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NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 15 TO OPERATING LICENSE NO. NPF-30

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

1.0 INTRODUCTION

By letter dated November 15, 1985, Union Electric Company (the licensee) made application to amend the license of the Callaway Plant, Unit 1, in order to reload and operate the unit for Cycle 2. In support of the application the licensee provided a report entitled, "Safety Evaluation for the Callaway Plant Transition to Westinghouse 17x17 Optimized Fuel Assemblies". Further information was provided in response to NRC requests. Also provided were proposed Technical Specification changes to assure the safe operation of the plant.

2.0 EVALUATION

As part of the core reload for Cycle 2 the licensee has elected to initiate a transition from standard Westinghouse LOPAR fuel to Optimized Fuel Assembly (OFA) fuel. In addition the analyses are being performed under the assumption of a core power level of 3565 thermal megawatts (MWt) in preparation for a future power up-rating. The plant will continue to operate at 3411 MWT during Cycle 2. Also Wet Annular Burnable Absorber (WABA) fuel is being introduced in Cycle 2.

Analyses have been performed for cores having partial OFA loadings (the Cycle 2 core will consist of approximately 40% OFA fuel) and for a core consisting entirely of that fuel. The operating limits and protection system settings have been based on the most limiting of the core loadings.

2. FUEL EVALUATION

The use of 17x17 OFA fuel has been approved in other Westinghouse reactors (e.g., McGuire) and its use in Callaway is acceptable. This fuel has been designed to be compatible with Westinghouse standard (LOPAR) fuel in order to facilitate the transition from one fuel to the other. The mechanical behavior of the two fuels has been examined for the Callaway plant and it is concluded that all applicable criteria are met. We conclude that the fuel mechanical evaluation is acceptable.

Thermal evaluation of the fuel was performed with the PAD code. PAD is now the standard Westinghouse code for this purpose and its use by Callaway is acceptable.

3. NUCLEAR EVALUATION

The transition from LOPAR to OFA fuel has a minimal effect on the neutronic parameters of the core. No changes will be required in the current nuclear design bases. The analyses of the transition cores and of the all-OFA core were performed with the Reload Safety Evaluation Methodology which has been used and approved in other reactors. We conclude that the nuclear evaluation for the transition to OFA fuel is acceptable.

As part of the transition (but not required by it) the multiplier in the algorithm for obtaining the permitted value of $F^N_{\Delta H}$ as a function of power has been changed from 0.2 to 0.3. This has the effect of permitting higher values of $F^N_{\Delta H}$ at low power levels than was previously the case. Account has been taken of this change in the safety analysis. The change has been previously approved for other plants and we find it acceptable for Callaway.

4. THERMAL-HYDRAULIC EVALUATION

The Callaway plant has been operating with a 17x17 low-parasitic (LOPAR) fueled core. It is planned to eventually operate with a full core of 17x17 Optimized Fuel Assembly (OFA) fuel. The presence of transitional mixed cores containing both the standard LOPAR and OFA fuel requires that particular attention be paid to the thermal-hydraulic analysis of the core. Cycle 2 is the first Callaway cycle utilizing OFA fuel and will contain 109 LOPAR fuel assemblies and eighty-four 17x17 OFAs (approximately 43% OFA fuel). A number of the OFAs will employ the Wet Angular Burnable Absorber (WABA) rods. The core safety analyses have been performed at a core Design Thermal Power of 3565 MWt and a slightly reduced reactor coolant flow to account for up to 10% steam generator tube plugging. However, the Callaway Cycle 2 core will be operated at the currently licensed Rated Thermal Power of 3411 MWt.

The licensee has presented a safety evaluation for the Callaway plant for transition to Westinghouse 17x17 OFA fuel in Attachment B of Reference 1. In response to questions, the licensee supplied information (Ref. 2) on the thermal-hydraulic design comparison. This includes Table 1, which presents values for cores using both 17x17 LOPAR fuel and 17x17 OFA fuel and presents Cycle 2 operating parameters and design parameters. From Table 1 it is seen that the following values are constant for LOPAR and OFA fuel: reactor heat input, core pressure, total flow rate, nominal inlet temperature, average temperature rise, average linear power (kW/ft) and peak linear power (kW/ft). The active heat transfer surface area for the OFA fuel is smaller than for the LOPAR fuel (fuel rods have smaller O.D). Also, the average velocity along the fuel rods is less for OFA fuel. However, the average heat flux for the OFA fuel is larger than for the LOPAR fuel. The core pressure drop for a OFA fueled core and a LOPAR fueled core are 26.4 ± 2 psia and 26.5 ± 2.6 psia respectively and are therefore approximately the same.

