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HAYES, J. G. Arizona Nuclear Power Project (formerly Arizona Public Serv
RECIP. NAME RECIPIENT AFFILIATION
Document Control Branch (Document Control Desk)

SUBJECT: Forwards response to NRC 870730 request for info re util 870629 Cycle 2 reload submittal. C-E ltr responding to NRC request to reevaluate validity of propiretary info in Rev 01-P to CEN-356(V)-P encl.

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TITLE: Startup Report/Refueling Report (per Tech Specs)

NOTES: Standardized plant. M. Davis, NRR: 1Cy. 05000528

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Arizona Nuclear Power Project
P.O. BOX 52034 • PHOENIX, ARIZONA 85072-2034

August 20, 1987
161-00455-JGH/LJM

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

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Unit 1
Docket No. STN 50-528 (License NPF-41)
Response to NRC Questions Regarding the
Unit 1 Cycle 2 Reload Submittal
File: 87-B-056-026

Reference: Letter from J. G. Haynes (ANPP) to Document Control Desk (NRC) dated June 29, 1987 (161-00321). Subject: Submittal of the Reload Analysis Report for Unit 1 Cycle 2.

Dear Sir:

On July 30, 1987, a meeting was held between NRC Staff and ANPP to discuss the Unit 1 Cycle 2 reload licensing submittal. During the meeting, several items were identified which required formal responses.

Attachment 1 is the response to the written questions the Staff provided ANPP prior to the meeting.

Attachment 2 is a copy of the letter, supplied by Combustion Engineering to ANPP, which responds to the Staff's request to reevaluate the validity of the proprietary information in the Modified Statistical Combination of Uncertainties Topical, CEN-356(V)-P, Revision 01-P.

Attachment 3 is a copy of the letter reference made in the Reload Analysis Report (see referenced letter) regarding Fuel Handling Interference.

Attachment 4 contains legible copies of the two pages of the Technical Specification changes which the Staff had difficulty reading.

Attachment 5 contains additional justification for the change to the ASI Technical Specification Amendment, which the staff requested during the July 30, 1987, meeting.

Attachment 6 contains a revised page of the Reload Analysis Report. The revision addresses the questions the staff had concerning the crediting of a new Variable Overpower Trip (VOPT) in the Steam Line Break event.

IE26
1/1

If you have any questions, please contact Mr. W. F. Quinn at (602) 371-4087.

Very truly yours,

J. G. Haynes

J. G. Haynes
Vice President
Nuclear Production

JGH/LJM/rw
Attachment

cc: O. M. De Michele
E. E. Van Brunt, Jr.
G. W. Knighton
J. B. Martin
E. A. Licitra (w/a)
A. C. Gehr
J. R. Ball (w/a)

ATTACHMENT 1

NRC QUESTIONS ON PALO VERDE

UNIT 1, CYCLE 2 RELOAD AND ASSOCIATED TECH SPEC CHANGES

1.a. Q. The justification for the proposed change to delete the Tech Spec allowance for degraded resistance temperature detector (RTD) response times beyond 8 seconds mentions that if a RTD response time is greater than 8 seconds, the associated CPC channel must be declared inoperable until repairs are completed. Where is this requirement stated in the Tech Spec?

A. Technical Specification 3/4.3, Table 3.3-2, Note ## states, "... The measured response time of the slowest RTD shall be less than or equal to 8 seconds." If the RTD response time is greater than 8 seconds, the above condition is obviously not met. Failure to meet the response time test implies the equipment is not functioning properly and therefore must be declared inoperable.

b. Q. Since the RTD response times need only be measured once per 18 months, what will be done about RTD's for which previous measurements indicate possible degradation beyond the 8 second response time before the next measurement?

A. An explicit allowance is not provided for RTD response time drift. If the response time measurements indicate a drift is occurring which might exceed the 8 seconds, appropriate action would be considered at that time. The RTD response times measured to date at Palo Verde do not indicate a trend that the 8 second response will be exceeded.

2. Q. Explain why the revision to Action Statement 6.b.1 in Technical Specification Table 3.3-1 does not require the LHR margin to be maintained also.

A. The current requirement in Action Statement 6.b.1 in Technical Specification Table 3.3-1 to maintain Technical Specification 3.2.1 is redundant. Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the Local Power Density Channels in the Core Protection Calculators (CPC's) with either CEAC's operable or inoperable, provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The linear heat rate LCO is based on LOCA analyses which are unaffected by CPC/CEAC operability. Therefore, no change to the LHR Technical Specification is needed for Action Statement 6.b.1.

3\. Q. In view of the increase in critical boron concentration and boron worth for Cycle 2, why was the inadvertent boron dilution event not reanalyzed?

A. The inadvertent boron dilution event was reanalyzed as part of the temperature dependent shutdown margin technical specification change package (ANPP Letter No. 161-00377, dated July 17, 1987). The conservative boron concentrations used in the analysis bound both Cycle 1 and Cycle 2 of Unit 1.

4.a. Q. Since the allowable MTC limit has been increased to $+0.5 \times 10^{-4}$, why weren't those transients and accidents which are adversely affected by a positive MTC (e.g. CEA withdrawal, CEA ejection, LOCA) reanalyzed for Cycle 2?

A. The Reload Analysis Report (RAR) gives MTC values for Cycle 2 in Figure 7.0-2 of $+0.5 \times 10^{-4}$ delta-rho/deg. F only at zero core power at BOC. At 100% core power, the MTC has a value of 0.0 which is the same value that the Cycle 1 MTC had at 100% power and BOC (see Figure 7.0-1). The Cycle 1 MTC value at zero power was $+0.22 \times 10^{-4}$ delta-rho/deg. F at BOC.

