Florida Power
CORPORATION
Crystal River Unit 3
Docket No. 50-302

September 30, 1994
3F0994-08

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

Reference: 1. NRC to FPC letter, 3N0573-04, dated May 23, 1973
2. NRC to FPC letter, 3N0994-06, dated September 2, 1994

Subject: Technical Specification Change Request No. 199, Revision 0
Rated Thermal Power Level Increase to 2568 MWt

Dear Sir:

Florida Power Corporation (FPC) is submitting Technical Specification Change Request No. 199, Revision 0 requesting amendment to Appendix A of Operating License No. DPR-72 to allow Crystal River Unit 3 (CR-3) to operate with a maximum rated thermal power of 2568 MWt, an increase of 24 MWt over the current licensed power of 2544 MWt. The B&W 177 Fuel Assembly (FA) Nuclear Steam Supply System (NSSS) is capable of operating at a thermal power level of 2772 MWt. However, FPC is only requesting authorization to operate at 2568 MWt thermal power because of limitations in the secondary area of the plant. These secondary plant limitations will not affect plant operation at 2568 MWt. FPC has performed a detailed engineering study on this power increase. Some of the results were presented at the August 31, 1994 NRC/FPC meeting (summarized in Reference 2) and are described in more detail herein. FPC has studied the economic benefits of this increase and the study shows that the total savings are approximately $1.5 million per year to our customers.

Since a B&W 177 FA NSSS is capable of operating at 2772 MWt, many of the B&W licensing topical reports, design documents and equipment performance specifications were developed based on operation at this thermal power. Even though FPC did not plan to operate CR-3 at this power level initially, many of these documents were used as the basis for the CR-3 design and licensing. Consequently, during CR-3's original licensing application, NRC review and licensing actions were based on a majority of the Final Safety Analysis Report

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(FSAR) Chapter 14 accident analyses being performed at a thermal power level of 2568 MWt or greater. Those Chapter 14 accidents which were originally evaluated at a lower thermal power have now been evaluated at 2568 MWt and found acceptable. Attachment 1 to this letter provides FPC's evaluation of the proposed change as it affects plant systems, radiological and non-radiological impacts, and environmental effects discussed in the Final Environmental Statement (Reference 1). Pursuant to 10 CFR 50.91(a)(1), FPC is providing in Attachment 1 our analysis of significant hazards consideration, which was performed using the standards in 10 CFR 50.92. Also included in this letter as Attachment 2 are the proposed replacement pages for Appendix A to CR-3's Technical Specifications.

This submittal also includes information that the NRC requested in the August 31, 1994 meeting be provided. Information concerning the scope and depth of system evaluations, offsite and main control room doses, overpressure protection, boron precipitation, reactor vessel material impacts, and environmental impacts were expanded or added based upon the NRC staff's feedback.

Based upon the evaluation results, FPC has concluded that operation of CR-3 at 2568 MWt will not affect the public health and safety. CR-3 will continue to meet the regulatory requirements and standards which form the basis for its operating license. FPC requests timely approval of this proposal so that it can become effective during Cycle 10 which began on June 3, 1994.

Sincerely,

P. M. Beard, Jr.
Senior Vice President
Nuclear Operations

PMB/JWT

Attachments

xc: Regional Administrator, Region II
Senior Resident Inspector
NRR Project Manager

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

IN THE MATTER	)	
	)	DOCKET NO. 50-302
FLORIDA POWER CORPORATION	)	

CERTIFICATE OF SERVICE

P. M. Beard, Jr. deposes and says that the following has been served on the Designated State Representative and Chief Executive of Citrus County, Florida, by deposit in the United States mail, addressed as follows:

Chairman,
Board of County Commissioners
of Citrus County
Citrus County Courthouse
Inverness, FL 34450

Administrator,
Radiological Health Services
Department of Health and
Rehabilitative Services
1323 Winewood Blvd.
Tallahassee, FL 32301

A copy of Technical Specification Change Request No. 199, Revision 0.

FLORIDA POWER CORPORATION

P.M. Beard, Jr.
Senior Vice President
Nuclear Operations

SWORN TO AND SUBSCRIBED BEFORE ME THIS 30th DAY OF SEPTEMBER 1994

JOAN E. MOUNTS
Notary Public (print)

Notary Public (signature)

Notary Public, State of Florida at Large
My Commission Expires:
Notary Public, State of Florida
My Commission Expires Oct. 17, 1994
Bonded Thru Troy Fain - Insurance Inc.