The OFA and LOPAR fuel assemblies have been tested for hydraulic characteristics (Ref. 3) and they have been shown to be hydraulically compatible. Since the core pressure drops in an all LOPAR core and an all OFA core are approximately the same, the core flow remains the same also. The actual measured flow rate for the Callaway plant in the last cycle (Cycle 1) was approximately 411,000 gpm (Ref. 11), which is well over the Technical Specification minimum measured flow of 382,630 gpm as shown in Table 1. The OFA fuel assemblies should resist liftoff as the current holddown spring design remains the same as for the LOPAR fuel and the pressure drop and reactor system flow also remain approximately the same as before.

The thermal-hydraulic analysis of this mixed core was performed using the approved "Improved Thermal Design Procedure" (ITDP) (Ref. 4) together with the WRB-1 DNB correlation (Ref. 5). Use of this correlation for OFA fuel has been demonstrated and documented in WCAP-9401-A for 17x17 fuel which has been approved.

In the Improved Thermal Design Procedure the safety analyses are performed using nominal values of the plant operating, nuclear, thermal, and fuel fabrication parameters. Uncertainties in the DNBR value due to variations in these parameters are combined statistically and added to the correlation DNBR limit (1.17) to obtain the design DNBR limit. The values obtained for this quantity for Callaway are 1.32 for thimble cells (three fuel rods and a thimble tube) and 1.34 for typical cells (four fuel rods). The licensee has provided information concerning the plant specific uncertainties for Callaway which support these values. Transition core and rod bow effects are not included in the design DNBR limit. In order to account for these effects additional margin is provided to arrive at analysis values which are 1.42 and 1.45 for thimble and typical cells, respectively.

In response to a question, the licensee supplied information (Ref. 6) which provided responses to the eleven items listed in the NRC cover letter for Safety Evaluation of WCAP-9500-A (Ref. 7) for plants using ITDP. This included information on plant specific margins used to offset reduction in DNBR due to rod bowing and for the transition core penalty. The OFA fuel assemblies have sufficient margin (approximately 7%) between the safety analysis minimum DNBR and the design limit DNBR, as shown below, to accommodate the rod bow penalty and transition core penalty.

	17x17 OFA	
	THIMBLE	TYPICAL
Correlation	WRB-1	WRB-1
Correlation Limit	1.17	1.17
Design Limit	1.32	1.34
Safety Analysis Minimum DNBR	1.42	1.45

Because of the rod bow phenomena as described in Reference 8, rod bow DNBR penalities for full-flow and low-flow are required. These have been identified as being less than 3% using the information in Reference 9 which has been approved. This penalty is accommodated by the 7% margin available between the safety analysis minimum DNBR and the design DNBR limit.

The approved method of calculating the transition core DNB is given in Reference 7 from which a 2% DNBR transition core penalty is applied to the Callaway plant. Using this penalty, the transition core is analyzed as if it were a full core of OFAs. The 7% margin available between the safety analysis minimum DNBR and the design DNBR limit accommodates the 2% transition core DNBR penalty as well as the 3% rod bow DNBR penalty.

For the Callaway Cycle 2 core, WABA rods will be used instead of the glass absorbers of the Cycle 1 core. Since the WABAs provide an additional by-pass flow path (in the annulus of the absorber) they will slightly increase the total thimble tube by-pass flow. However, the number of WABA rods is well within the limit of acceptability as specified in Reference 10 which has been approved.

An RCS flow measurement uncertainty analysis, which is needed for the ITDP, was presented in Reference 6. This included a description of a generic calculational method (Appendix A of Reference 6) and a plant specific calculation (Appendix B of Reference 6). The plant specific calculation for Callaway supports a value of RCS flow measurement uncertainty of ±2.2%, which is the value used in the Callaway Technical Specifications. This value includes 0.1% to account for feedwater venturi fouling. The 2.2% value is based on using normalized elbow tap instrumentation readings after flow calorimetric measurements. We find the flow measurement analyses for the 2.2% flow measurement uncertainty to be acceptable.

5. TRANSIENTS AND ACCIDENTS

Each of the transients and accidents which were evaluated in the FSAR have been examined to determine whether a reanalysis is required to account for the effects of the transition from LOPAR to OFA fuel. The effects of the change in the $F^N_{\Delta H}$ multiplier and of the increase in design thermal power are also treated.

The change to OFA fuel affects the thermal-hydraulic performance of the fuel (see Section 4) in a negative way. In order to regain calculated margin the WRB-1 DNB correlation and the Improved Thermal Design procedure are used. Another effect of the use of OFA fuel is the increase in control rod scram time due to slightly reduced diameter of the guide tubes. This effect is accounted for in the analysis.

The increase in the $F^N_{\Delta H}$ multiplier is accounted for in establishing the core safety limits. The analyses are performed at the Design Thermal Power of 3565 MWt instead of the Current Rated Thermal Power of 3411 MWt. Each of the accidents reanalyzed is discussed below.