The three transients mentioned in the request (CEA withdrawal, CEA ejection and LOCA) will thus be unaffected by the MTC when initiated at 100% core power conditions. The CEA withdrawal transient at low power for Cycle 2 is bounded by the CESSAR analysis. The CESSAR analysis used a MTC of $+0.5 \times 10^{-4}$ delta-rho/deg. F.

Note that the RAR only lists events which are not bounded by the reference cycle or events for which analysis methodology differed from the reference cycle. The CEA ejection and withdrawal events are most limiting at full power including consideration of a positive MTC at low powers.

4.b. Q. How does the use of an MTC as a function of power level affect the results of these transients and accidents as compared to an MTC as a function of temperature as was used in Cycle 1.

A. There is no affect on the results of the analysis as the MTC is only being more clearly represented as a function of power level instead of temperature.

c. Q. As stated in the letter from A. Thadani (NRC) to J. K. Gasper (CE Owners Group) dated June 12, 1987, the staff is concerned about trends in current reload designs that may lead to initial MTC values that are less conservative than those that were assumed in the ATWS analyses. Based on the more positive MTC anticipated for Cycle 2, please comment on the justification for the continued applicability of your ATWS analyses.

A. In the above referenced letter, the NRC requested additional information from the CE Owners Group concerning the impact of more positive MTC's on the previously performed ATWS analyses. These analyses were performed on a generic basis in the 1970's by CE for the Owners Group. At the time that the ATWS issue was originally addressed, both the NRC and the individual plant owners considered the Owners Groups an acceptable vehicle by which to respond to the staff's concerns.

ANPP proposes to address the staff's concerns on the validity of the original ATWS analyses generically via the CE Owners Group instead of separately through individual plant reload analyses. This will allow the Palo Verde Unit 1, Cycle 2 reload to proceed in an orderly and timely manner. Also, utilizing this proposed approach will provide sufficient time to fully consider the implications of the impact of more positive MTC's on the ATWS analyses.

5.a. Q. How was acceptable LOCA ECCS performance for both small and large breaks confirmed for up to 400 plugged steam generator tubes?

A. A large number of plugged steam generator (SG) tubes could potentially affect the ability of the Emergency Core Cooling System to limit the consequences of a Loss of Coolant Accident (LOCA). Tube plugging increases the RCS flow resistance, and decreases the SG heat transfer area. The increased flow resistance primarily affects the large break LOCA causing a reduction in the core reflood rate which increases the peak clad temperature for the event. The reduction in the SG heat transfer area primarily affects the small break LOCA ultimately causing greater core uncovery which increases the peak clad temperature.

The large break LOCA is the limiting case for demonstrating conformance to the 10CFR50.46 criteria. As indicated above, the major effect of steam generator tube plugging on large break LOCA is the increased

resistance to flow. The current LOCA analyses assume a conservatively large pressure drop through the steam generators. The assumed value is more than adequate to offset the small increase due to tube plugging. The best estimate SG pressure drop with 400 plugged tubes/SG is 39.5 psi vs. 42.0 psi assumed in LOCA analyses. Therefore, the current large break LOCA analyses cover up to 400 plugged tubes/SG.

Estimates of the increase in peak clad temperature (PCT) are much less than 100 °F for a small break LOCA assuming 400 plugged tubes/SG. Therefore, the estimated PCT for PVNGS is less than 1730 degrees which demonstrates that the small break LOCA PCT is well within the 10CFR50.46 limit and that the large break LOCA PCT remains limiting.

b. Q. Was asymmetric plugging modeled?

A. Asymmetric steam generator tube plugging was considered and it was concluded that the symmetric tube plugging analysis implicitly covers unequal plugging between generators, provided no more than 400 tubes are plugged in either steam generator. The basis for this conclusion is as follows:

Tube plugging affects primarily the refill/reflood portion of the large break LOCA transient. Steam generator tube plugging will have no significant effect on the blowdown portion of the transient. The main impact of steam generator tube plugging is to increase the resistance to flow passing through the primary side of the steam generator, thereby inhibiting steam venting from the core outlet plenum to the break. This reduces the refill/reflood rates and increases the peak cladding temperature. With regard to this effect, plugging fewer than 400 tubes in either or both steam generators will reduce the flow resistance and will improve the refill/reflood rates. Consequently, a reduction in the number of plugged tubes in either steam generator will reduce the peak cladding temperature relative to the analysis results with 400 plugged tubes in each generator.

c. Q. Since the loop resistance is affected by tube plugging, justify why new hydraulics calculations were not performed for the Cycle 2 LOCA analysis.

A. As discussed in (a) and (b), it was not necessary to perform new hydraulic calculations for the Cycle 2 LOCA analyses. The assumed pressure drop through the steam generators used in the analyses has enough conservatism to provide for 400 plugged tubes per steam generator.

6. Q. If 400 plugged steam generator tubes are accommodated in Cycle 2, how do the CPC algorithms account for tube plugging in the determination of core flow?

A. The core flow used by CPC in its DNBR calculation is calibrated every shift to a pump differential pressure or calorimetric flow measurement per Technical Specification Table 4.3-1, note 7. This calibration accounts for any change in flow resulting from plugged steam generator tubes.

7. Q. Please present the calculated k_{eff} and 95/95 probability/confidence uncertainties for the Palo Verde new and spent fuel racks containing 4.05 weight percent U-235 fuel enrichment.

