f. Except for low power physics tests, the shutdown margin with allowance for a stuck control rod shall exceed the applicable value shown on Figure 3.2-2 under all steady-state operating conditions from zero to full power, including effects of axial power distribution. The shutdown margin as used here is defined as the amount by which the reactor core would be subcritical at hot shutdown conditions (540°F) if all control rods were tripped, assuming that the highest worth control rod remained fully withdrawn, and assuming no changes in xenon, boron concentration or part-length rod position.

g. During physics tests and control rod exercises, the insertion limits need not be met, but the required shutdown margin, Figure 3.2-2 must be maintained or exceeded.

2. MISALIGNED CONTROL ROD
If a part length* or full length control rod is more than 12 steps out of alignment with its bank, and is not corrected within 8 hours, power shall be reduced so as not to exceed 75% of interim power for 3 loop or 45% or interim power for two loop operation, unless the hot channel factors are shown to be no greater than allowed by Section 6a of Specification 3.2

3. ROD DROP TIME
The drop time of each control rod shall be no greater than 2.4 seconds at full flow and operating temperature from the beginning of rod motion to dashpot entry.

4. INOPERABLE CONTROL RODS
a. No more than one inoperable control rod shall be permitted during sustained power operation, except it shall not be permitted if the rod has a potential

* Any reference to part-length rods no longer applies after the part-length rods are removed from the reactor.

The various control rod banks are each to be moved as a bank, that is, with all rods in the bank within one step (5/8-inch) of the bank position. The control system is designed to permit individual rod movement for test purposes. Position indication is provided by two methods: a digital count of actuating pulses which shows the demand position of the banks and a linear position indicator (LVDT) which indicates the actual rod position.[2] The relative accuracy of the linear position indicator (LVDT) is such that, with the most adverse error, an alarm will be actuated if any two rods within a bank deviate by more than 15 inches. In the event that an LVDT is not in service, the effects of a malpositioned control rod are observable on nuclear and process information displayed in the control room and by core thermocouples and in-core movable detectors. Complete rod misalignment (part-length* or full-length control rod 12 feet out of alignment with its bank) does not result in exceeding core limits in steady-state operation at rated power. If the condition cannot be readily corrected, the specified reduction in power to 75% (3 loop) or 45% (2 loop) will insure that design margins to core limits will be maintained under both steady-state and anticipated transient conditions. The 8-hour permissible limit on rod misalignment is short with respect to the probability of an independent accident. The 24-hour period ensures that no significant burnup effects would be caused by the inserted rod.

The specified rod drop time is consistent with safety analyses that have been performed.[X]

The In-Core Instrumentation has five drives with detectors each of which has ten thimbles assigned[3]. This provides broad capability for detailed flux mapping.

The ion chambers located outside the reactor vessel measure flux distribution at the top and bottom of the core. Core traverses in a few of the in-core instrument paths will establish that the fixed flux measurement equipment is properly calibrated.

Operating experience has established that the flux measurement system is of a reliable design, and that the 10% load reduction, in the event of recalibration delay, is ultra conservative compensation.

References:
(1) FSAR - Section 14
(2) FSAR - Section 7.2
(3) FSAR - Section 7.6
(X) FPL licensing submittal for transition cores to the NRC

* Any reference to part-length rods no longer applies after the part-length rods are removed from the reactor.

5.2 REACTOR

REACTOR CORE

1. The reactor core contains approximately 71 metric tons of uranium in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy - 4 tubing to form fuel rods. The reactor core is made up of 157 fuel assemblies. Each fuel assembly contains 204 fuel rods.

2. The average enrichment of the initial core is a nominal 2.50 weight per cent of U-235. Three fuel enrichments are used in the initial core. The highest enrichment is a nominal 3.10 weight per cent of U-235.

3. Reload fuel will be similar in design to the initial core. The enrichment of reload fuel will be no more than 3.5 weight per cent of U-235.

4. Burnable poison rods in the form of rod clusters, which are located in vacant rod cluster control guide tubes are used for reactivity and/or power distribution control.