5.1 Increase in Heat Removal by the Secondary System
Events in this Category include:

1. Feedwater system malfunctions that result in a decrease in feedwater temperature.
2. Feedwater system malfunctions that cause an increase in feedwater flow.
3. Excessive increase in secondary steam flow.
4. Inadvertent opening of a steam generator relief or safety valve.
5. Steam supply piping failure.

The first four of these events are classified as Condition II events (anticipated transients) while the fifth is classified as a Condition IV event (design basis accident). Of the first three events the third (a 10% step increase in steam demand) is the limiting event. For that event analyses were performed with approved methods and procedures for both manual and automatic control at both minimum and maximum reactivity feedback. In no case was the DNBR safety limit violated. We conclude that the analysis of these events is acceptable.

For the fourth event (opening of a steam relief, safety or dump valve) a conservative calculation is performed with a bounding value of steam flow and hot standby conditions at end of cycle. The analysis shows that DNB does not occur for this event. Since the analysis was performed with acceptable methods and procedures and conservative input conditions were assumed we conclude that the analysis of this event is acceptable.

The fifth of these events is the steam line break accident. For this event the W-3 DNBR correlation was used for the W OFA fuel rather than the WRB-1 correlation as the minimum pressure falls below the range of the WRB-1 correlation (1440 ≤ P ≤ 2490 psia). The minimum pressure also falls below the pressure range given in most references (1000 psia) for the W-3 correlation. However, the licensee justified the use of the W-3 correlation for lower pressure based on data (Ref. 2) that showed no abnormality exists for pressure (the pressure does not show trends in predicted and measured DNB heat fluxes as a function of pressure), which reinforces its acceptablity. As part of the generic review for Westinghouse plants these data have been used to arrive at a new DNBR value slightly larger than 1.3 for the W-3 correlation for lower pressure. Also, the results of the analysis performed by the licensee (Ref. 11) show that the minimum DNBR value (over 1.8) during the SLB accident is well above the limit of 1.3. On the basis of the data presented and the substantial DNBR margin available, we find the W-3 correlation acceptable for the SLB analysis presented for Callaway.

Rupture of a steam pipe is assumed to include any accident which involves inadvertent steam release from a steam generator. Under no load conditions, a negative temperature coefficient, and the most reactive rod stuck out of the core, the cooldown would result in reduction of the shutdown margin.

Return to power would be a potential problem to the extent that there is a large increase in the hot channel factor when the highest reactivity rod is fully withdrawn. A number of protection systems will be activated in case of steam pipe rupture such as: safety injection, overpower trips, isolation of the feedwater lines and trip of the steam line isolation valves. The transient analysis is accomplished using the LOFTRAN code to compute the reactor and coolant system status and the THINC IV code to compute whether the DNB ratio falls below the minimum value. Analyses were performed using a .013 reactivity shutdown margin, a negative temperature coefficient corresponding to the EOC with all but the most reactive rod inserted, assumption of a single failure in the ECCS, power peaking factors corresponding to one rod stuck out, and different sizes of the steam line break. The results indicate that following a steamline break the DNBR will remain higher than the design DNBR limit. Therefore, the assumed reactivity shutdown margin is adequate and the results are acceptable.

5.2 Decrease in Heat Removal by Secondary System Events in this category include the following:

1. Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow
2. Loss of External Electrical Load
3. Turbine Trip
4. Inadvertent Closure of Main Steam Isolation Valves
5. Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip
6. Loss of non-emergency AC Power to the Station Auxiliaries
7. Loss of Normal Feedwater Flow
8. Feedwater System Pipe Break

The above items are considered to be ANS Condition II events, with the exception of a Feedwater System Pipe Break, which is considered to be an ANS Condition IV event.

The first event is not applicable to PWRs. The Loss of External Electrical Load event is less limiting than the Turbine Trip event. Events 4 and 5 are also bounded by the Turbine Trip event.

5.2.1 Turbine Trip

The Turbine Trip event is more severe than the loss of load event because of a more rapid loss of steam flow due to the more rapid closure of the turbine stop valve than is the case for the turbine control valve. The analysis is performed with the approved LOFTRAN code and the following assumptions are made:

1. Both minimum and maximum reactivity feedback calculations are performed.
2. Cases taking credit for pressurizer spray and power operated relief valves to reduce coolant pressure are analyzed as well as those for which such credit is not taken. Safety valves are operable.

3. Credit is taken only for the safety valves in limiting secondary pressure.
4. No credit is taken for auxiliary feedwater flow during the event.
5. No credit is taken for the direct reactor trip on turbine trip.

The results of the analyses show that in each case the DNBR remains well above the design DNBR limit and the coolant system and steam generators are protected against over pressure by their respective safety valves. We find the analysis of the Turbine Trip event to be acceptable.