A. An enrichment of 4.30 weight percent U-235 was assumed in the original criticality analyses of the Palo Verde new and spent fuel racks. A description of the analyses and assumptions utilized as well as the results of the analyses are presented in Section 9.1 of the Palo Verde FSAR. The largest k_{eff} that was calculated over the range of conditions which were analyzed was 0.777 for the new fuel racks and 0.889 for the spent fuel racks. The associated 95/95 probability/confidence uncertainties for these calculations were not explicitly determined for Palo Verde due to the large margins that exist between the calculated k_{eff}'s and the Regulatory Guide requirements. It should be noted that using the same methodology at other CE plants has always yielded an uncertainty of less than 2.6%. Also, an additional conservatism exists for the k_{eff}'s reported above in Unit 1, Cycle 2 since the maximum enrichment will be 4.05 weight percent U-235.

ADDITIONAL NRC QUESTIONS

1. Q. A value of 8.5% was used for the radial peaking factor increase in the CEA drop analysis. How was this obtained and how is it verified to be the largest change obtainable for a CEA drop into a radial configuration allowed by the Tech Spec PDIL transient insertion limit?

A. The CEAC's will provide a bounding penalty in the CPC calculation of DNBR and LPD for drops of 12-finger CEA rods so that the CPC's will generate a reactor trip if necessary. The drops of 4-finger CEA rods are protected against by the initial margin maintained by the power operating limits and the Tech Spec requirements for power reduction following a drop. The drop of a 4-finger rod rather than a 12-finger rod is presented because this event more closely approaches the SAFDL presented in the RAR.

The 8.5% radial peaking factor increase is the maximum radial peaking factor increase immediately after the drop of <u>any</u> 4-finger rod at full power before the effect of short-term xenon redistribution. The maximum radial peaking factor increase including the effect of 15 minutes of xenon redistribution is 11.4% as given in Table 7.4.3-1. This is the value used in the CEA drop analysis.

The radial peaking factor increases were calculated with the ROCS code. Calculations were performed at BOC, MOC, and EOC equilibrium xenon conditions with ARO and with full and part length rods at the PDIL.

In addition, limiting cases were examined for the effect of non-equilibrium xenon initial conditions by simulating a scram from hot full power equilibrium xenon conditions followed by a return to power starting after 6 hours and completing after 8 to 10 hours. Experience has shown that these initial conditions are limiting for the CEA drop events.

2. Q. The proposed change to Tech Spec 3.1.2.3 would permit more than one charging pump to be operable during Mode 5. Please verify that the assumptions used in the postulated mass addition event analyzed for supporting the Low Temperature Overpressure Protection (LTOP) system design remain valid for this proposed change. Does the Bases for the Boration Systems have to be modified for this proposed change?

   A. The postulated mass addition event analyzed for supporting the LTOP system design assumed three operating charging pumps. (See Figure 5.2-7 of the PVNGS FSAR.) The Boration System Tech Spec Bases do require modification. The attached sheet shows the required deletion.

3. Q. What additional changes have to be made to the Cycle 2 reload analyses because of the revised Unit 1, Unit 2, and Unit 3 Cycle 1 shutdown margin curve?

   A. The main steamline break analysis determines the required shutdown margin at high temperatures. Therefore the shutdown margin curve will be modified to agree with the main steamline break reload analysis. The high break point will increase from 6% to 6.5% shutdown margin. No additional changes have to be made to the Cycle 2 reload analyses.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System cold leg temperature less than 552°F. This limitation is required to ensure (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, and (3) to ensure consistency with the FSAR safety analysis.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include (1) borated water sources, (2) charging pumps, (3) separate flow paths, and (4) an emergency power supply from OPERABLE diesel generators. The nominal capacity of each charging pump is 44 gpm at its discharge. Up to 16 gpm of this may be diverted to the volume control tank via the RCP control bleedoff. Instrument inaccuracies and pump performance uncertainties are limited to 2 gpm yielding the 26 gpm value.

With the RCS temperature above 210°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 4% delta k/k after xenon decay and cooldown to 210°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 23,800 gallons of 4000 ppm borated water from either the refueling water tank or the spent fuel pool.

With the RCS temperature below 210°F one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single injection system becomes inoperable. ~~The restrictions of one and only one operable charging pump whenever reactor coolant level is below the bottom of the pressurizer is based on the assumptions used in the analysis of the boron dilution event.~~

The boron capability required below 210°F is based upon providing a 4% delta k/k SHUTDOWN MARGIN after xenon decay and cooldown from 210°F to 120°F. This condition requires 9,700 gallons of 4000 ppm borated water from either the refueling water tank or the spent fuel pool.

ATTACHMENT 2

July 30, 1987
V-CE-35072

COMBUSTION ENGINEERING

Mr. Paul F. Crawley
Arizona Nuclear Power Project
11202 N. 24th Ave.
Phoenix, AZ 85029

Subject: Response to NRC Question on CEN-356(V)

Reference: "Modified Statistical Combination of Uncertainties," Combustion Engineering, Inc., CEN-356(V)-P, Revision 01-P, July 1987.

Dear Mr. Crawley,

C-E is pleased to provide you with the following clarification to an NRC question relative to the referenced report. Please forward this information to the NRC at your earliest convenience.

The Modified SCU Program involves the statistical combination of several uncertainty components previously applied deterministically including system parameter uncertainties, secondary calorimetric power measurement uncertainties, and neutron flux power synthesis uncertainties. In addition, minor changes were made including the treatment of the radial peaking factor uncertainty.