5. There are 45 full-length RCC assemblies and 8 partial-length* RCC assemblies in the reactor core. The full-

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* Any reference to part-length rods no longer applies after the part-length rods are removed from the reactor.

B2.1 Bases for Safety Limit, Reactor Core

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95 percent probability with 95% confidence that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used. The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

The curves of Figures 2.1-1, 2.1-1a, and 2.1-1b show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the calculated DNBR is no less than the design DNBR value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

The curves are based on a enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.55 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} \leq 1.55\ [1 + 0.2\ (1 - P)]$$

where P is the fraction of RATED THERMAL POWER.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion limit assuming the axial power imbalance is within the limits of the f(Δq) function of the Overtemperature ΔT trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core safety limits.

The f(Δq) function in the Overpower ΔT and Overtemperature ΔT protection system setpoints includes effects of fuel densification on core safety limits. The setpoints will ensure that the safety limit of centerline fuel melt will not be reached and the applicable design limit DNBR will not be violated.[10]

Pressurizer

The low pressurizer pressure reactor trip trips the reactor in the unlikely event of a loss-of-coolant accident.[6]

The high pressurizer pressure reactor trip is set below the set pressure of the pressurizer safety valves and limits the reactor operating pressure range. The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. The specified setpoint allows margin for instrument error[3] and transient level overshoot before the reactor trips.

Reactor Coolant Flow

The low flow reactor trip protects the core against DNB in the event of loss of one or more reactor coolant pumps. The setpoint specified is consistent with the value used in the accident analysis.[7] The low frequency and under voltage reactor trips protect against a decrease in flow. The specified setpoints assure a reactor trip signal before the low flow trip point is reached. The underfrequency trip setpoint preserves the coastdown energy of the reactor coolant pumps, in case of a system frequency decrease, so DNB does not occur. The undervoltage trip setpoint will cause a trip before the peak motor torque falls below 100% of rated torque.

Steam Generators

The low-low steam generator water level reactor trip assures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting of the auxiliary feedwater system.[8]

Reactor Trip Interlocks

Specified reactor trips are by passed at low power where they are not required for protection and would otherwise interfere with normal operation. The prescribed set points above which these trips are made functional assures their availability in the power range where needed.

An automatic reactor trip will occur if any pump is lost above 55% power which will prevent the minimum value of the DNBR from going below the applicable design limit during normal and anticipated transient operations when only two loops are in service,[9] and the overtemperature ΔT trip setpoint is adjusted to the value specified for three loop operation.

Reset of reactor trip interlocks will be done under strict administrative control.

References

(1) FSAR 14.1.1
(2) FSAR 14.1.2
(3) FSAR 14.1
(4) FSAR 7.2, 7.3
(5) FSAR 3.2.1
(6) FSAR 14.3.1
(7) FSAR 14 (page 14-30 and 14.1.9)
(8) FSAR 14.1.11
(9) FSAR 14.1.9
(10) WCAP-8074

B.3.1 BASES FOR LIMITING CONDITIONS FOR OPERATION, REACTOR COOLANT SYSTEM

1. Operational Components

The specification requires that significant number of reactor coolant pumps be operating to provide coastdown core cooling in the event that a loss of flow occurs. The flow provided will keep DNBR well above the applicable design limit. When the boron concentration of the Reactor Coolant System is to be reduced the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the reactor coolant system volume in approximately one half hour.

Each of the pressurizer safety valves is designed to relieve 283,300 lbs. per hr. of saturated steam at the valve setpoint Below 350°F and 450 psig in the Reactor Coolant System, the Residual Heat Removal System can remove decay heat and thereby control system temperature and pressure. If no residual heat were removed by any of the means available the amount of steam which could be generated at safety valve lifting pressure would be less than the capacity of a single valve. Also, two safety valves have capacity greater than the maximum surge rate resulting from complete loss of load.[2]

The 50°F limit on maximum differential between steam generator secondary water temperature and reactor coolant temperature assures that the pressure transient caused by starting a reactor coolant pump when cold leg temperature is $\leq$ 275°F can be relieved by operation of one Power Operated Relief Valve (PORV). The 50°F limit includes instrument error.

