and conditions anticipated during actual WNP-2 reactor operation. The range of conditions is presented in Table 1.

A data summary for all of the WNP-2 specific analysis is shown in Table 2. This table provides the conditions of each case analyzed in terms of cycle exposure, core power, and core flow. The initial and final MCPR values and the associated delta MCPR value are presented for each case.

The data presented in Table 2 indicates a variation in the initial MCPR value for the conditions analyzed. This variation is anticipated due to the changes in the reactor operating conditions (reactor power and flow) analyzed as well as the change in cycle exposure. Thus, a procedure was developed to bound this variation and provide a conservative value of the initial MCPR value which would preclude penetration of the MCPR safety limit of 1.06 during a loss of feedwater heating event. This procedure consists of determing the largest value of delta MCPR/initial MCPR for the entire WNP-2 data base and calculating the initial MCPR value corresponding to a final MCPR value of 1.06 (MCPR safety limit). Examination of the data presented in Table 2 indicates that the largest delta MCPR/initial MCPR value is 0.0729. This value of delta MCPR/initial MCPR results in an initial MCPR value of 1.15 in order to preclude penetration of the 1.06 value during the loss of feedwater heating event. This calculation is illustrated as:

$$\left(\frac{\text{Delta MCPR}}{\text{Initial MCPR}}\right)_{max} = 0.0729$$

or

$$\frac{\text{Initial MCPR - Final MCPR}}{\text{Initial MCPR}} = 0.0729$$

To preclude penetration of the MCPR safety limit, the final MCPR is set equal to 1.06. Thus,

$$\frac{\text{Initial MCPR - 1.06}}{\text{Initial MCPR}} = 0.0729$$

Solving for the initial MCPR, a value of 1.15 is achieved.

TABLE 1 RANGE OF OPERATING CONDITIONS
WNP-2 LOFH EVENT

Variable	Range
Cycle Exposure (GWd/MT)	0 to 6.10
Core Power (% Rated)	50 to 100
Core Flow (% Rated)	33 to 100

TABLE 2 LOFH TRANSIENT ANALYSIS DATA SUMMARY - WNP-2 REACTOR

Case Number	Cycle Exposure (GWd/MT)	Core Power (% Rated)	Core Flow (% Rated)	MCPR Initial	MCPR Final	Delta MCPR	Delta MCPR/ Initial MCPR
1	0.00	100.00	100.00	1.523	1.412	.111	.0729
2	2.00	100.00	100.00	1.447	1.352	.095	.0657
3	4.00	100.00	100.00	1.445	1.349	.096	.0664
4	6.10	100.00	100.00	1.414	1.325	.089	.0629
5	2.24	99.79	97.28	1.410	1.315	.095	.0674
6	2.58	99.85	100.30	1.523	1.421	.102	.0670
7	3.57	54.56	33.18	1.799	1.668	.131	.0728
8	3.95	49.92	45.97	2.015	1.888	.127	.0630
9	4.53	72.80	53.92	1.604	1.507	.097	.0605

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3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

ANF

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE for GE fuel and average bundle exposure for [redacted] fuel shall not exceed the limits shown in Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3 WHEN IN TWO LOOP OPERATION AND ~~The limits for single loop operation are shown in~~ Figures 3.2.1-4, 3.2.1-5, and 3.2.1-6 WHEN IN SINGLE LOOP OPERATION.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

APPLICABLE

With an APLHGR exceeding the limits of Figure 3.2.1-1, 3.2.1-2, or 3.2.1-3 IN TWO LOOP OPERATION OR FIGURE 3.2.1-4, 3.2.1-5, OR 3.2.1-6 IN SINGLE LOOP OPERATION, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits determined from Figures 3.2.1-1, 3.2.1-2, ~~and~~ 3.2.1-3, 3.2.1-4, 3.2.1-5 AND 3.2.1-6,

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

ATTACHMENT 3

Table S.2-1 of NEDO-24011-A-4-United States supplement lists the uncertainties utilized by General Electric in deriving the MCPR safety limit. Notes 2 and 3 indicate the differences between initial core uncertainties and reload core uncertainties. The differences are an increase in "TIP Uncertainty" of 2.4% and an increase in R Factor by .1%. The applicable paragraphs and notes are included below:

TIP Readings	8.7(2)	These sets of data are the base from which gross power distribution is determined. The assigned uncertainties include all electrical and geometrical components plus a contribution from the analytical extrapolation from the chamber location to the adjacent fuel assembly segment. Also included are uncertainties contributed by the LPRM system. LPRM readings are used to correct the power distribution calculations for changes which have occurred since the last TIP survey. The assigned uncertainty affects power distribution in the same manner as the base TIP reading uncertainty.
R Factor	1.6(3)	This is a function of the uncertainty in local fuel rod power.

(2) This value is for reload cores only. For initial cores, this is 6.3%.

(3) This value is for reload cores only. For initial cores, this is 1.5%.

It can be seen from the TIP reading uncertainty that this value is a lumped parameter which contains both electrical and geometric uncertainties plus LPRM uncertainties and a contribution from the analytical extrapolation from the chamber location to the adjacent fuel assembly segment. The geometrical and electrical uncertainties would not change appreciably from cycle to cycle however, the correlation from TIP reading to cover average pin power is critical in the GE process computers calculation of power distribution.

The ANF technique of determing core power distribution also uses the TIP to calibrate the LPRM, hence geometric, electrical and LPRM uncertainties need to be accounted for. However, unlike the GE P1 process computer, POWERPLEX (a core simulator) calculates core power distribution on-line in real time using ANF's NRC approved neutronics 3D design code. The uncertainties in core power measurements in the core simulator have been addressed by ANF as power measurement uncertainties in the ANF Topical Report XN-NF-80-19(A), Vol. 1. Utilizing these uncertainties, a MCPR safety limit has been justified which is lower than that justified for a core monitored by the GE P1 process computer methodology.

ATTACHMENT 4

ASEA-ATOM FUEL CHANNEL COMPATIBILITY

For over 20 years ASEA-ATOM has been manufacturing channels for the BWR industry. Through May 1986, they have delivered 15,678 channels to 25 different BWRs world wide. This includes 40 channels delivered to Quad-Cities in 1983 and 12 delivered to Fitzpatrick in 1986.

The ASEA-ATOM channels supplied to WNP-2 for use on assemblies for its Reload 2 fuel, cycle 3, are fully compatible with the original equipment channels. The new fuel channels made by ASEA-ATOM are of the same material and fit the same dimensional envelope when compared to those supplied by General Electric Company for the WNP-2 original core fuel. This is shown in the table below. Manufacturing tolerances are also similar.

TABLE I
COMPARISON OF ASEA-ATOM AND GE ORIGINAL CORE CHANNELS

	ASEA-ATOM	GE ORIGINAL
Dimension Data:		
Length (in.)	166.91	166.91
Width (in.)	5.278 X 5.278	5.278 X 5.278
Thickness (in.)	.100	.100
Relieved Area (in.) (to match Upper Guide Plate)	.020 deep 20.5 long 2 sides	.020 deep 19.85 min. long 2 sides
Spacer Button Dimensions (in.)	2.70 X 1.24 X .200	2.70 X 1.24 X .200
Material Properties:		
Material	Zircaloy-4	Zircaloy-4

For WNP-2 a pre-purchase examination of the specifications of the ASEA-ATOM proposed channels found all differences between the two channel types to be extremely small. It was determined that all differences that were identified would not affect or compromise the anticipated performance of the channels in any way. A subsequent examination of the as-delivered product substantiated this claim.