As described in Section 3.2, the fuel rod bow and HID-1 grid penalty factors are applied independent of the SCU Program in the deterministic manner shown in the equation on Page 18 and in the example on Page 19.

If we can be of further help, please contact Mr. J. F. Church in Windsor.

Sincerely,

JFChurch For

C. Ferguson
Project Manager

JFC/CF/kls
JFC139.TXT

REL-87-V-239

cc: J. F. Church
R. Land
L. Marker
E. E. Van Brunt, Jr.

ATTACHMENT 3

RECEIVED APR 1986 NUCLEAR FUEL MANAGEMENT

COMBUSTION ENGINEERING

April 10, 1986
V-CE-33635

Mr. P. F. Crawley
Manager-Nuclear Fuels
Arizona Nuclear Power Project
Mail Station 3003
P.O. Box 2166
Phoenix, Arizona 85036

Subject: Palo Verde Nuclear Generating Station
Fuel Handling Interference

References: (A) ANPP-922-PFC-98.20, dated December 24, 1985
(B) A-24765-D, Spent Fuel Handling Tool (Assembly)
(C) D-24766-D, Inner Grapple
(D) D-24767-D, Outer Grapple
(E) E-SYS80-630-120, New Fuel Handling Tool Assembly and Details
(F) D-25690-D, Grapple (Refueling Machine)

Attachment: (1) Review of Potential for Interference Between C-E Fuel Assembly and Palo Verde Fuel Handling Tools

Dear Mr. Crawley:

Based on the concerns of the potential interference between the grappling mechanism on a) the spent fuel and transport container handling tools, References (B, C, and D); b) the new fuel handling tool, Reference (E) and c) the refueling machine, Reference (F) with respect to the fuel assembly upper end fitting hold down plate which was identified in Reference (A), C-E has performed a detailed study of the problem. A summary of this study and its results are reported in Attachment (1).

C-E used two separate methods in evaluating this problem. The first was to perform a tool/hold down plate layout to establish whether the worst case tolerance conditions, summed linearly, actually did result in the potential for interference. Using this approach, which would effectively produce the worst possible case scenario, C-E arrived at the conclusion that only the spent fuel handling tool has the potential for interference with C-E supplied fuel.

The second method involved quantifying the degree of potential interference that could exist between the spent fuel handling tool and the hold down plate. C-E's approach in quantifying potential interferences is to consider the probability of interrelated tolerance conditions occurring concurrently. C-E's approach, which is an established and accepted engineering practice, differs from the linear summation approach which was apparently used by Westinghouse in their evaluation.

Based on C-E's approach to this problem, the maximum value of potential interference is reduced from .036" to .003" at the handling tool/fuel interface. While this potential interference cannot be ignored, C-E does not believe that the necessity to eliminate this potential condition has been established. It should be noted that the machining needed to effectively eliminate this potential condition also tends to reduce the locking capability of the tool with respect to the fuel.

In fact, C-E believes there is sufficient evidence to support the position that the tool should not be modified. This position is based on the fact that a) the interference problem represents no safety concern, and b) the possibility of a tolerance mismatch producing interference is remote.

C-E's bases its position on the following:

1) If both the hold down plate and the grapple on the SFHT (or TCHT) were at their highest respective tolerance dimensions, there is a realistic probability of interference occurring between the tool and the hold down plate. In this situation, the grapple would not be able to engage the hold down plate and the underload interlock on the spent fuel or refueling machine would cause the hoist motion to stop. This fuel assembly would be unable to be lifted.

2) If a minor interference condition such as point or line contact were to occur, the hoist would continue its travel until the grapple hook were below the hold down plate. The fuel assembly could be engaged and lifted without any problem. This is true since the grapple cannot be rotated to the lift position unless it is either completely above or below the hold down plate. There is no way to partially engage a fuel assembly. Therefore, the integrity of the fuel assembly cannot be jeopardized.

3) The C-E supplied hold down plates are made from castings rather than being machined. The investment casting process which is used to produce the hold down plate requires the use of a single mold for all castings. This process offers a high degree of dimensional uniformity. C-E has checked the applicable dimensions on several hold down plates and has found that they are within several mils of the nominal dimensions and believes this to be the case for all of the hold down plates.

4) Since the grapple on the tools was machined, C-E recognizes that there is the possiblity that each tool has the potential for a mismatch with the upper end fitting. This necessitated the tolerance study. However, it must be recognized that a minimum of 482 fuel assemblies have been

handled at the Arizona site without having revealed any interference type problems. Since both Units 1 and 2 loaded fuel successfully and no problems were reported that indicated the presence of any interference type problems during the fuel handling operations, C-E believes that there is sufficient statistical evidence to support the position that a mismatch between the spent fuel handling tool and the C-E supplied hold down plate is unlikely.

It should be pointed out that at the time the problem was first discovered, and prior to reviewing Westinghouse's study, the possibility of taking as-builts dimensions of the grapples was discussed. After reviewing the results of the Westinghouse study and after recognizing the problems associated with taking accurate as-builts of the critical dimensions in the field, it was decided that the only assured approach to obtaining accurate dimensions was to return the tools to the vendor. C-E feels the need for this action has been temporarily eliminated by the results of their tolerance study.

Please be advised that C-E's study compares the dimensions and tolerances for C-E supplied equipment. Therefore, C-E's conclusions may vary from those of Westinghouse.

Should you have any further concerns or questions on this issue, please contact this office.