5.2 Loss of Non-Emergency AC Power to Plant Auxiliaries

Loss of non-emergency power may result in loss of power to plant auxiliaries, i.e., reactor coolant pumps, condensate pumps, etc. The transient is more severe than the turbine trip event because the decrease in heat removal by the secondary system is accompanied by a coolant flow coastdown which further reduces the capacity of the primary coolant to remove heat from the core.

The approved LOFTRAN code is used to perform the analysis. Conservative input assumptions are used including operation at 102 percent power, low value of average coolant temperature, conservative residual heat, no credit for reactor trip on loss of power, operation of pressurizer spray and power operated relief valves and secondary steam relief through the safety valves.

The results of the analysis show that DNBR remains above the design DNBR limit, that auxiliary feedwater capacity is sufficient to prevent water relief through the pressurizer relief and safety valves, and that natural circulation flow is sufficient to remove the residual heat from the fuel. We conclude that this analysis is acceptable.

5.2.3 Loss of Normal Feedwater Flow

A loss of normal feedwater may occur due to a pump failure, valve failures or loss of offsite AC power. The limiting event is that of total loss of normal feedwater. An analysis of this event is performed to show that fuel thermal design limits are met and that the auxiliary feedwater system is capable of removing the stored and residual heat and thus of returning the plant to a safe condition.

The approved LOFTRAN code is used as well as conservative input assumptions including prior operation at 102 percent design power, conservative decay heat, late reactor trip and initiation of auxiliary feedwater flow, worst single failure in auxiliary feedwater system, operability of pressurizer sprays and PORVs, and failure of the steam generator PORVs and relief valves.

Results of the analyses show that DNB is not approached during the transient and that the auxiliary feedwater system is capable of removing the stored and decay heat from the fuel. We conclude that the evaluation of the loss of normal feedwater event is acceptable.

5.2.4 Feedwater System Pipe Break

The Feedwater System Pipe Break is treated as a design basis (Condition IV) event. Analyses are performed to demonstrate that the reactor coolant pressure will remain below 110% of design value, that the core remains coolable and that resultant doses remain below acceptable limits.

The analyses were done with the approved LOFTRAN code for cases both with and without loss of offsite power. Conservative assumptions are made with respect to plant operating power, decay heat, initial values of reactor coolant temperature and pressure, pressurizer water level, operation of the protection system and ECCS equipment, and break size and location. The results of the analysis show that the core remains covered, that the hot leg temperature does not reach saturation, and that the Auxiliary Feedwater System provides sufficient cooling to remove decay heat. The radioactivity doses are bounded by those of the steamline break. We conclude that the analysis for this event is acceptable.

5.3 Decrease in Reactor Coolant System Flow Rate

Events in this category include the following:

1. Partial Loss of Forced Reactor Coolant Flow
2. Complete Loss of Forced Reactor Coolant Flow
3. Reactor Coolant Pump Shaft Seizure (Locked Rotor)
4. Reactor Coolant Pump Shaft Break

The first of these events is an anticipated transient (Condition II), the second is an unanticipated occurrence (Condition III), and the final two are design basis (Condition IV) events.

5.3.1 Partial Loss of Forced Reactor Coolant

The loss of two pumps with four loops in operation is analyzed for this event. Three codes which have been previously used by the licensee and accepted by the staff are used in the analysis - LOFTRAN, FACTRAN and THINC. LOFTRAN is used to obtain power and flow conditions during the transient, FACTRAN is used to obtain the heat flux as function of time and THINC is used to obtain DNBR as a function of time. The Improved Thermal Design Procedure is used and conservative reactivity coefficients are supplied as input.

The results of the analysis show that DNBR does not decrease below the design DNBR limit at any time during the transient. The applicable criterion for this event is thus met and we conclude that the analysis is acceptable.

5.3.2 Complete Loss of Forced Reactor Coolant Flow

This event is analyzed in the same manner as that described in Section 5.3.1 above except that loss of all pumps is assumed and trip occurs on loss of pump power instead of low core flow.

The results show that DNBR does not fall below the safety analysis value. Thus the criterion for a Condition II event is met which is acceptable for this event.

5.3.3 Locked Rotor

The coolant pump shaft seizure (Locked Rotor) is treated as a design basis event (Condition IV). Analyses are performed to show that the core remains in a coolable condition and that appropriate limits on offsite radiation doses are met. The analysis is performed with two codes - LOFTRAN, with which the power and flow transients are calculated and FACTRAN, with which the thermal behavior of the fuel is calculated.

Conservative assumptions made in the calculations include operation at 102 percent of Thermal Design Power, maximum coolant pressure and temperature, failure of pressurizer spray and power operated relief valves, and onset of DNB at initiation of the event. The effect of the zirconium-steam reaction is also included.

