Regulatory Docket File

Consumers Power Company

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Director of Nuclear Reactor Regulation
Att: Mr Robert A. Purple, Chief
Operating Reactor Branch No 1
US Nuclear Regulatory Commission
Washington, DC 20555

DOCKET 50-255, LICENSE DPR-20
PALISADES PLANT, ANSWERS TO
QUESTIONS ON TECHNICAL SPECIFICATIONS

Your letter dated March 26, 1976 asked a number of questions concerning the Palisades Plant Technical Specifications and relating to reloading (Cycle 2) of the Palisades reactor. These questions (presented to aid in the discussion) and our responses are enclosed as Attachment 1.

Some of the responses to these questions indicate that appropriate changes to our Plant Technical Specifications should be made. Accordingly, proposed Technical Specifications Changes are enclosed as Attachment 2.

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3448

ATTACHMENT 1

Answers to Technical Specifications Questions

Relating to the Reloading (Cycle 2)

of the Palisades Reactor

April 1976

## SECTION 5.0 TECHNICAL SPECIFICATIONS

Question 5.1

Because the system pressure has been lowered to 1800 psia, more restrictive control rod insertion limits, as set forth in Figure 2-4 of the Technical Specifications, should apply. Either explain the discrepancy between Figure 2-4 and Figure 3-6 or replace Figure 3-6 with Figure 2-4. Justification of whichever figure is ultimately used must be provided.

Response

Figure 2-4 represents the rod insertion limits presently planned for Core II. It is identical to the curve that was used during the last half of Cycle 1. It is believed to represent the minimum rod insertion that allows reasonable flexibility in the control of the reactor. ECCS acceptance criteria and to some extent thermal limits at reduced operating pressure require that the core be operated with a relatively low total peaking factor. Because of the relatively wide variety of combinations of power levels, times, transient xenon and possibly axially unstable xenon that may occur during the life of Core II, it is unlikely that any control rod insertion limit curve can provide, by itself, assurance that core peaking requirements will be met.

In recognition of this fact, provisions for both periodic and continuous monitoring of core power distribution are provided in other portions of the Technical Specifications.

By inspection of the proposed curve, it can be seen that it would extrapolate to essentially zero rod insertion at 100% power. In order to allow reasonable control of the plant and to provide for control of unstable xenon, should it occur, the curve goes vertical at 80% allowing 25% insertion of Group 4 at any power level above 80%. By inspection of Figure 3-14 in the FSAR, it can be seen that relatively minor increases in peaking would be expected from 25% insertion of Group 4. Operation with this rod insertion curve during Cycle 1 has shown that very low total peaking factor (maximum LHGR of 11 to 13 kW/ft at full power) can be achieved.

Figure 3-6 and the rest of Section 3.10 were not modified because it is our intention to return to normal operating pressure during Cycle 2. (An appropriate Technical Specifications Change will be requested at that time.) With this change, the assumed conditions of Section 3.10 would be restored. Since it appears that ECCS considerations will be more limiting in Cycle 2, the thermal limits of Section 3.10 will be well covered by the more restrictive Section 3.18 limits on core power distribution.

Question 5.2

State whether the limiting safety system settings given in Figures 2-1, 2-2, and 2-3 are based on the reduced flow due to the additional steam generator tube plugging. If they are not, provide appropriately revised figures or justification for using the figures as they exist.

Response

The reactor core safety limits shown in Figures 2-1, 2-2, and 2-3 of the Palisades Technical Specifications are based on initial design flow and a total core peaking factor of 3.62. The proposed Technical Specifications flow rate of $124 \times 10^6$ lb/h is less than 1% below the design flow rate of $125 \times 10^6$ lb/h.

To investigate the suitability of Figures 2-1, 2-2, and 2-3 for Cycle 2, Exxon Nuclear performed a DNB analysis using the same techniques as used in XN-76-3. The conditions for this analysis were as follows:

Core Power: 102% (2244 MWth)
Pressure: 1750 Psia
Core Flow: $118.4 \times 10^6$ lb/h

The nuclear peaking factors used were consistent with those used in XN-76-3 Supplement 1.

Core inlet temperature was raised until a MDNBR of 1.3 was achieved. This MDNBR was reached at a $T_{in}$ of 569°F. Figure 2-3, by comparison, shows a limit of approximately 546°F for the same condition. Clearly, the lower nuclear peaking factors allowed for Cycle 2 more than accommodate the slight reduction in flow.

It is concluded that the reactor core safety limits shown in Figures 2-1, 2-2, and 2-3 are appropriately conservative for operation in Cycle 2.