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above the applicable design limit during all normal operations and anticipated transients. In power operation with one reactor coolant loop not in operation this specification requires that the plant be in at least Hot Shutdown within 1 hour.

In Hot Shutdown a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be operable.

In Cold Shutdown, a single reactor coolant loop or RHR coolant loop provides sufficient heat removal capability for removing decay heat, but single failure considerations require that at least two loops be operable. Thus, if the reactor coolant loops are not operable, this specification requires two RHR loops to be operable.

Design criteria have been chosen for normal and operating transient events which are consistent with the fuel integrity analyses. These relate to fission gas release, pellet temperature and cladding mechanical properties. Also, the minimum DNBR in the core must not be less than the applicable design limit in normal operation or in short term transients.

In addition to conditions imposed for normal and operating transient events, the peak linear power density must not exceed the limiting Kw/ft values which result from the large break loss of coolant accident analysis based on the ECCS Acceptance Criteria limit of 2200°F. This is required to meet the initial conditions assumed for loss of coolant accident. To aid in specifying the limits on power distribution, the following hot channel factors are defined.

$F_Q(Z)$, Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

F_Q^E, Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

$F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to average rod power.

It should be noted that $F_{\Delta H}^N$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus, the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F_{\Delta H}^N$.

$$W(Z) = \text{Max}\left(\frac{F_Q(Z)(\text{Base Load Case(s), 150 MWD/T})}{F_Q(Z)(\text{ARO, 150 MWD/T})}, \frac{F_Q(Z)(\text{Base Case(s), 85\% EOL BU})}{F_Q(Z)(\text{ARO, 85\% BOL BU})}\right)$$

For Radial Burndown operation the full spectrum of possible shapes consistent with control to a ± 5% ΔI band needs to be considered in determining power capability. Accordingly, to quantify the effect of the limiting transients which could occur during Radial Burndown operation, the function $F_z(Z)$ is calculated from the following relationship:

$$F_z(Z) = [F_Q(Z)]_{\text{FAC Analysis}}/[F_{xy}(Z)]_{\text{ARO}}$$

As discussed above, the essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This can be accomplished without part length rods* by using the boron system to position the full length control rods to produce the required indicated flux difference.

For Operating Transient events, the core is protected from overpower and a minimum DNBR of less than the applicable design limit by an automatic protection system. Compliance with operating procedures is assumed as a precondition for Operating Transients; however, operator error and equipment malfunctions are separately assumed to lead to the cause of the transients considered.

Above the power level of P_T, additional flux shape monitoring is required. In order to assure that the total power peaking factor, F_Q, is maintained at or below the limiting value, the movable incore instrumentation will be utilized. Thimbles are selected initially during startup physics tests so that the measurements are representative of the peak core power density. By limiting the core average axial power distribution, the total power peaking factor F_Q can be limited since all other components remain relatively fixed. The remaining part of the total power peaking factor can be derived based on incore measurements, i.e., an effective radial peaking factor $\bar{R}$, can be determined as the ratio of the total peaking factor resulting from a full core flux map and the axial peaking factor in a selected thimble.

*Any reference to part-length rods no longer applies after the part-length rods are removed from the reactor.

References
FSAR - Section 14.3.2

Safety Evaluation

Florida Power and Light Company
Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251

1.0 Introduction

Turkey Point Units 3 and 4 have been operating with all Westinghouse 15x15 low-parasitic (LOPAR) fueled cores. For Turkey Point 3, Cycle 9 and Turkey Point 4, Cycle 10 and subsequent cycles, it is planned to refuel both units with 15x15 optimized fuel assembly (OFA) regions supplied by the Westinghouse Electric Corporation. As a result, future core loadings would range from approximately a 1/3 OFA-2/3 LOPAR mixed core to eventually an all OFA fueled core. The OFA fuel will decrease neutron parasitic capture and thereby attain more efficient fuel usage. The 15x15 OFA fuel has design features similar to 15x15 LOPAR fuel which has had satisfactory operating performance in a number of nuclear plants. The major design difference involves the use of 5 intermediate Zircaloy grids for the OFA fuel versus 5 intermediate Inconel grids for LOPAR fuel.