Very truly yours,

C. Ferguson
C. Ferguson
Project Manager

CF/dy/PFM58
xc: G. A. Butterworth
S. L. Schey, w/attach.
J. W. Dilk
E. E. Van Brunt, Jr. - w/e
R. Fullmer
J. G. Haynes
J. R. Bynum
R. M. Butler
R. Fullmer - w/e

REVIEW OF POTENTIAL
FOR INTERFERENCE
BETWEEN C-E FUEL ASSEMBLY
AND PALO VERDE FUEL HANDLING TOOLS

1.0 Purpose

This review was performed in response to APS letter ANPP-922-PFC-98.20.

A Westinghouse study has concluded that a potential for interference exists between their fuel and the C-E supplied handling tools. APS requested C-E to determine if a similar potential interference exists between the C-E supplied fuel and the handling tools.

2.0 Scope

This review looked at the potential interference between the fuel assembly holddown plate and the following handling equipment: new fuel handling tool, spent fuel handling tool and the refueling machine. In all cases, the most adverse tolerances were used to maximize any potential for interference. Both the "engage" and "lift" conditions were investigated to ensure the tools will fit onto the fuel and that the grapples can be fully locked to prevent dropping of the fuel.

3.0 References

3.1 Drawing E-STD-161-824 Rev. 04, Upper End Fitting Details.

3.2 PaR Dwg. D-25690-D, Grapple, Refueling machine.

3.3 PaR Dwg. D-24767-D, Outer Guide, Grapple.

3.4 Drawing D-SYS80-630-120 Rev. 02, New Fuel Handling Tool Assy and Details.

3.5 "Design of Machine Elements," 3rd Edition, MF Spotts.

4.0 Method of Review

Three different methods were used to assess the potential for any interference. Two of them were "paper and pencil" layouts at four times full scale: one by an engineer and another by a draftsman. The other was an engineering analysis. In all cases the tolerances were used in a manner that would increase any potential for interference.

5.0 Results of Review

5.1 All three methods came to the same conclusion: the only potential for interference between the C-E supplied fuel and handling tools exists for the Spent Fuel Handling Tool.

5.2 The three methods of review assumed that all tolerances at all locations of all pieces would be at their most adverse limit simultaneously. Since this assumption is not statistically valid, a more realistic approach was used to quantify the magnitude of the potential interference between the fuel hold down plate (Ref. 3.1) and the Spent Fuel Handling Tool (Ref. 3.3).

5.3 The result of the draftsman's layout for the three tools reviewed is included as Appendix A. As can be seen, the only tool which has a potential for interference is the Spent Fuel Handling Tool.

5.4 The amount of potential interference with the Spent Fuel Handling Tool based on the most adverse tolerances combined linearly is approximately 0.036 inches.

5.5 Utilizing the equation shown on page 508 of Ref. 3.5, a more realistic value for the potential interference becomes 0.003 inches.

5.6 The investment casting process used for manufacturing the C-E supplied fuel assembly hold down plates results in excellent uniformity from piece to piece. Therefore, any interference would become evident when the first bundle is handled at initial core load.

5.7 Since the hold down plate manufacturer can be confident that the dimensions will be uniform from piece to piece, they produce the parts so that dimensions are nearer to the minimum material condition, thereby saving money while still meeting drawing requirements. This has been confirmed by measurement, and reduces the potential for interference.

6.0 Conclusions and Recommendations

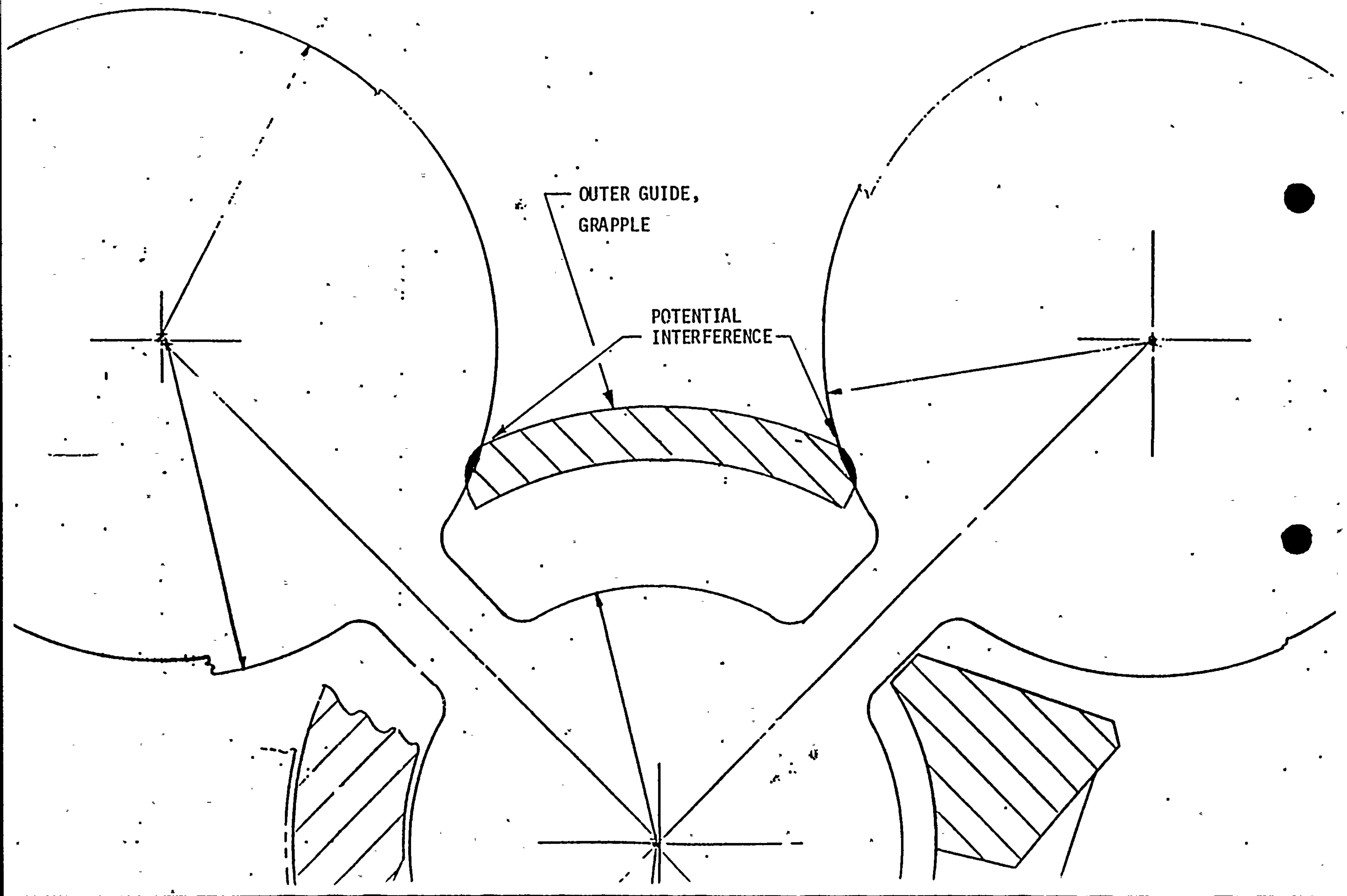
6.1 Since the potential interference between the fuel and the Spent Fuel Tool would prevent the tool from going onto the fuel, this precludes the safety concern resulting from the possibility of dropping the fuel. If the tool can fit onto the fuel, the tool can then be latched securely. For the other tools, no condition was identified that would allow only partial engagement of the lifting hooks with C-E fuel.