Question 5.3

The basis for Section 2.3.4 on Page 2-9 of the Technical Specifications refers to plant operation at 45% of rated power for three pump operation and 25% of rated power for two pump operation. Since such plant operation at these power levels will no longer be permitted, modify the basis accordingly.

Response

Consumers Power Company will be performing an ECCS analysis with respect to operation with less than four pumps. Until this analysis has been completed and approved, plant operation is being limited to four pump operation by other parts of the Technical Specifications. We believe that the basis presented in Section 2.3.4 will be shown to be appropriate, by the additional evaluation, and that the basis should be left unchanged. Changes deemed appropriate (if any) will be proposed with the results of the analysis for operation with less than four pumps.

Question 5.4

State whether the flows given in Items F and G of the January 30, 1976 letter reflect the additional steam generator tube plugging. If not, modify the flows accordingly.

Response

The minimum flow specified in Items F and G of our January 30, 1976 letter did include an allowance for tube plugging, but the actual number of tubes plugged exceeds this allowance. Appropriate revisions to the proposed Technical Specifications Changes have been made and are proposed as a separate attachment (see Attachment 2, Items A and B).

Question 5.5

The proposed Technical Specifications for Palisades Cycle 2 are based on measured core flow rate (with 3.0% margin) rather than the design flow rate. The use of measured rather than design flow rate is acceptable to the staff provided:

(1) The flow measurement technique is acceptable.

(2) An error analysis is provided for the data measured.

(3) There is adequate conservatism between the flow rate actually measured and the flow which was used to develop plant Technical Specifications.

(4) There is periodic surveillance and/or additional testing to assure that the core flow rate does not decrease as a result of crud buildup, steam generator tube plugging, or other causes.

Based on these considerations, provide the following:

(1) A description of the flow measurement technique used along with the data measured and an error analysis for the measurements.

(2) A discussion of the bases for the power to flow ratio reactor trip setting and the overpower trip setting (if it is based on measured flow).

(3) A proposed surveillance and/or test program to confirm that the value of the core flow rate has not decreased below the value used as the basis for reactor power/flow trip (and/or the overpower trip), including appropriate uncertainties.

Response

(1) The flow measurement technique and an error analysis substantiating the conservatism in assuming a 3% measurement uncertainty were submitted to the NRC on March 21, 1975. No data is yet available regarding primary coolant pump differential pressure after the latest steam generator tube plugging campaign.

(2) The coefficients used in the thermal margin/low pressure trip circuit will be evaluated based upon the flow measured at the start of Cycle 2 operation. If changes to these coefficients are necessary to prevent operation of the reactor when the DNBR is less than 1.3, the changes will be made during the power test program prior to achieving 75% power.

The reactor trip on high neutron flux (106.5%) is not changed, as the basis for using a margin to overpower of 122% in the DNBR analysis includes this set point.

(3) The flow monitoring equipment used to measure primary coolant flow at Palisades is installed on a temporary basis inside containment to gain the necessary data. The steam generator pressure differential flow monitoring equipment, which is permanent, is not sufficiently accurate to be used for flow monitoring surveillance.

Consumers Power Company proposes that a measurement of primary coolant flow with four primary coolant pumps in operation be made after each refueling and/or steam generator tube plugging. This surveillance provides assurance that the value of core flow rate used in the thermal margin/low pressure trip setting is conservative compared to measured core flow.

An appropriate proposed Technical Specifications Change may be found in a separate attachment (see Attachment 2, Item C).

Question 5.6

Provide a table in the Specifications which lists the maximum cold leg temperature, the minimum pressurizer pressure and the minimum reactor coolant flow rate for the various number of reactor coolant pumps for which reactor operation is permitted. These limits should show that the values used in the transient and accident analysis will not be exceeded during plant operation. In a separate document specify how these parameters are measured and provide the uncertainty associated with these measurements.

Response

The maximum cold leg temperature is currently specified in Paragraph 8 of Appendix B to the Palisades Technical Specifications. This value is 525°F, and applies to all steady state operation above 80% power. The plant transient analyses (XN-75-67), DNB analyses (XN-76-3) and LOCA analyses (XN-76-4) all conservatively assume a cold leg temperature of 530°F. A description of how cold leg temperature is measured may be found in Section 7.4.2.1 of the FSAR.