The 15x15 OFA grids have design features similar to the Westinghouse 17x17 OFA which has been generically approved by the NRC via the review of WCAP-9500, "Reference Core Report - 17x17 Optimized Fuel Assembly" (Reference 1). Operating experience has been obtained on six demonstration 17x17 OFAs which contained Zircaloy intermediate grids (Reference 2). Two assemblies have satisfactorily completed three cycles of irradiation to about 28,000 MWD/MTU burnup, two have completed two cycles to about 19,400 MWD/MTU, and two have completed one cycle in excess of 9,000 MWD/MTU. All the demonstration OFAs were examined and provide performance data for the 15x15 OFA design.

The Westinghouse Wet Annular Burnable Absorber (WABA) rods will be a new feature introduced with the OFA fuel in order to enhance fuel cycle economy. A Westinghouse generic WABA evaluation topical (Reference 3) has been submitted for NRC review and approval. Also, new thimble plugs will be used to accomodate a reduced OFA thimble tube diameter.

2.0 Summary and Conclusions

Consistent with the Westinghouse standard reload methodology for analyzing cycle specific reloads (Reference 4), parameters were chosen to maximize the applicability of the transition evaluations for each reload cycle and to facilitate subsequent determinations of the applicability of 10 CFR 50.59. The mechanical, thermal and hydraulic, nuclear, and accident evaluations considered the transition core effects described for mixed cores in Chapter 18 of Reference 1. The summary of these evaluations for the Turkey Point Units 3 and 4 transitions to an all OFA core are given in the following sections of this submittal.

The transition design and safety evaluations consider the following nominal conditions which are consistant with 100% of FSAR thermal design flow. These nominal conditions are 2200 MWt core power, 2250 psia system pressure, 546.2°F core inlet temperature, and 268,500 gpm RCS thermal design flow.

The results of evaluation/analyses and tests discussed in this report lead to the following conclusions:

2.1 The Westinghouse OFAs and WABA rods/assemblies are mechanically and hydraulically compatible with the LOPAR fuel assemblies, control rods, and reactor internals interfaces.

2.2 Demonstration experience with 17x17 OFAs containing Zircaloy grids provides evidence of the satisfactory operation of 15x15 OFA Zircaloy grids.

2.3 As shown in its evaluation topical, Reference 3, the WABA rod design is compatible with the OFA and LOPAR assembly and satisfies all performance requirements for its design life.

2.4 Changes in the nuclear characteristics due to the transition from LOPAR to OFA fuel will be within the range normally seen from cycle-to-cycle due to fuel management effects.

2.5 Plant operating limitations given in the Technical Specifications will be satisfied with the proposed changes noted in Section 7.0 of this report.

3.0 Mechanical Evaluation

The OFAs have been designed to be compatible with the LOPAR assemblies, reactor internals interfaces, and fuel handling and refueling equipment. Figure 1 presents a comparison of the OFA and LOPAR fuel assembly. The grid elevations for the two assembly designs match, minimizing mechanical and hydraulic interaction. The assembly envelopes, fuel rod design/dimensions, and the top and bottom Inconel grids are the same.

The five intermediate 15x15 OFA Zircaloy grids have thicker and wider straps than the 15x15 LOPAR Inconel grids to compensate for the difference in material strength properties. Impact tests that have been performed at 600°F to obtain the dynamic strength data verify that the Zircaloy grid strength data at reactor operating conditions is structurally acceptable. The 15x15 OFA Zircaloy grid design has approximately 7 percent less crush strength than the 15x15 Inconel grid design.