The results of the analysis show a maximum pressure of less than 110 percent of the design value, a maximum clad temperature at the hot spot of less than 2000 degrees Fahrenheit and zirconium water reaction at the core hot spot of 0.3 weight percent. We thus conclude that the core will remain in a coolable condition following the event. The offsite radiation dose is discussed in Section 5.7.3 below and assumes that cladding failure occurs for all fuel rods with DNB less than the safety limit.

5.4 Reactivity and Power Distribution Anomalies

This category includes the following events:

1. Uncontrolled Rod Cluster Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition
2. Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power
3. Rod Cluster Control Assembly Misoperation
4. Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature
5. A Malfunction or Failure of the Flow Controller in a BWR Recirculation Loop that Results in an Increased Reactor Coolant Flow Rate (not applicable to Callaway).
6. Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant.
7. Spectrum of Rod Cluster Control Assembly Ejection Accidents.

Of these the seventh is a design basis (Condition IV) event, and the third contains both Condition II and Condition III events. The rest are Condition II events.

5.4.1 Uncontrolled Rod Bank Withdrawal

Uncontrolled Rod Withdrawal events are analyzed for both startup conditions and operation at power. For these events the amount of reactivity which may be inserted and the rate of insertion is limited by the permitted rod insertions as a function of power level.

The startup event is analyzed by the TWINKLE, FACTRAN and THINC codes. TWINKLE, a spatial neutron kinetics code, is used to obtain the core power as a function of time, FACTRAN provides the fuel rod temperature transient and THINC is used for the transient DNBR calculation. Conservative input assumptions, including maximum reactivity insertion rate, minimum reactivity feedback and bounding values of axial and radial power shapes were used. The results of the calculations show that the safety analysis value of DNBR is not violated. We conclude that the analysis of the rod bank withdrawal event at startup conditions is acceptable.

The analysis of the event at power operating condition was performed with the LOFTRAN code. The Improved Thermal Design Procedure was used. Analyses were done as a function of both power level and reactivity insertion rate. Protection is provided by the combination of the high neutron flux trip and the overtemperature - delta T trip. Conservative values of trip setpoints are assumed and both maximum and minimum reactivity feedback cases are analyzed. The results of the calculations show that in no case does the DNB fall below the design DNBR limit. We conclude that the analyses of the rod bank withdrawal events are acceptable.

5.4.2 Rod Misoperation Events

These events include misalignment of a rod or rods in a bank, the dropped rod, dropped rod bank, and the accidental withdrawal of a single rod (as opposed to a bank withdrawal). The last of these is a Condition III event while the others are Condition II.

The limiting static misalignment events - a single rod at bottom with the rest of the bank withdrawn and the reverse situation have been analyzed with the standard Westinghouse nuclear design codes TURTLE and LEOPARD. In neither case is the DNBR criterion violated when the core is at full power. We conclude that this analysis is acceptable.

For a dropped rod bank the reactor is tripped by the negative flux rate trip and DNBR rises from its initial value. For a dropped single rod when operating in manual mode the core power reaches a stable value below full power and the reduction in power offsets the increase in radial peaking. Thus the core DNBR value is not decreased. In automatic mode the controller withdraws the other rods to increase core power and a power overshoot may result. The limiting case has been analyzed and the results show that DNBR does not fall below the design DNBR limit. We conclude that the analyses of rod bank and single rod drop events are acceptable.

The withdrawal of a single rod from the core requires multiple equipment failures or multiple operator errors. Thus this event is classified as Condition III. This classification has been previously approved for this event and is acceptable. The Condition III classification permits a limited amount of fuel failure. The power distributions in the core are calculated by the standard Westinghouse core parameter computer codes. The THINC code is then used to obtain the resultant DNBR values.

The calculation was performed with minimum reactivity feedback and resulted in the conclusion that the bounding value of failed fuel is 5 percent of the rods in the core. We find the analysis of this event to be acceptable.

5.4.3 Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature

The inadvertent startup of an idle reactor coolant pump can, under certain conditions, result in the injection of water colder than the reactor coolant into the core. This would cause an increase in power and a reduction in DNBR. This event has been analyzed with the LOFTRAN-FACTRAN-THINC code combination previously described. Conservative input assumptions included adverse reactivity feedback and conservative trip setpoints in the protection system. The results of the analysis show that DNBR remains above the safety analysis value during the transient. We conclude that the analysis of this event is acceptable.

5.4.4 Boron Dilution Events

A decrease in the boron concentration in the core may occur if an operator error or equipment malfunction results in pumping unborated water into the core. Such events are classified as Condition II events. Analyses were performed for dilution during refueling, cold shutdown, hot shutdown, hot standby, start-up, and power operation.