6.2 Since the manufacturing process used for the fuel hold down plates normally yields parts closer to the minimum material condition rather than the maximum material condition, different tolerances can be used in the interference calculations. In this case, the potential interference in Para. 5.5 reduces to zero.

6.3 Any modifications made to the spent fuel handling tool to increase clearances in the area of potential interference would tend to reduce the amount of engagement between the tool and fuel when the bundle is being lifted.

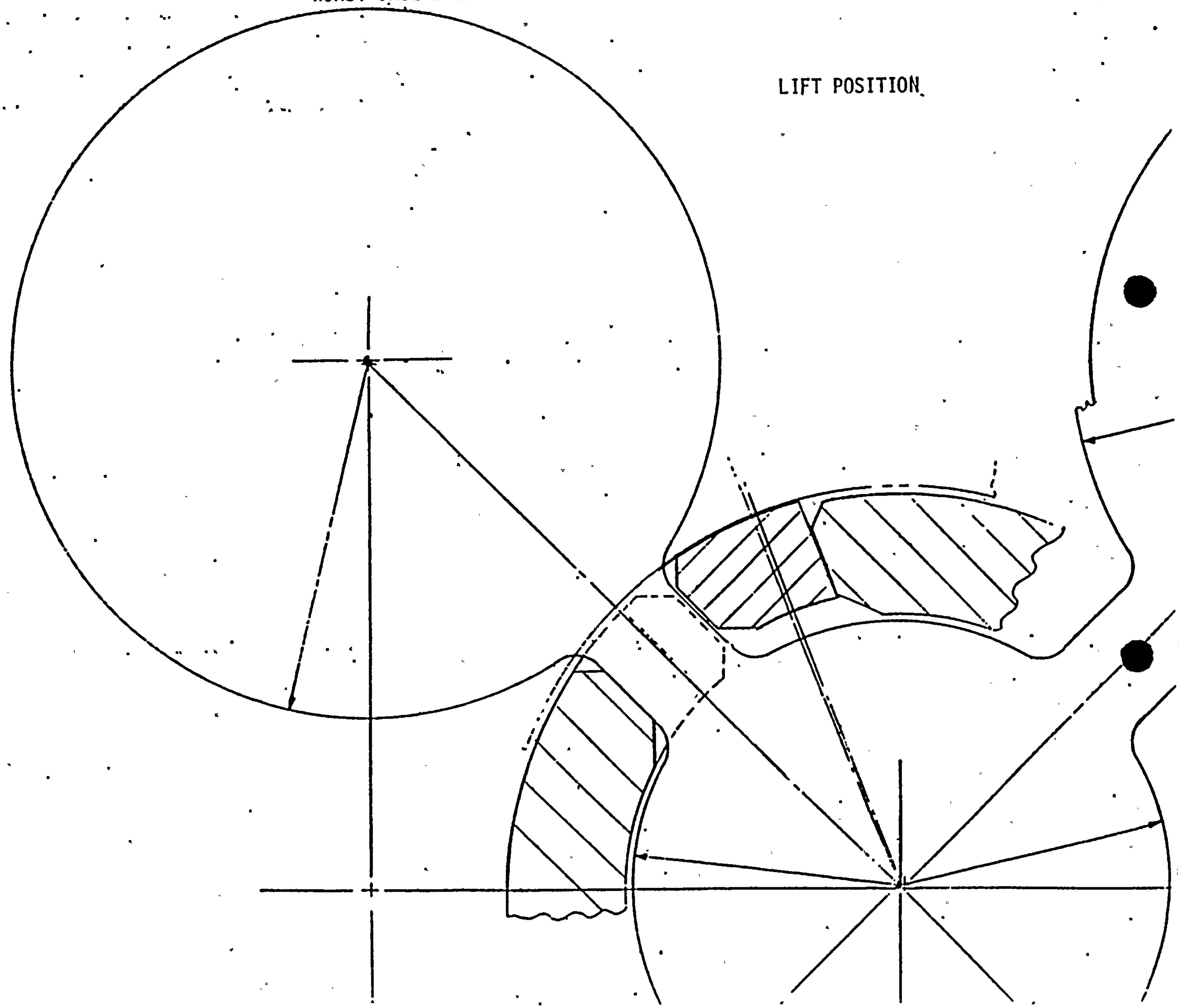
6.4 Based on the small magnitude of any potential interference, the strong possibility that the clearances will be larger than nominal and the adverse effect an additional clearance increase would have on the grapple engagement, it is recommended that no modifications be made to the spent fuel handling tool.