The maximum nominal system primary pressure is specified in Paragraph 4 of Appendix B to the Palisades Technical Specifications. This value is 1800 psia. The plant transient analyses (XN-75-67) and DNB analyses (XN-76-3) conservatively assume an operating pressure of 1750 psia. The LOCA analyses (XN-76-4) assume a pressure of 1800 psia. The time-to-collapse calculations submitted on March 20, 1976 in response to Question 1.1 conservatively assume an operating pressure of 1850 psia. A description of how pressurizer pressure is measured may be found in Section 7.4.2.1 of the FSAR.

The minimum pressurizer pressure for various pump combinations is specified in Table 2.3.1 and Figures 2-1, 2-2, and 2-3 of the Palisades Technical Specifications.

The minimum reactor coolant flow rate is specified in the proposed Paragraph 3.1.1(c) of the Palisades Technical Specifications for four reactor coolant pumps in operation. Table 2.3.1 of the Technical Specifications indicates the required flow rates for two pump and three pump operation. The method by which this flow rate is measured, and the error analysis associated with this measurement, was sent to the NRC in a letter dated March 21, 1975.

Question 5.7

Provide the basis for arriving at the limiting axial power distribution as given in the transient and accident analysis. Provide assurance that this axial power distribution limit will not be exceeded during plant operation.

Response

The location of the axial peak was determined by previous analysis to be conservative for LOCA evaluations.[1] In addition, a top-peaked axial distribution is conservative from an MDNBR standpoint because it combines high coolant enthalpy with the maximum heat flux at the same location. The magnitude of the peak was conservatively chosen on the basis of previous operating experience. The largest steady state axial peak observed during Cycle 1 was 1.45, so a value of 1.5 was chosen for the Cycle 2 analysis. Reasonable assurance that the axial power distribution will not exceed specified limits is provided in that: A peaking factor higher than 1.5 is not predicted by physics calculations and is unlikely to occur, and the core axial as well as radial power distributions are monitored continuously by core instrumentation.

(1) Final Acceptance Criteria Emergency Core Cooling System Analysis for the Palisades Plant, prepared by Combustion Engineering, Inc, submitted on October 21, 1974. See Section II.C.

Question 5.8

Section 3.10 of the Technical Specifications would, under some circumstances, allow continued indefinite operation with one dropped or misaligned rod. The uncertainties involved in many of the analyses would not apply to such a highly perturbed power distribution. Either provide a detailed analysis of the effect of a dropped rod on power distribution uncertainties, DNBR, and core performance or limit operation with one dropped rod to 7 days per occurrence not to exceed a total of 14 days annually.

Response

Because of the asymmetric power distribution produced, some penalty in core performance is experienced when operating with a dropped control rod. Either margins to core limits are reduced or, if margins are inadequate, a derate must be applied to remain within thermal limits. Investigation of the effects of a dropped control rod reveal that the major perturbation involved is in the immediate vicinity of the rod itself, while in the areas of the core removed from the rod, the power distribution, except for an overall increase due to tilt, is relatively unchanged. Therefore, the major part of the increase in power distribution uncertainty will be in an area of the core where the power is depressed, while the uncertainties in the limiting regions of the core (away from the dropped rod) are relatively unaffected. Since the Technical Specifications (3.10.4) require that both the power distribution and the shutdown margin are verified (or prescribe appropriate power reduction), we conclude that the imposition of additional uncertainty factors would not add significantly to the margin of safety and that the establishment of arbitrary time limits is both unnecessary and undesirable.

Question 5.9

The peaking factors in Specification 3.10.3.a(1) are not consistent with the LOCA analysis. Provide corrected numbers.

Response

New Section 3.18 of the Technical Specifications is designed to provide limits consistent with LOCA analysis. Section 3.10 addresses thermal margins. Comparable peaking factors to Section 3.10.3.a(1) for Cycle 2 might be $F^n_{\Delta h} = 1.97$ and $F^n_q = 2.96$ as used in XN-76-3(P). It should be noted that the basis of the Cycle 2 numbers is 122% overpower margin whereas Table 3-13 of the FSAR implies that the 3.10.3.a(1) numbers were based on 112%. As discussed in December 18, 1972 submittal for Technical Specifications Change No 3 and January 3, 1973 submittal for Technical Specifications Change No 5, the reduction in pressure is at least partially mitigated by a corresponding reduction in core inlet temperature. Inspection of the peaking factors indicates that the primary difference for Core II is a reduction in the axial factor. Inspection of FSAR Figure 3-14, for instance, would indicate that reduction of the allowable rod insertion almost completely compensates for the reduction in $F^n_q$.