During refueling the introduction of non-borated water into the core is precluded by locking the relevant valves in the closed position. The only available sources of water contain borated water. In the cold shutdown mode the increase in the source range monitor response is detected by the nuclear instrumentation and an alarm is sounded. The valves through which the clean water is being inserted are automatically closed and valves which initiate boration are opened. This stops the dilution before criticality is reached. In hot shutdown and hot standby the same instrumentation stops the dilution.

In the startup and power operation modes the shutdown and regulating rods are withdrawn. In the event of an inadvertent dilution the power will rise to the trip setpoint and the reactor will be shut down. The operator then has adequate time (20 to 40 minutes) to take action to prevent return to criticality.

Thus in all modes of operation the reactor is protected against damage due to the inadvertent dilution of the boron concentration in the core. We conclude that the analysis of this event is acceptable.

5.4.5 Rod Ejection Accident

This is a design basis (Condition IV) event and is hypothesized to occur in order to investigate the effects of the very rapid insertion of a significant amount of reactivity. The mechanical failure of the control rod mechanism pressure housing is assumed, resulting in the complete ejection of the control rod from the core in approximately 0.1 seconds. The analysis of this event was performed by the same methods and techniques which were found to be acceptable in the FSAR. Analyses were performed at zero power and at full power at both beginning and end of cycle. Conservative assumptions on reactivity feedback and power distributions were made. The results show that in no case did the peak fuel enthalpy exceed our acceptance criterion of 280 calories per gram. The pressure surge from the event was mild and did not exceed our criterion for this event. Less than 10 percent of the fuel in the hot pellet was melted as a result of the event. We conclude that the analysis of this event is acceptable.

5.5 Increase in Reactor Coolant Inventory

There are two events in this category:

1. Inadvertent Operation of the ECCS During Power Operation
2. Chemical and Volume Control System (CVCS) Malfunction that increases reactor coolant inventory.

The events are considered to be Condition II events.

5.5.1 Inadvertent Operation of the ECCS

Inadvertent operation of the Emergency Core Cooling System may occur through operator error or through equipment failure. The effect is to inject borated water having a boron concentration of 2000 ppm into the core. This has the effect of reducing the reactor power and creating a mismatch between the core and turbine. As a result the coolant decreases in temperature and shrinks. The reactor may trip on the spurious safety injection signal or on low pressurizer pressure. A turbine trip will follow and the coolant temperature will rise due to decay heat. The DNBR value increases during the transient and at no time does the pressurizer empty. We conclude that the analysis of this event is acceptable.

5.5.2 CVCS Malfunction

Increases in coolant inventory caused by the CVCS malfunction may occur due to operator error or equipment failure. In this case the injected water has the same temperature and boron concentration as that in the core and no power change or change in DNBR occurs. The effect of the malfunction is simply to initiate filling of the pressurizer. If the failure of the level trip in the pressurizer is postulated reactor trip will not occur. Alarms will however alert the operator to the situation.

Analyses have been performed for four cases: minimum and maximum reactivity feedback, each with and without automatic pressurizer spray. In each case the operator has more than 30 minutes from the receipt of the first alarm until the pressurizer fills. We conclude that this is sufficient time to permit diagnosis and correction of the error and thus the analysis is acceptable.

5.6 Decrease in Reactor Coolant Inventory

The events in this category include:

1. Inadvertent opening of a pressurizer safety or relief valve.
2. Break in instrument line or other lines
3. Steam generator tube rupture (SGTR)
4. Loss of coolant accident (LOCA)

The last two of these events is a design basis (Condition IV) accident. The rest are Condition II events.

5.6.1 Inadvertent Opening of a Pressurizer Safety or Relief Valve

The most severe event in this category is the accidental opening of a pressurizer safety valve since it has approximately twice the steam flow rate of a relief valve. The result is a rapidly decreasing reactor pressure leading to a reduction in reactor power due to the positive moderator density coefficient of reactivity. The event is terminated by the over temperature delta T or a pressurizer low pressure trip.

The event is analyzed with the LOFTRAN code and the Improved Thermal Design Procedure is used. Conservatism in the analysis includes use of most conservative reactivity feedback coefficients, neglect of void effects, and operation of the automatic rod control system. The results show that DNBR remains above the safety analysis value throughout the transient. This satisfies the criterion for the event and is acceptable.

5.6.2 Break in Instrument Line

The FSAR analysis is still valid for this event. Radiological consequences are discussed in Section 5.7 below.

5.6.3 Steam Generator Tube Rupture

The licensee provided an analysis of the Steam Generator Tube Rupture (SGTR) accident in their submittal of December 1985. Although the SGTR issue is not yet fully resolved, we have determined that there is sufficient assurance that the Callaway plant can operate safely for the next fuel cycle for the following reasons: (1) all components necessary for mitigation of the design basis SGTR are safety related; (2) the Callaway plant steam lines and supports are designed for the resulting loads if the steam lines are filled with water; and (3) there is a low probability of a SGTR approaching the severity of the design basis event during the next cycle of operation.