PALO VERDE SPENT FUEL HANDLING TOOL/CE CAST HOLDDOWN PLATE
WORST CASE LAYOUT

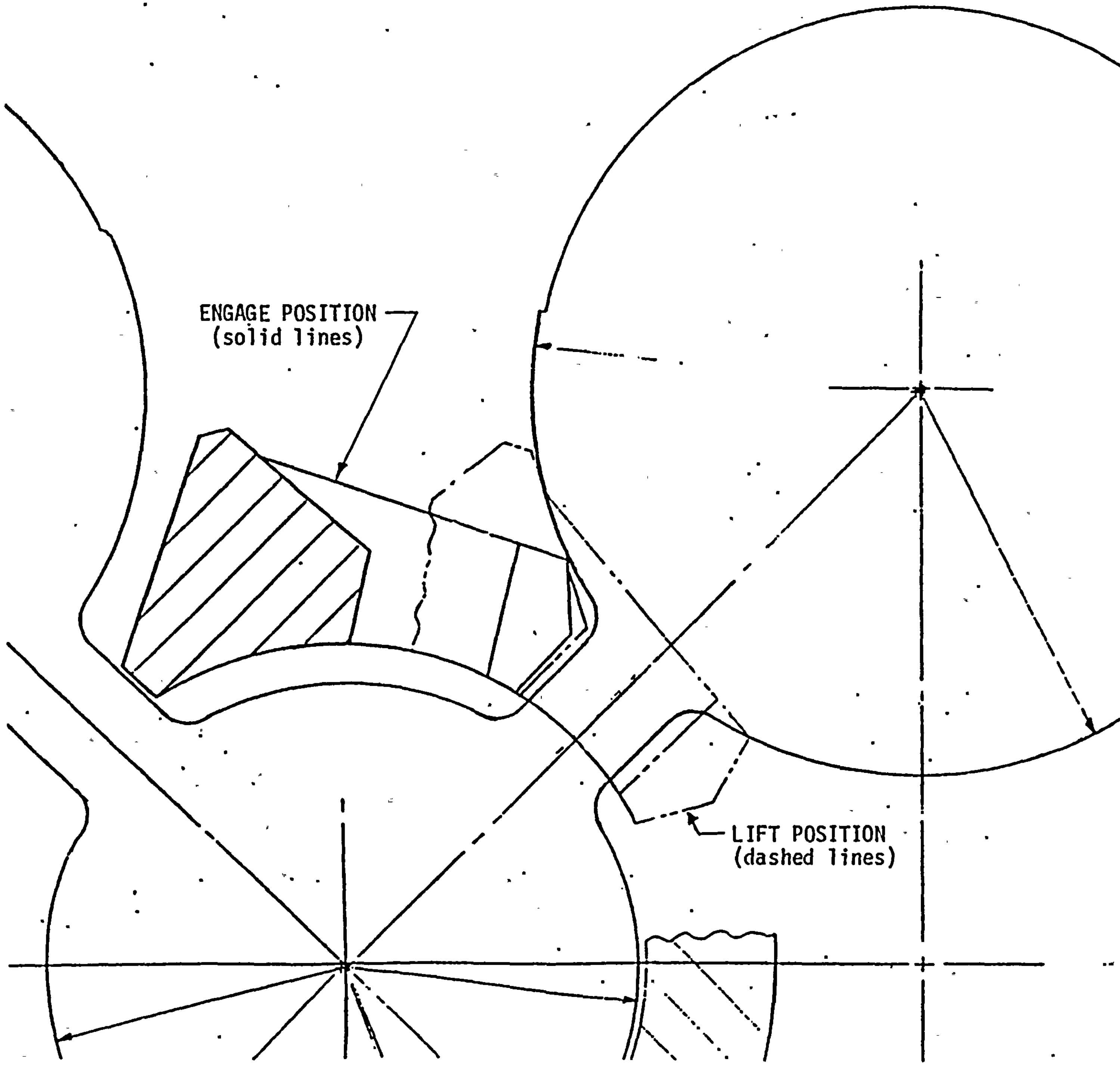


PALO VERDE NEW FUEL HANDLING TOOL/CE CAST HOLDDOWN PLATE

WORST CASE LAYOUT



PALO VERDE REFUELING GRAPPLE/CE CAST HOLDDOWN PLATE
WORST CASE LAYOUT



ATTACHMENT 4

LIST OF FIGURES

AZIMUTHAL POWER TILT - T_q (Continued)

P_{tilt}/P_{untilt} is the ratio of the power at a core location in the presence of a tilt to the power at that location with no tilt.