As previously discussed, Section 3.10 was not changed because of the existence of the new, more limiting Section 3.18 and our intentions to restore normal operating pressure at sometime in Cycle 2 after appropriate analysis has been completed.

Question 5.10

The quadrant to core average power tilts allowed by Specification 3.10.3 are much larger than those used by other plants (typically 2 to 5%). Either limit indefinite operation to less than 5% tilt or provide justification for the larger values, including an evaluation of the effect of tilt on the uncertainty in the measurement of total peaking factor.

Response

Section 3.10.3 of the Technical Specifications limits power operation, if tilts in excess of 10% exist for more than 24 hours. In addition, if tilt exceeds 15%, immediate verification of peaking factors or reduction to 75% power is required. If tilt exceeds 20%, immediate reduction to 50% power is required. These actions are quite severe and more than compensate for the degree of tilt observed. Proposed Technical Specifications Section 3.11 requires that in-core alarms be set to insure the power distribution is maintained within the design envelope. Any short-term tilt that threatens to cause operation above allowable local power limits will cause in-core alarms and bring the remedies of 3.11.b into play. The specifications of Section 3.11 apply to any tilt (including those less than 10%) that would cause allowable kW/ft to be violated. In addition, for long-term tilts, Sections 3.11 and 3.18 essentially require any observed tilt beyond instrument sensitivities, no matter how observed or how large, to be factored into the power distribution evaluation and in-core alarm limit calculations. Section 3.11.b also provides an additional margin of 15% when only in-core monitoring is available. In light of these monitoring requirements, we conclude that further Technical Specifications requirements in Section 3.10.3 are not appropriate. We believe the other plants referred to in your question (with 2 to 5% limits) are allowed to operate to full limits with only ex-core monitoring.

Question 5.11

Provide justification for the minimum number of operable incore detectors given in Specification 3.11.a. It will be necessary to demonstrate that the radial, axial, and azimuthal components of the power distribution can be measured with sufficient accuracy to support the measurement-calculational uncertainty given in Specification 3.18.1 with any permissible operable detector configuration. The Specification as written has no radial constraint and would allow operation with more than three quarters of the detectors out of service.

Response

As initially designed, it was not intended that the in-core system be required for operation of the reactor. The core has an average power density of 69 kilowatts per liter or approximately 4.7 kilowatts per foot. The average power density is approximately 30% lower than typical designs recently licensed. It is, therefore, possible to accommodate a considerable amount of radial, local and axial peaking without violating fuel design limits. Since it is generally easier to detect larger peaks, it is understandable that fewer detectors might be required in the conservative Palisades design than in the current designs which require far flatter power distributions to meet fuel limits.

Due to the inability to predict the failure pattern of in-cores, to anticipate "worst case" configurations, or to assign specific importance functions to each in-core, it would be very difficult to assign a unique uncertainty factor directly to various numbers of in-cores. This is particularly true since various degrees of analytical support can be used to supplement any given number of available in-cores.

The following assumptions seem to be inherent in the reasoning used to arrive at the Section 3.11.a minimum operability requirements. It is assumed that in-core chambers will fail in a reasonably random fashion. Because of the distribution requirement, it follows that when Section 3.11.a is just met there will be considerably more than 40 detectors available; ie, the probability of failing such that exactly the minimum configuration exists is essentially zero. It is also assumed that as the numbers of in-cores approach Technical Specifications limits, extraordinary means will be taken to track the power

distribution including, if necessary, daily core follow calculations. Since an operating cycle would not be started with near the minimum number of in-cores, the follow calculations could be compared with and "normalized to" previous operating history with that specific core. Such a normalized scheme of calculations would be compared "on the spot" with at least 40 in-core readings from various points in the core. If good agreement is obtained at 40 points, in our opinion, there would be good reason to believe (to a high confidence level) that the power distribution was known to within the 10% measurement/calculational allowance. The procedure is also attractive in that an ongoing estimate of the error would be available. A corollary assumption is that in-cores fail relatively randomly in time as well as space; ie, they do not all fail at once.

In-core failures observed during Cycle 1 were consistent with the above assumptions. During a portion of Cycle 1 operations, fewer than 75% of the in-cores were available. This posed no particular difficulty using "standard" methods.

It should also be pointed out that Section 3.18 specifications stand regardless of the status of the in-cores. In light of this fact, we do not believe that additional requirements are necessary. This seems especially true since the consequences of errors in the assumed power distribution are relatively minor due to the low average power density of the Palisades core.

Question 5.12

If part-length rods are to be used in the operation of this reactor, (a) the effect of the part-length rods on DNB must be evaluated, and (b) the use of a steady-state power distribution as a basis for incore alarms must be justified.