**FLORIDA POWER CORPORATION
CRYSTAL RIVER UNIT 3
DOCKET NO. 50-302/LICENSE NO. DIR-72
REQUEST NO. 199, REVISION 0
RATED THERMAL POWER INCREASE**

LICENSE DOCUMENT INVOLVED: Technical Specifications

PORTIONS: Specifications 1.1

DESCRIPTION OF REQUEST:

This change request proposes to allow an increase in the rated thermal power for CR-3 from 2544 MWt to 2568 MWt. Table 1 provides a comparison of key plant parameters at the new power level to those at the current 2544 MWt limit.

REASON FOR REQUEST:

The B&W 177 Fuel Assembly nuclear plant has been designed for operation at an ultimate power level of 2772 MWt. In order to make available to our customers the increased potential of CR-3, FPC requests an increase in the licensed power level for CR-3 to 2568 MWt. Cycle 10 for Crystal River Unit 3 has been analyzed to allow a core thermal power of 2568 MWt. The limits provided in the current CR-3 Core Operating Limits Report (COLR) reflect operation at the higher power level. Plant system capability and safety analyses were reviewed, as were relevant operational and environmental considerations, to assure nuclear safety and environmental protection would be maintained.

EVALUATION OF REQUEST:

Evaluation Process

The power upgrade evaluation was a three step process which includes a feasibility study, a detailed engineering evaluation, and plant implementation. The first two steps are complete. The third step is awaiting the approval of this request.

FPC began the power upgrade process several years ago with a feasibility study which evaluated the overall benefit of performing various power level upgrades in terms of the gain in electrical power output and cost. Current and expected plant operation were examined to confirm that appropriate design, operating, and safety criteria could be met. Engineering evaluations were performed to support preliminary decisions concerning operating plant conditions at the increased power and to identify potential major hardware modifications. At the completion of this phase, FPC decided that increasing the power to 2568 MWt was feasible and cost beneficial at this time.

Following the completion of the feasibility study, detailed engineering evaluations were performed. These engineering evaluations provided a detailed analysis and assessment of potentially affected areas of plant operation at 2568 MWt. In the detailed phase, FPC evaluated the CR-3 Nuclear Steam System, Balance of Plant Systems, and Reactor Building structure to assure their adequacy at the increased power level. Included in these evaluations are systems and components that are not power level related or directly affected by an increase in power level, as well as, associated issues such as environmental considerations. Equipment performance and plant operation were evaluated with respect to actual performance versus projected operating conditions to identify any hardware modifications required to achieve the upgraded power. For CR-3, there are no hardware modifications necessary.

Plant features, such as systems and components, that are important for safety received an in-depth review of possible effects that operation at an increased power level could have on that system or component. The plant features listed in Table 2 were further evaluated to determine if the components could perform their function at 2568 MWt in accordance with applicable licensing basis requirements and criteria. Each plant feature was evaluated considering the operating conditions at the slightly higher power, including system setpoints and response characteristics. Evaluations consisted of comparisons of the conditions expected at the upgraded power to the current system design bases. The ability of a system or component to perform its safety function(s) in accordance with applicable regulatory requirements was confirmed by design engineers, verified by the appropriate system engineer, then documented in an engineering report.

As an example of the depth of these engineering reviews, the adequacy of the Reactor Building Spray (BS) System to perform at the higher power level considered items such as:

1. The basis for the current peak pressure in the Reactor Building and whether it would change at the higher power.

2. How the pressure change could effect BS System operation.

3. What impact would an increase in the assumed fission product release following the LOCA event and the effect of the BS System response have on the Offsite Dose and the Control Room Habitability.

4. Whether an increased flow rate was required for the BS System.

5. What impact the increased flow rate would have on the Borated Water Storage Tank inventory, emergency diesel generator load, Reactor Building flooding, recirculation fluid (ECCS sump) pH, etc.

Table 3 lists those plant features that are not considered important for safety which were also evaluated. These are primarily Balance of Plant (BOP) systems. For the BOP systems, a review of possible effects was performed by the lead project engineer, and any systems which could be affected were identified to the appropriate system engineers for their review and evaluation. The evaluations

consisted primarily of a comparison of system and component design bases to the expected conditions at the upgraded power. For example, the Main Feedwater (FW) System flow is expected to increase from 10.71 x 10^6 lbm/hr to 10.84 x 10^6 lbm/hr. The FW System evaluation compared the upgraded plant conditions for FW system piping as well as significant components, such as heaters and pumps, to the design bases for these items. The results of these BOP evaluations were also documented in an engineering report.