The 15x15 OFA guide thimbles are similar in design to their counterparts in the LOPAR fuel assemblies except for 13 mil ID and OD reduction in the guide thimble above the dashpot. The reduction in OFA thimble diameter is due to the use of the thicker Zircaloy grid straps. The new design continues to provide an adequate nominal diametral clearance for control rods as well as other core components. Due to the reduced clearance, the rod drop time to the dashpot for accident analyses has conservatively been determined to increase from 1.8 seconds for the LOPAR assembly to 2.4 seconds for the OFA. The increase in rod drop time required accident reanalyses as described in Section 6.0.

The OFA has about a 4.5 percent increase in hydraulic resistance to flow compared to a LOPAR assembly, primarily due to the thicker and wider OFA Zircaloy grid straps. This results in an increased OFA lift force and requires the use of 3-leaf holddown springs in the top nozzle instead of the LOPAR assembly 2-leaf springs (see Figure 1). The 3-leaf spring has the same height and provides additional holddown force margin when compared to the LOPAR 2-leaf spring. The 3-leaf spring design has been successfully used in the 17x17 OFA demonstration program and other 15x15 LOPAR assemblies. The change to the 3-leaf spring is fully compatible with the LOPAR assembly and the handling tools at the Turkey Point plants. The 15x15 OFA bottom nozzle has similar design features as the LOPAR assembly bottom nozzle. The OFA bottom nozzle design has a reconstitutable feature which allows it to be easily removed. A locking cup is used to lock the thimble screw of a guide thimble tube in place, instead of the lockwire as used for the standard LOPAR nozzle design. The reconstitutable nozzle design facilitates remote removal of the bottom nozzle and relocking of thimble screws as the bottom nozzle is reattached.

The rod bow magnitude for the optimized fuel assemblies is expected to be less than that of the 15x15 LOPAR assemblies. The optimized fuel assembly will have reduced grid forces (due to Zircaloy grid) and the same fuel tube thickness-to-diameter ratio (t/d) as the standard assembly, which should tend to decrease OFA rod bow compared to LOPAR fuel. For a given burnup, the magnitude of rod bow DNBR penalty for the optimized fuel assembly is conservatively taken to be the same as that applied to the 15x15 LOPAR fuel assembly.

The wear of fuel rod cladding is dependent on both the support provided by the assembly skeleton and the flow environment to which it is subjected. Hydraulic flow tests as described in Section 5.0 were performed to verify the compatibility of the 15x15 OFA and LOPAR designs. The results of the tests show that no significant OFA or LOPAR fuel rod clad wear occurs due to the small amount of crossflow between fuel assemblies.

4.0 Nuclear Evaluation

The transition from LOPAR to OFA fuel will not result in changes from the current nuclear design bases given in the FSAR[5] and applied to subsequent Unit 3 and 4 Reload Safety Evaluation (RSE)s. Although the physics characteristics are slightly different for the OFA fuel when compared to LOPAR, evaluations show that differences are within the normal range of variations seen from cycle to cycle.

Changes in the nuclear characteristics in transition from a LOPAR to OFA cores will be primarily due to fuel management considerations (number of feed assemblies, feed enrichment, cycle burnup, etc.) and not due to the change in fuel assembly design to OFA fuel. The standard calculational methods as described in WCAP-9273, "Westinghouse Reload Safety Evaluation Methodology"[4] continue to apply. As is current practice, each reload core design will be evaluated to assure that design and safety limits are satisfied according to the reload methodology[4].

5.0 Thermal and Hydraulic Evaluation

The hydraulic effects of having a mixed OFA-LOPAR core were evaluated by performing hydraulic tests at the Westinghouse Fuel Assembly Test System (FATS) facility. A side-by-side OFA and LOPAR fuel assembly arrangement was tested under hydraulic flow conditions which approximate reactor conditions. Test results provided lift forces, pressure drops, cross flow and fuel vibrations, and fuel rod clad wear. Based on these test results, it was concluded that hydraulic compatibility exists between OFA and LOPAR assemblies.