Response

It is not presently planned to use the part-length rods routinely during operation of the reactor. However, in order to maintain adequate operational flexibility, their availability for use is required. Possible uses during Cycle 2 include those involved with both testing and control of xenon oscillations (if they should occur).

We conclude that the part-length rods can be used without violating the design peaking factors for either DNB or LOCA analysis. If the part-length rods were used during high power operation, their use would be accompanied with either increased power distribution monitoring or additional analysis or both to verify that design peaking factors are maintained. Existing or proposed Technical Specifications in Section 3.11 require that power distribution be maintained within the design envelope and that sufficient monitoring be performed to assure that the power distribution remains acceptable.

The method used to set in-core alarms does not necessarily assume the power distribution remains unchanging. The purpose for in-core alarms would seem to render such an assumption invalid a priori. All of the in-cores at a given axial level in the core are set to correspond to the maximum power level existing at any point in that level. Consequently, if the peak power at an axial level were 90% of the limit, all alarms at that axial level would be set 10% above their then current reading. Operation during Cycle 1 has shown that the in-core alarms respond very well to power distribution changes caused by rod motion. When power distribution limits were approached as a result of rod motion, great numbers of in-core alarms were received. Shadowing of a few in-cores due to the presence of control rods does not debilitate the alarm function. Use of the steady state power distribution is more a method of dynamic calibration of the in-core signal to power factors than it is an assumption inherent in the monitoring function.

Question 5.13

There is no quantitative justification for the 85% level given in Specification 3.11.b. Either provide justification that this reduction will provide sufficient margin between actual and limiting peak linear heat generation rates for all power distributions permissible under the axial imbalance and quadrant tilt limits, or replace the 85% power level with "hot standby."

Response

Numbers like the 85% level of proposed Section 3.11.b (also essentially as presently existing in Appendix B) to some extent represent judgments of the system and the problem "taken as a whole." Because of this, discussions of their justification can become wide ranging and in certain aspects qualitative in nature. In order not to reiterate all the various aspects that represent past judgments and hours of previous deliberations, we present the following:

At present, the most limiting aspect of the Core II operating envelope appears to be LOCA and associated acceptance criteria. Core II LOCA analysis was performed assuming a total core peaking factor of approximately 2.89. At 100% power, when appropriate factors (engineering, densification, etc) are applied, this corresponds to 14.19 kW/ft. Since it is known to be conservative to trade core power level for increased peaking, reducing power to 85% of limits allows a 17.6% increase in core peaking to be accommodated. At this point, allowable total peaking factor would be 3.40. Taking credit for the 10% measurement/calculational uncertainty factor that is required by present Technical Specifications (but which was not required by the original Technical Specifications and FSAR) provides a factor 3.73 to be compared with a factor of 3.62 used in Technical Specifications Section 3.10.3.a. As stated in the Section 3.10 bases, 3.62 is consistent with the limits of 3.10.3.b, c and d as well.

Since the original margins are available, we conclude that the 85% level is justified.

Question 5.14

The 85% and 75% levels given in Specifications 3.11.e and 3.11.f must be justified in the manner described in the previous question. The minimum number of detectors logged must be justified in a manner consistent with the justification requested for Specification 3.11.a. Alternatively, 3.11.e and 3.11.f could be replaced with a specification requiring tilt and imbalance to be maintained constant during data logger repair.

Response

Section 3.11.e is designed to be consistent with Section 3.11.b. Therefore, justification for the 85% is the same as discussed under Question 5.13 for 3.11.b. It is assumed that "instantaneous" alarms are not available but that manual reading of in-cores (a minimum number every two hours with the ability to read them all periodically, eg, once a week) provides for long-term follow of core power distribution. This, when taken along with other available indications such as rod positions and ex-core detector readings, provides additional assurance that allowable local power levels are not exceeded.

The 75% level used in 3.11.f is justified in a manner similar to that given for the 85% in 3.11.b as discussed in the answer to Question 5.13. Since the ex-cores provide full protection when maintained within their Technical Specification bounds, the in-cores are not required. It should be noted that Section 3.18 requirements are still binding and that core follow calculations, rod positions monitoring and ex-core monitoring would still be performed. Results of these efforts would have to indicate that limits were being met or additional power reductions would be required.