The detailed engineering evaluation concluded that CR-3 systems will perform within their design with reactor power at 2568 MWt and nuclear safety will be maintained. No physical modification to any system will be required as a result of the power level upgrade. Some recalibration of existing plant instrumentation will be required.

Transient and Safety Analyses

The design basis event analyses listed in Table 4 were evaluated to confirm their applicability for a rated thermal power of 2568 MWt or higher. Most design basis events were originally analyzed for a 2568 MWt power level, therefore, the analytical conclusions presented in the CR-3 FSAR remain bounding and these events will not increase the risk to public health and safety.

Since the moderator dilution accident, letdown line failure accident, loss of feedwater event, and SBLOCA HPI Line Break were originally analyzed at 2544 MWt, FPC reviewed the relevant accident parameters for changes due to the higher thermal power. The results showed that (1) the consequences of the evaluated accidents are not increased, and (2) the radiological consequences remain well below 10 CFR 100 guidelines.

Other Related Analysis

FPC's evaluation confirmed that the CR-3 reactor coolant system overpressure protection described in FSAR Section 4.2.4 remains bounding for 2568 MWt. The basis for FSAR Section 4.2.4, "Pressure Control and Protection," is described in B&W Topical Report BAW-10043, "Overpressure Protection for B&W's Pressurized Water Reactors." The analysis documented in this report was performed at 2772 MWt. The proposed power increase also will not impact the secondary system overpressure protection provided by the main steam safety valves.

The impact on reactor vessel materials due to the increased power level is judged to be insignificant. This conclusion is based upon B&W Owners Group (BWOG) efforts which have examined the effects of fluence levels on reactor vessel materials. CR-3 has a very low leakage core design which has reduced the fluence level significantly below the values considered in the original reactor vessel design. The slightly increased fluence levels due to the power upgrade will remain well within the original design limits, and will addressed in the ongoing Reactor Vessel Material Program efforts (PTLR, updated PTS evaluation, etc.).

Boron precipitation is not a concern at CR-3 for operation at 2568 MWt or higher. The BWOG has evaluated this issue for all B&W plants at a power level of 2772 MWt. FPC was a participant in this analysis program and the BWOG results bound CR-3.

FPC confirms that the conclusions provided in responses to the Station Blackout rule remain bounding for the increase in thermal power level. CR-3 will retain its capability to cope with a station blackout for at least 4 hours.

FPC confirms that the increased power level will not impact the environmental qualification (EQ) program. As noted in Table 1, the variables which impact the EQ program limits, such as fluid temperature, remain basically unchanged at the higher power level. Radiation environments would tend to increase proportionally because of increased source terms, however, FPC has established radiation margins in its EQ program which are in excess of the calculated radiation levels. Therefore, FPC concludes that present EQ program remains bounding for operation at 2568 MWt.

As noted in Table 4, there were four Design Basis Accidents (DBA) which were re-evaluated in support of the increased power level. Of these four DBAs, only the Letdown Line Rupture results in an increase in the offsite dose reported in FSAR Section 14.2.2.6. and FSAR Table 14-43. A comparison of the revised offsite dose resulting from this event with the present FSAR value is provided below:

	Letdown Line Rupture Accident			
	Thyroid, Rem		Whole Body, Rem	
	2568 MWt	2544 MWt	2568 MWt	2544MWt
Two (2) Hour Dose for EAB	1.33	1.15	0.077	0.067
30 Day Dose for LPZ	0.117	0.101	0.0068	0.0059

These revised doses remain well within 10 CFR 100 limits.

The existing Control Room dose analysis is also affected by the power level upgrade. The original analysis assumed the current power level of 2544 MWt. The new control room doses at 2568 MWt are:

	Post LOCA Control Room Doses, Rem	
	2544 MWt	2568 MWt
Thyroid	23.2	23.68
Whole body Gamma γ	1.88	1.899
Beta (β)	17.7	17.86

These doses are within the applicable regulatory limits of 5 Rem Whole Body, or its equivalent identified in Criterion 19 of 10 CFR 50, Appendix A.