Calculational methods for the 15x15 OFA are the same as those currently used on the 15x15 LOPAR fuel assembly and described in the FSAR[5] and fuel densification documents[6]. These methods are applicable to the

evaluation of a core containing both 15x15 LOPAR and 15x15 OFAs, except for the application of the approved WRB-1 DNB correlation[7] for the OFAs. The present DNB safety evaluations for LOPAR fuel use the W-3 L-Grid DNB correlation with a design limit DNBR of 1.30. The evaluations contain a generic DNBR margin of 18.0 percent. Due to significant improvement in the accuracy of the critical heat flux prediction with the WRB-1 correlation[7] over previous DNB correlations a DNBR limit of 1.17 is used in this application. In addition to the use of the WRB-1 correlation, a specific plant DNBR margin allowance has been considered in the Turkey Point OFA analysis. In particular, the DNBR value of 1.56 was employed in safety analyses, resulting in 25 percent DNBR margin as defined in the following equation:

$$\frac{\text{DNBR LIMIT}}{\text{1-MARGIN}} = \frac{1.17}{1-0.25} = 1.56$$

The plant allowance available between the DNBR's used in the safety analyses and the WRB-1 design limit DNBR value of 1.17 is not required to meet the design basis, but will be used for the flexibility in the design, operation, and analyses for Turkey Point Units 3 and 4. For instance, the allowance may be used for improved fuel management or increased plant availability. In the Turkey Point transition cycles, the 25 percent margin is more than sufficient to cover the maximum 14 percent rod bow penalty at Loss of Flow Conditions[8] and a 3 percent transition core penalty. The 3 percent penalty was determined by analyzing 15x15 OFA and LOPAR assembly loading patterns at various core conditions in the same manner as for the 17x17 OFA/LOPAR fuel which was reviewed and approved by the NRC[9]. When the full transition is complete (all LOPAR assemblies removed from core), the transition core penalty will no longer apply to OFA assemblies.

The current Technical Specficification core safety limits (T.S. Figures 2.1-1, 2.1-1a, and 2.1-1b) and the associated set points continue to be applicable for the transition cores and the all OFA cores.

6.0 Accident Evaluation

Those accidents analyzed and reported in the FSAR[5] which could potentially be affected by the OFA reload have been reviewed. As discussed in Section 3.0, rod drop time is increased from 1.8 to 2.4 seconds (to dashpot) for control rods in OFA guide thimbles. Accident transients significantly affected are "fast" transients for which the protection system responds by tripping the reactor within a few seconds after the transient begins. The transients that fall into this category are Loss of Flow, Locked Rotor, and Rod Ejection. All other non-LOCA accidents analyzed in the FSAR are evaluated to be minimally affected by the increased rod drop time.

The simultaneous Loss of Flow from all three coolant pumps, Locked Rotor, and Rod Ejection accidents were reanalyzed to account for the increased rod drop time. Results for these accidents showed that all the safety limits and criteria are satisfied for the increased rod drop time.

The difference in fuel assembly flow resistance (K/A^2) for the 15x15 LOPAR and 15x15 OFA designs impact the blowdown and reflood transients of a postulated large break LOCA. The 15x15 OFA increases the hydraulic resistance to flow by about 4.5 percent. Evaluations of hydraulic resistance increases has led to a conclusion that hydraulic resistance mismatches of less than 10 percent have an insignificant effect on blowdown cooling. Sensitivity studies have shown that for the reflood portion of the large LOCA transient, a flow reduction of five percent would lead to a PCT increase of 19°F. The hydraulic resistance increase for 15x15 OFA will cause approximately a 2.2 percent reduction in reflood flow rate for the 15x15 OFAs during the transition period. This will

result in approximately a 10°F increase in PCT for the transition period. Once a full core of 15x15 OFA is achieved, the reflood penalty due to the increased core hydraulic resistance will be about .02 percent in flooding rate. Therefore the transition from LOPAR to OFA does not significantly affect the current LOCA analysis (ΔPCT<20°F). If there is insufficient ΔPCT margin in the approved LOCA analysis to account for the 10°F PCT penalty, the measured F_Q will be administratively increased by 0.01 during the transition period. Once a full core of 15x15 OFA has been installed, an F_Q penalty will not be required.