The minimum number of in-cores that are to be read "manually" as required by Section 3.11.f is, of course, consistent with the number required in Section 3.11.a for obvious reasons. The justification also includes similar reasoning. However, in this section, consideration is also given to the number that can physically be read while maintaining reasonable accuracy and reliability in the readings. One in-core must be ready every three minutes. The reader must verify the correct terminals, attach leads, read a low level signal, and record

the reading in an appropriate blank within this time interval. Due to the small size of the terminal panel, more than one reader is impractical. Causing undue hurry in taking readings would also seem counterproductive. Because of the tedious nature of the work, the manpower required, and the economic penalty of the 15% derate, it is unlikely that the plant would be operated very long in this mode. It is also anticipated that if more in-cores were available a slightly different set might be read each two-hour period so that over a longer interval all available in-cores could be read.

Question 5.15

References to ECCS and LOCA analyses should be updated to reflect the most current information. Revise these references, eg, reference #3 in Item b of Section 3.16, and FSAR Figures 14.17.9 to 14.17.13 on Page 3-33, as appropriate.

Response

We agree that the FSAR and Technical Specifications need to be updated as suggested. We conclude, however, that this updating would best be done following approval of this specification change. This procedure would permit inclusion of appropriate parts of all the materials submitted and reviewed as a part of this Technical Specifications Change.

We anticipate the need for extensive revision of the FSAR and will begin this process when your review of our various submittals has been completed.

Question 5.16

The phrase, "Unless otherwise justified..." beginning the second paragraph of Specification 3.18.1 is ambiguous. Either provide clarification or delete this phrase.

Response

A proposed change deleting this phrase, "Unless otherwise justified ..." is contained in another attachment (see Item D, Paragraph 2 of Attachment 2). We have concluded that, at the present time, the only mutually acceptable way to justify changes in the various uncertainty factors is through a formal Technical Specifications Change.

Question 5.17

Provide text in Specification 3.18.1 giving the actual peak linear heat generation rate value from the LOCA analysis (when available).

Response

During initial development of the Technical Specifications changes relating to the ECCS analysis and Reload 2 core, we had concluded that it would be desirable to develop the specification such that they relate to basic considerations such as the requirements of Appendix K. Discussion with members of your staff have lead us to believe that their approach is not considered acceptable at the present time. We consider this apparent decision to be unfortunate and to place a significant and unnecessary financial burden upon Consumers Power Company and its customers without any real safety benefit. Unfortunately, arguing this point further at this time, at the expense of plant operating time, would likely be more expensive and we have, therefore, proposed a Technical Specifications Change in accordance with your request (see Attachment 2, Item D, Paragraph 1).

Question 5.18

According to XN-207 "Power Spike Model for Pressurized Water Reactor Fuel" an augmentation factor is to be calculated according to either an assumed densification limit of 96.5% theoretical density or a density limit determined from a resintering test. Specify which of these methods is used and provide a calculation of the power spike. The augmentation factor of unity as specified on Page 2, Item J, of the January 30, 1976 letter is not acceptable.

Response

A densification augmentation factor due to axial stack shortening has been calculated for Batches D, E and F fuel. The calculation was based on densification to the 96.5% limit from as-manufactured density for ENC fuel. The increase in LHGR due to stack shortening for the ENC fuel was approximately 1.3% for Batches E and F. Credit was taken for thermal expansion of dished pellets in accordance with the approved ENC model for net augmentation factor of 1.008. An augmentation factor of 1.0175 was uniformly applied to the more limiting Batch D fuel. The application of this factor to the DNB analysis is described on Pages 92 and 97 of XN-76-3. This augmentation factor was applied to the DNB analysis and, therefore, is included in calculation of margin to DNB. The factor is not included in the ECCS analysis documented in XN-76-4 and Supplements and, therefore, needs to be separately applied to the LOCA thermal limit.

The effect of power spikes due to uranium dioxide pellet densification with consequent pellet gapping is considered both in the ECCS and DNB analysis. The spike augmentation factor applied in both instances was unity. Such treatment has been previously generically justified and is discussed in XN-75-43, Page 33, Section 5.4 as part of the ENC ECCS model.

The effect on DNB of power spikes up to 20% in PWRs is documented in WCAP-8219. The authors conclude that power spikes of this magnitude located at the point of MDNBR do not reduce the DNB heat flux as calculated by the W-3 correlation below that of rods without the power spike.

The magnitude of power spikes induced by urania densification was calculated in accordance with XN-207 for Batches D and E fuels. Batch F fuel is clearly not limiting in Cycle 2 and, therefore, the power spike magnitude was not calculated for F fuel.