The AZIMUTHAL POWER TILT allowance used in the CPCs is defined as the value of CPC addressable constant TR-1.0.

3/4.2.4 DNBR MARGIN

The limitation on DNBR as a function of AXIAL SHAPE INDEX represents a conservative envelope of operating conditions consistent with the safety analysis assumptions and which have been analytically demonstrated adequate to maintain an acceptable minimum DNBR throughout all anticipated operational occurrences, ~~of which the loss of flow transient is the most limiting.~~ Operation of the core with a DNBR at or above this limit provides assurance that an acceptable minimum DNBR will be maintained in the event of a loss of flow transient.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the DNBR channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the DNBR does not violate its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core operating limit corresponding to the allowable minimum DNBR. ~~Reactor operation at or below this calculated power level assures that the limits of Figure 3.2-1 are not violated~~. The COLSS calculation of core power operating limit based on DNBR includes appropriate penalty factors which provide, with a 95/95 probability/confidence level, that the core power limits calculated by COLSS (based on the minimum DNBR Limit) ~~is~~ are conservative with respect to the actual core power limit. These penalty factors are determined from the uncertainties associated with planar radial peaking measurement, engineering heat flux, state parameter measurement, software algorithm modelling, computer processing, rod bow, and core power measurement.

Parameters required to maintain the margin to DNB and total core power are also monitored by the CPCs. Therefore, in the event that the COLSS is not being used, operation within the limits of Figures ~~3.2-2~~ 3.2-2 and 3.2-2a can be maintained by utilizing a predetermined DNBR as a function of AXIAL SHAPE INDEX and by monitoring the CPC trip channels. The above listed uncertainty and penalty factors are also included in the CPCs which assume a minimum core power of 20% of RATED THERMAL POWER. The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being less ~~in~~accurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings.

A ~~The~~ DNBR penalty factor~~s~~ has been included in the COLSS and CPC DNBR calculations ~~listed in Specification 4.2.4.4 are penalties used~~ to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher average burnup will experience a greater magnitude of rod bow. Conversely, lower burnup assemblies will experience less rod bow. In design calculations the penalty for each batch required to compensate for rod bow is determined from a batch's maximum average assembly burnup applied to the batch's maximum integrated planar-radial power peak. A single net penalty for COLSS and CPC is then determined from the penalties associated with each batch, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

JUSTIFICATION FOR ASI AMENDMENT

Removal of the numerical values from the ASI LCO as stated in Attachment 13 of the Palo Verde Unit 1, Cycle 2 reload Technical Specification amendment is needed in order to reduce the possibility of requiring an exigent change at a later date. The underlying facts and bases that support this requested change to the ASI Technical Specification follow below. It should be noted that the proposed change is fully consistent with the requirements for monitoring other parameters such as DNBR in Technical Specification 3/4.2.4.

The safety analyses performed in support of the Unit 1, Cycle 2 reload were performed parametrically over a range of initial ASI's from -0.30 to +0.30. If the change in Attachment 13 is not granted and the ASI LCO is restricted to a range from -0.28 to +0.28, then an allowance of only 0.02 ASI units is reserved to account for the uncertainties associated with ASI measurement. Since the uncertainty analysis is the last major analysis performed, the results are not generally known until approximately 30 days before startup. If, at that time, the ASI measurement uncertainty is greater than 0.02, then an exigent Technical Specification change would have to be requested to restrict the ASI LCO range to something less than -0.28 to +0.28. Past experience has shown that in general the ASI uncertainty is less than the allowance but specific instances have occurred where the allowance has been exceeded.

By granting the change requested in Attachment 13, the possibility of ANPP requesting an exigent Technical Specification change to the ASI LCO will be reduced. The requested change is fully consistent with the role that COLSS fulfills in the monitoring of LCO's. The change is also consistent with the requirements for monitoring other important parameters such as DNBR.

ATTACHMENT 6

7.1 Increase in Heat Removal by the Secondary System

7.1.1 Decrease in Feedwater Temperature

The results are bounded by the Reference Cycle.

7.1.2 Increase in Feedwater Flow

The results are bounded by the Reference Cycle.

7.1.3 Increased Main Steam Flow

The results are bounded by the Reference Cycle.

7.1.4 Inadvertent Opening of a Steam Generator Safety Valve or Atmospheric Dump Valve

The results are bounded by the Reference Cycle.

7.1.5 Steam System Piping Failures

7.1.5a Steam System Piping Failures: Inside and Outside Containment Pre-Trip Power Excursions

This event credits the software VOPT which is being added for Cycle 2 as part of the CPC Improvement Program. The CPC VOPT logic, which is designed to provide protection during fast power excursions, would generate a reactor trip more quickly than the CPC DNBR trip logic for the limiting SLB event. The NRC accepted the CPC VOPT for SONGS Units 2 and 3 (Reference 7-14). The analysis results are bounded by the Reference Cycle.

7.1.5b Steam System Piping Failures: Post-Trip Return to Power

The Steam Line Break event at zero power initial conditions was re-evaluated because the Cycle 2 Moderator cooldown reactivity insertion curve is more adverse than the Cycle 1 curve. Figure 7.1.5-1 compares the two curves. In addition, a sweep-out volume of 119 ft^3 before Safety Injection reaches the RCS was assumed for Cycle 2, which is more conservative than the 34.7 ft^3 assumed for the Reference Cycle.