7.0 Technical Specification Changes

Based on the preceeding evaluations the following technical specification changes for Turkey Point Units 3 and 4 are required to support the transition to OFA:

1. Modifications to Specification 3.2.3 and specification bases B3.2 to permit an increase in the shutdown and control rod drop time to 2.4 seconds.

2. Modifications to Specification 5.2.4 to permit the use of the WABA rods.

3. Modifications to Specification Bases: B2.1, B2.3, B3.1, and B3.2 incorporate the design limit DNBR for OFA.

These changes are given in the proposed Technical Specification page changes.

8.0 References

1. Letter from R. L. Tedesco (NRC) to T. M. Anderson (Westinghouse), Safety Evaluation of WCAP-9500, "Reference Core Report - 17x17 Optimized Fuel Assembly," NRC SER letter dated May 22, 1981.

2. Jones, R. G., Iorii, J. A., "Operational Experience with Westinghouse Cores (up to December 31, 1981)", WCAP-8183, Rev. 11, May 1982.

3. E. P. Rahe (Westinghouse) letter to C. Thomas (NRC) Number NS-EPR-2670, October 18, 1982, Subject: "Westinghouse Wet Annular Burnable Absorber Evaluation Report", WCAP-10021, Revision 1 (Proprietary).

4. Bordelon, F. M., et. al., "Westinghouse Reload Safety Evaluation Methodology", WCAP-9273 (Non-Prop.), March 1978.

5. Turkey Point Units 3 and 4 Final Safety Analysis Report", Docket Numbers: 50-250 and 50-251.

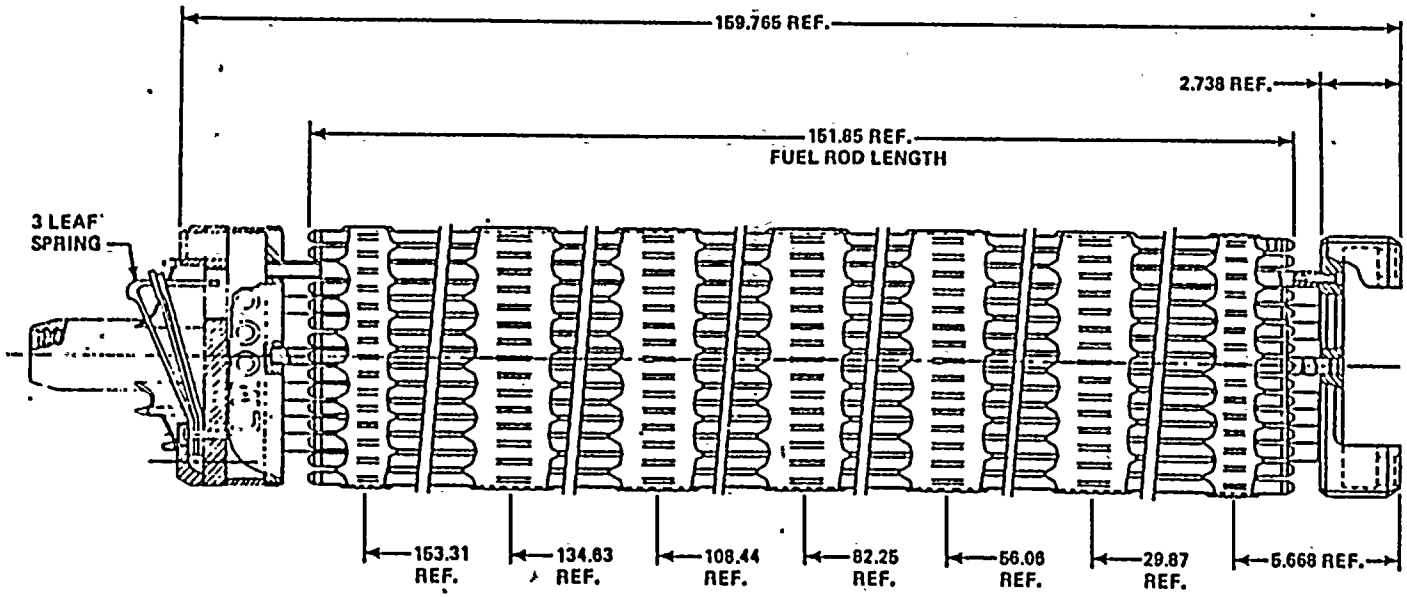
6. "Fuel Densification - Turkey Point Plant Units No. 3 and 4 Low Pressure Analysis", June 1973.