5.6.4 LOCA

The licensee evaluated the consequences of both large and small break loss of coolant accidents. These analyses were performed at the stretch power level which is approximately 4.5% greater than the licensed power level of 3411 MWT.

The large break LOCA calculations for Cycle 2 utilize the approved 1981 Westinghouse model which was modified to include the BART computer code for calculation of core heat transfer during reflood. Use of the BART code has been approved by the NRC staff. In the FSAR the highest cladding temperature was calculated for a double ended cold leg rupture and was determined to be 2174.2°F which is less than the acceptance criterion of 2200°F. The FSAR calculation was performed utilizing the 1978 Westinghouse model which had been superseded.

License condition 14 requires that following the first refueling outage the licensee shall submit the worst large break LOCA using the 1981 Westinghouse model. The option of using the BART code for core reflood heat transfer evaluation was included. Using the 1981 evaluation model with BART the peak cladding temperature was calculated to be 2153°F for a double ended cold leg break. The reduced cladding temperature in the revised calculation results from more realistic heat transfer modeling. BART utilizes local fluid conditions to calculate the hot channel heat transfer coefficients. The previous model utilized empirical correlations using inlet conditions and data from the FLECHT reflooding experiments.

The licensee recalculated the consequences of a spectrum of small break LOCAs using the NOTRUMP computer code which has been approved by the staff. The NOTRUMP code was developed in response to staff requirements described in Section II.K.3.30 of the TMI Action Plan (NUREG-0737).

Small break LOCA analysis methods were required to be developed which would be in compliance with Appendix K to 10 CFR 50 and which would conservatively predict trends in data from recent test loop experiments. Licensees were required to submit small break LOCA analyses using the new model under Item II.K.3.31 of the action plan.

Union Electric Company submitted small break LOCA analyses for a spectrum of postulated small break LOCA events for the stretch power level. The highest peak cladding temperature was determined for a 3 inch equivalent diameter break in a cold leg (1299°F). Larger breaks resulted in lower calculated temperatures and smaller breaks were determined not to result in core uncovery. The limiting small break LOCA analysis currently in the FSAR was performed for the licensed power level of 3411 MWT and resulted in a peak cladding temperature of 1790°F. The WFLASH code was used for this analysis. Even though the initial power level was increased, a lower cladding temperature was calculated using NOTRUMP. This is the result of models in NOTRUMP allowing draining of the hot legs into the core and improved modeling of the cold leg loop seals which reduce the extent and duration of core uncovery.

The staff concludes that license condition 14 requiring reanalysis of the worst large break LOCA is met. In addition, Callaway conforms to the requirements of TMI Action Item II.K.3.31 for a plant specific analysis of a small break LOCA. The analyses were performed at a power level 4.5% greater than that required for the current licensed power level. The results therefore indicate adequate margin for meeting the criteria of 10 CFR 50.46.

5.7 Radiological Consequences

The use of OFA fuel has a negligible impact on the source term presented in the FSAR. The use of the Improved Thermal Design Procedure results in a reduction, in some cases, of the amount of failed fuel. For most of the accidents evaluated the conclusions in the FSAR are not changed. The exceptions are discussed below.

5.7.1 Steam Line Break

The analysis of this event shows a slight reduction in steam releases compared to the FSAR values. This results in a slight (1-3 percent) reduction in doses and is acceptable.

5.7.2 Loss of AC Power to Plant Auxiliaries

As a result of the reanalysis the steam releases during the first two hours are reduced but those during the next six hours are slightly increased. Corresponding changes occur in the thyroid dose rates but the results remain well within 10 CFR 100 limits and are acceptable.

5.7.3 Locked Rotor

As a result of the use of the Improved Thermal Design Procedure, the amount of fuel which suffers DNB is reduced from that which was calculated in the FSAR. This results in a reduction (by about 25 percent) in the resultant doses. This is acceptable.

6. TECHNICAL SPECIFICATIONS

Changes in the Technical Specifications are required in order to account for the introduction of the OFA fuel, the use of the Improved Thermal Design Procedure (ITDP), the change in the $F^N_{\Delta H}$ multiplier, and the introduction of the concept of the Design Thermal Power. Each of the changes is discussed below.

<u>Definition 1.10 DESIGN THERMAL POWER</u>

This definition was added in order to permit reference to this quantity in the Technical Specification. This is acceptable.

<u>Definitions 1.11 to 1.41</u>

These definitions were renumbered to account for the insertion of Definition 1.10. This is an editorial change and is acceptable.