Radiological Evaluation

Plant radiation protection features are designed to limit the radiation exposure to plant personnel and the general public to 10 CFR 20 limits. Calculated post-accident doses to the plant personnel and the general public are based upon 1 percent failed fuel rods. The integrated off-site exposures are within 10 CFR 100 limits. Changes in radiation sources due to the increased thermal power level may be generally estimated by a direct ratio of the power levels (i.e, a 1% power level equals a 1% increase in source term). While the radiological source terms will increase slightly as a result of the approximately 1% thermal power increase, these increases will have little or no impact on FPC's radiation protection/ALARA plans and programs. Likewise, since systems and procedures controlling normal radioactive releases are based upon limiting plant effluents to a small fraction of regulatory limits, CR-3 will not exceed 10 CFR 20 or 10 CFR 100 limits.

Final Environmental Statement Impacts

While 10 CFR 51 requires an environmental assessment (EA) or environmental impact statement (EIS) for any "major Federal action significantly affecting the quality of the human environment," it does allow the NRC discretion in evaluating the extent to which EAs or EISs are necessary. EAs or EISs are not required for any action included in the list of "categorical exclusions" set forth in 10 CFR 51.22(c). Specifically, 10 CFR 51.22(c)(9), provides that an EA is not required for the issuance of an amendment provided that:

(i) the amendment involves no significant hazards consideration,

(ii) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and

(iii) there is no significant increase in individual or cumulative occupational radiation exposure.

FPC believes that the provisions of 10 CFR 51.22(c)(9) are applicable to this power increase[1]. For the reasons described in this submittal, FPC believes that the three criteria of 10 CFR 51.22(c)(9) are satisfied. Therefore, this Technical Specification amendment should be considered under the "categorical exclusions" provisions of 10 CFR 51.22(c)(9). The environmental impacts at the higher power level are bounded by the impacts assumed in the existing Final Environmental Statement (FES) for CR-3. That FES was issued in May 1973 to FPC by Reference 1. Even if the NRC chooses to perform an EA, information provided in the FES, together with this submittal should assist the NRC in making a "finding of no significant impact" in accordance with 10 CFR 51.32.

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[1] 10 CFR 51.22(c)(9) has been applied by the NRC in the review of power level increases for other plants

The May 1973 Final Environmental Statement (FES) was based upon an operating power level of 2544 MWt. However, the conclusions of that report remain valid at 2568 MWt given the conservatisms in the FES. This is based upon an assessment of the assumptions used in the FES and subsequent operating experience for CR-3 through 1993, the last year that reports were submitted to the NRC.

The proposed thermal power increase to 2568 MWt is approximately 1% above the present licensed limit. This power level increase could result in an increase of 1% in total core fission product inventory. Therefore, with all other parameters (e.g. radwaste system efficiency, plant integrity) being equal, offsite doses are expected to increase no more than 1%. The assumptions used in the FES to project offsite doses postulate releases of radionuclides that are 10 times greater than those typically released from CR-3. Examples of actual releases are described in the Semiannual Radioactive Effluent Release Reports submitted by FPC in accordance with 10 CFR 50.36(a)(2). Consequently, a small increase in offsite releases, similar to that which may result from the power upgrade, continue to be bounded by the FES.

FPC has considered the incremental increase in occupational (on-site) exposure as part of its assessment of the proposed 1% power increase. While the 1973 FES did not address occupational exposure for CR-3, FPC is committed to assure that individual radiation doses and population doses be maintained as low as reasonably achievable (ALARA). Since CR-3 is operating with refueling outages every 2 years, FPC expects that operation in future years at 2568 MWt will show a decrease on an annualized basis in occupational exposure such that CR-3 will be below the 1995 INPO occupational dose goal of 185 man-rem for operating PWRs in the United States. Since most of the exposures are received during maintenance and refueling periods, and not while the reactor is operating, an increase of 1% in operating power level will have an insignificant effect on occupational exposures at CR-3, especially with FPC's Part 20 ALARA program.