7. Letter from John F. Stolz (NRC) to C. Eicheldinger (Westinghouse), Safety Evaluation of WCAP-8762, "New Westinghouse Correlation WRB-1 Predicting Critical Heat Flux in Rod Bundles with Mixing Grids"; NRC SER letter dated April 19, 1978.

8. Stolz, J. F., NRC letter to T. M. Anderson, Westinghouse, "Staff Review of WCAP-8691, "April 5, 1979.

9. Letter from C. O. Thomas (NRC) to E. P. Rahe (Westinghouse), Subject: Supplemental Acceptance Number 2 for Referencing of Licensing Topical Report WCAP-9500, NRC SER letter dated January 24, 1983.

16X16 OPTIMIZED FUEL ASSEMBLY



16X16 LOPAR FUEL ASSEMBLY

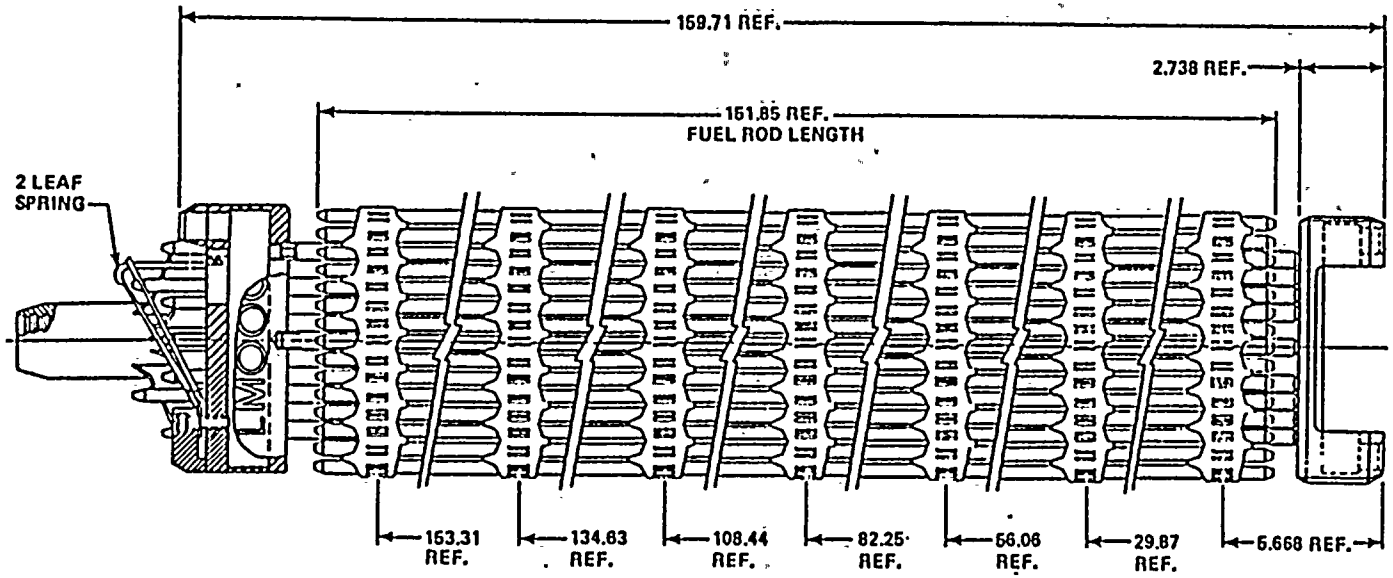


Figure 1