The power spikes calculated for D and E fuels are in the neighborhood of 2.5%-3.5% in the axial region of MDNBR. These spikes are much less than the 20% for which no penalty was found and, therefore, imposition of a spike penalty on DNB calculations is inappropriate.

It is concluded that an augmentation factor due to fuel pellet column shortening because of in-pile densification of urania to 96.5% of theoretical density must be applied to the calculated nuclear peaking factors for comparison to ECCS performance limits. This augmentation factor (1.75%) is included in the proposed Technical Specification 3.18.1. It is further concluded that an additional augmentation factor for "power spike" is not required.

Question 5.19

Unless LOCA analyses for three and two loop operation are forthcoming, provide an additional specification restricting the plant to four pump operation or provide further justification for the specifications presently proposed.

Response

A new paragraph has been added to Section 3.1.1(c) (see Attachment 2, Item A) to clarify our intent concerning less than four loop operation. This restriction will remain in effect until our analysis for less than four loop operation has been completed and submitted for review.

A change to Table 2.3.1 has also been proposed to clarify operation with less than four pumps (see Attachment 2, Item E). Limited operation with less than four pumps will permit time to effect repairs without requiring reactor shutdown.

Question 5.20

In the review of the ECCS, certain valves have been identified requiring power lockout to satisfy the single failure criteria, ie, 4 safety injection tank isolation valves, 2 mini-flow bypass valves, 1 shutdown cooling flow control valve and possibly a hot leg injection valve. For isolation purposes, power must be restored to the mini-flow bypass line valves to allow valve closure prior to the recirculation mode. Similarly, if it is necessary to remove power from a valve in the hot leg injection system, provision must be made for restoring the power prior to initiating hot leg injection. Include in your Technical Specifications appropriate requirements for power removal and restoration.

Response

Appropriate Technical Specifications have been prepared for the four safety injection tank isolation valves and the shutdown cooling flow control valve (see Attachment 2, Items F and G).

With respect to the hot leg injection valves, the equipment and/or valves in the two long-term cooling alignments (hot leg suction and hot leg injection) <u>have</u> <u>not</u> been altered from original failure modes.

During the qualification program for long-term cooling equipment (boron concentration problem), the possibility of reversing the operation of certain valves was considered. This was rejected because of the implications on normal operation. All valves were/or will be qualified such that their failure mode is unaltered from the original design criteria.

The operation of the two miniflow bypass valves is still under review. Appropriate Technical Specifications changes will be submitted with the results of the review of these valves.

Question 5.21

Item B on Page 1 of the January 30, 1976 letter states that core plugs may be used to replace one or more fuel assemblies. Provide clarification of the design and purpose of these plugs.

Response

Core plugs are described in Section 3.3.4.3 (Page 3-65) of the FSAR. We conclude that this description will provide the information requested.

ATTACHMENT 2

Proposed Technical Specifications Changes

Relating to the ECCS Analysis and

the Reloading (Cycle 2) of the Palisades Reactor

April 1976

Other proposed changes relating to this subject were presented by letters dated July 9, 1975 and January 30, 1976.

CONSUMERS POWER COMPANY

Docket 50-255

Request for Change to the Technical Specifications

License DPR-20

For the reasons hereinafter set forth, it is requested that the Technical Specifications contained in Provisional Operating License DPR-20, Docket 50-255, issued to Consumers Power Company on October 16, 1972, be changed as follows:

I. Changes

A. Change Section 3.1.1(c), as proposed in our January 30, 1976 submittal, to read as follows:

"c. The minimum flow for various power levels shall be as shown in Table 2.3.1. The measured 'Four Primary Coolant Pumps Operating' reactor coolant vessel flow (as determined by reactor coolant pumps differential pressure and pump performance curves) shall be $124.0 \times 10^6$ lb/h or greater, when corrected to 532°F.

In the event the measured flow is less than that required above, the limits specified in 3.18.1 shall be reduced by 2% for each 1% of reactor coolant flow deficiency.

Continuous operation at power shall be limited to four pump operation. Following loss of a pump, thermal power shall be reduced as specified in Table 2.3.1 and appropriate corrective action implemented. With one or more pumps out of service, return the pumps to service (return to four pump operation) within 24 hours or be in hot standby (or below) within an additional 12 hours. Start-up (above hot standby) with less than four pumps is not permitted."