<u>Figure 2.1-1 REACTOR CORE SAFETY LIMITS</u>

This figure was revised to reflect the use of the ITDP and the Design Thermal Power. These are consistent with the values used in the safety analyses and are acceptable.

Table 2.2-1 REACTOR TRIP SYSTEM SETPOINTS

Changes in this table include use of minimum measured flow instead of design flow and revisions to the Overpower delta T and Overtemperature delta T trip setpoints. These changes are required to account for the use of the ITDP and the WRB-1 DNB correlation. The setpoints were derived using standard Westinghouse methods and are acceptable.

Bases 2.1.1

The bases are altered to be consistent with the altered Technical Specifications and are acceptable.

Specification 3.1.3.3 ROD DROP TIME

The rod drop time has been increased to 2.4 seconds to account for the presence of the OFA fuel. This is consistent with the standard value used for OFA fuel and is acceptable.

Figure 3.1-1 ROD BANK INSERTION LIMITS

The change from "Relative Thermal Power" to "Rated Thermal Power" is for clarification and is acceptable.

Specification 3.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The multiplier in the $F^N_{\Delta H}$ algorithm has been changed from 0.2 to 0.3. Since this change has been accounted for in the safety analyses, including the safety system setpoints, we find the change acceptable.

The curve of flow as a function of R has been deleted and the core flow requirements have been transferred to Specification 3.2.5. This is done to simplify the Technical Specification and to account for the revised handling of instrument uncertainties in the ITDP. The content of the Specification has not been changed. We find this change acceptable.

Specification 3.2.5

See discussion under Specification 3.2.3 above.

Table 3.2.1

The maximum indicated reactor coolant system average temperature was increased to account for the use of Design Thermal Power and the indicated flow value was added to the table as described above. These changes are acceptable.

Specification 4.10.2.2

The reference to Specification 4.2.3.2 was changed to Specification 4.3.2.1 to account for a numbering change. This is acceptable.

Bases

The bases of the various specifications have been altered to make them consistent with the revised Specifications. This is acceptable.

7. CONCLUSIONS

We conclude that the licensee may reload and operate the Callaway Plant, Unit 1 for Cycle 2, at the rated power of 3411 thermal megawatts, without undue hazard to the health and safety of the public. This conclusion is based on the following considerations.

1. The use of 17x17 OFA fuel, Wet Annular Burnable Poison Rods, the Improved Thermal Design and the WRB-1 DNB correlation have been generically approved for use in Westinghouse reactors. The licensee has provided the required plant specific information to support their use.

2. The methods used for the safety analyses are the same as those which were used and approved for the FSAR analyses or have been subsequently approved for use.

3. Conservative input assumptions have been used in the safety analyses.

4. The results meet the applicable acceptance criteria.

With respect to the operation at higher than current rated power, the staff may require additional information in some areas to justify operation at higher power.

8.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant change in the types or significant increase in the amounts of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

9.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (51 FR 6831) on February 16, 1986, and consulted with the state of Missouri. No public comments were received, and the state of Missouri did not have any comments.

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

10. REFERENCES

1. Letter from Donald F. Schnell, Union Electric Company, to H. R. Denton, NRC, dated November 15, 1985 (has six attachments).

2. Letter from Robert J. Schukai to H. R. Denton, NRC dated February 18, 1986.

3. Davidson, S. L.; Iorii, J. A., (Eds.), "Verification Testing and Analyses of the 17x17 Optimized Fuel Assembly," WCAP-9401-P-A and WCAP-9402-A, August 1981.

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5. Motley, F. E.; Hill, K. W., et al., "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," WCAP-8762-P-A and WCAP-8763-A, July 1984.

6. Letter from Donald F. Schnell, Union Electric Company, to H. R. Denton, NRC dated December 13, 1985.

7. Davidson, S. L.; Iorii, J. A., "Reference Core Report - 17x17 Optimized Fuel Assembly," WCAP-9500-A, May 1982.

8. Skaritka, J., (Ed.), "Fuel Rod Bow Evaluation," WCAP-8691, Revision 1 (Proprietary), July 1979.

9. "Partial Response to Request Number-1 for Additional Information on WCAP-8691, Revision 1" Letter, E. P. Rahe, Jr., (Westinghouse) to J. R. Miller (NRC), NS-EPR-2515, dated October 9, 1981; "Remaining Response to Request Number 1 for Additional Information on WCAP-8691, Revision 1," Letter, E. P. Rahe, Jr. (Westinghouse) to J. R. Miller (NRC) NS-EPR-2572, dated March 1[illegible], 1982.

10. J. Skaritka, "Westinghouse Wet Annular Burnable Absorber Evaluation Report," WCAP-10021, February 1982.

11. Letter from Donald F. Schnell, Union Electric Company, to H. R. Denton, NRC, dated February 24, 1986.

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