B. Change the basis to Section 3.1.1 as proposed in our January 30, 1976 submittal as follows (this adds a new paragraph to the basis of Section 3.1):

"The ECCS analysis has been conducted at a vessel flow of $124.0 \times 10^6$ lb/h, and the primary system flow areas and loss coefficients used in the analysis were forced to agree with this flow. The DNB analysis (assuming 122% margin to overpower) has also been performed at this flow with a 3% uncertainty. The ECCS limits associated with this flow rate, which may be more restrictive than the DNB limits, are specified in Section 3.18.1.

In the event the measured flow is less than the required flow, a decrease in allowable thermal limits is required. This decrease in thermal limits, at twice the percentage by which flow is decreased, conservatively maintains the power to flow ratio and provides adequate margin for transients and accidents."

C. Add new part to Section 4 as follows:

"4.15 Primary System Flow Measurement

<u>Applicability</u>

Applies to the measurement of primary system flow rate with four primary coolant pumps in operation.

<u>Objective</u>

To provide assurance that the primary system flow rate is equal to or above the flow rate required in 3.1.1(c).

<u>Specification</u>

After each refueling outage, or after plugging 10 or more steam generator tubes, a primary system flow measurement shall be made with four primary coolant pumps in operation before the reactor is made critical.

<u>Basis</u>

This surveillance program assures that the reactor coolant flow is consistent with that assumed as the basis for Specification 3.1.1(c)."

D. Change proposed Section 3.18.1 (as proposed in our January 30, 1976 letter) as follows:

"3.18.1 The linear heat generation rate with appropriate consideration of normal flux peaking, measurement-calculational uncertainty, engineering factor, increase in linear heat rate due to axial fuel densification, power measurement uncertainty, and flux peaking augmentation shall not exceed 14.19 kW/ft.

The measurement-calculational uncertainty shall be 10%, the engineering factor shall be 3%, the increase in linear heat rate due to axial densification shall be 1.75% (as applied to hot dimensions), the power measurement uncertainty shall be 2%, and the flux peaking augmentation factor shall be as given in Figure 3-6 for uncollapsed fuel and Figure 3-7 for collapsed fuel. Augmentation factors for pressurized densification resistant ENC fuel and pressurized high density CE fuel shall be 1.0."

E. Change Table 2.3.1, Item 1, for two and three primary coolant pumps operating as follows (this revises our proposed change of January 30, 1976):

| Three Primary Coolant Pumps Operating | Two Primary Coolant Pumps Operating |
|---|---|
| "45% of rated power(4) (Continuous operation not permitted.) | 25% of rated power(4) (Continuous operation not permitted.) |

(4) Operation with two or three pumps is permitted to provide a limited time for repair/restart; to provide for an orderly shutdown, or to provide for the conduct of reactor internals, noise monitoring test measurements (a maximum of 12 hours of operation each time this test is conducted)."

F. Change Section 3.3.1 by adding the following parts:

"h. The Low Pressure Safety Injection Flow Control Valve shall be opened and disabled to prevent spurious closure.

i. The Safety Injection bottle motor-operated isolation valves shall be opened with the electric power supply defeated."

G. Change the Basis of Section 3.3 by adding the following:

"Malfunction of the Low Pressure Safety Injection Flow control valve could defeat the Low Pressure Injection feature of the ECCS; therefore, it is disabled in the 'open' mode (by isolating the air supply) during plant operation. This action assures that it will not block flow during Safety Injection.

The inadvertent closing of any one of the Safety Injection bottle isolation valves in conjunction with a LOCA has not been unanalyzed. To provide assurance that this will not occur, these valves are electrically locked open by a key switch in the control room. In addition, prior to critical the valves are checked open, and then the 480 volt breakers at MCC 9 are opened. Thus, a failure of a breaker and a switch are required for any of the valves to close."

II. Discussion

The Technical Specifications Changes proposed in this attachment have been prepared as a result of Questions and Responses of Attachment 1. As a reference to a discussion of each of the changes, the following table is provided:

| Technical Specifications Item Number | Discussion (Reference to Question in Attachment 1) |
|---|---|
| A | 5.4 & 5.19 |
| B | 5.4 |
| C | 5.5 |
| D | 5.16 & 5.17 |
| E | 5.19 |
| F | 5.20 |
| G | 5.20 |

III. <u>Conclusion</u>

Time restraints have not permitted a complete review by either the Palisades Plant Review Committee or the Safety and Audit Review Board. This review will be conducted and we will advise you should any of these proposed changes be deemed inappropriate.

CONSUMERS POWER COMPANY

By ______________________

C. R. Bilby, Vice President

Sworn and subscribed to before me this 5th day of April 1976.

______________________

Sylvia B. Ball, Notary Public

Jackson County, Michigan

My commission expires May 18, 1976.