FPC has also reviewed the environmental impacts attributable to the transportation of fuel and waste. With respect to the normal conditions of transport and possible accidents in transport, FPC concludes that the environmental impacts are bounded by those identified in Table S-4, "Environmental Impact of Transportation of Fuel and Waste To and From One Light Water-Cooled Nuclear Power Reactor" of 10 CFR 51.52. The bases for this conclusion are that:

1. New Fuel: The FES stated that each year 60 fuel assemblies would be shipped to CR-3. Since CR-3 is operating on a 24 month fuel cycle, fuel assemblies are not required to be shipped every year. In a refueling year, CR-3 may receive 60-80 new fuel assemblies. Since the radiation and heat emitted by new fuel are small and the proposed power level increase will not cause any significant increase in these, there will be essentially no effect on the environment during transport under normal conditions for the 60-80 fuel assemblies. Therefore, the FES estimates of the exposure to transportation workers and general public remain conservative.

2. <u>Spent Fuel</u>: Table S-4 is based on annual refueling and an assumption of 60 spent fuel shipments per reactor year. As stated in Item 1 above, CR-3 is operating on a 24 month fuel cycle which would, by itself, require fewer spent fuel shipments per reactor year; and, therefore, reduce the overall impacts related to population exposure and accidents discussed in Table S-4. FPC has not shipped any CR-3 irradiated fuel offsite and has no plans to do so in the near future. FPC also has no plans to return to annual refueling.

3. <u>Enrichment and Irradiation</u>: The NRC evaluated the impact of higher fuel enrichment and extended irradiation on the environment in its assessment entitled "NRC Assessment of Environmental Effects of Transportation Resulting from Extended Fuel Enrichment and Irradiation," dated July 7, 1988 and published in the <u>Federal Register</u> (53 FR 30355). The assessment indicated that the environmental cost contribution of increased fuel enrichment and irradiation limits are either unchanged or may in fact be reduced from those summarized in Table S-4 as set forth in 10 CFR 51.52(c). Consequently, operation of CR-3 at 2568 MWt will not impact the environment.

4. <u>Waste Transport</u>: The transportation of solid radioactive waste from CR-3 has remained with the assumptions of the FES. The FES assumed that 470 drums of solid radioactive waste consisting of spent resin filters and evaporator bottoms with an activity of 21 curies per drum would be generated and shipped per year. For the years 1989 through 1993, CR-3 shipped spent resins and evaporator bottoms in resin liners and high integrity containers rather than drums. For these years, the following approximate quantities of spent resins and evaporator bottoms drum equivalents were shipped:

Year	Drums(equivalent)	Average Curies/Drum(equival.)
1989	306	7.88
1990	257	0.32
1991	326	0.69
1992	743	0.29
1993	380	2.32

The FES also assumed that 1200 drums of dry active waste containing less than 5 curies would be shipped offsite per year. For the years 1989 through 1993, CR-3 shipped dry active wastes in drums and strong, tight Sealand containers. For these years, the approximate quantities of dry active wastes drum equivalents shipped were:

Year	Drums(equivalent)	Average Curies/Drum (Equival.)
1989	672	0.008
1990	612	0.032
1991	353	0.056
1992	310	0.009
1993	347	0.011

Historically, the solid waste shipments have been much less than the FES amount. FPC concludes that the overall impact on the environment due to transportation of wastes will not exceed the levels considered in the establishment of Table S-4.

Non-Radiological Evaluation

The non-radiological environmental concerns for the Crystal River Energy Complex that impact CR-3 are discharge canal water temperature and flow. In developing the site permit for Units 1, 2, and 3, FPC negotiated with the Environmental Protection Agency (EPA) to establish a set of effluent limitations and monitoring requirements which serve to protect the environment. The agreements established between FPC and the EPA for the National Pollutant Discharge Elimination System (NPDES) limits on the discharge canal are reflected in Permit No. FL0000159 which states:

(1) the combined condenser flow from Crystal River Units 1, 2, and 3 shall not exceed 1897.9 million gallons per day (MGD) during the period May 1st through October 31st of each year, nor 1613.2 MGD during the remainder of the year; and

(2) the discharge temperature at the site point of discharge shall not exceed 96.5°F for a period of more than three consecutive hours, or a daily maximum value of 97.0°F.

This NPDES permit was issued on September 30, 1993 by the EPA to assure that the environmental consequences of Units 1, 2, and 3 remain within acceptable limits. Operation of CR-3 in accordance with the NPDES permit limits assures that the provisions of the Non-Radiological Technical Specifications contained in Appendix B-Part II of the CR-3 Operating License are met. FPC has considered the non-radiological effects of operation at the higher power level as specified in

the NPDES permit. The only anticipated change is an increase in the CR-3 circulating water discharge temperature. It is expected to increase by 0.5°F while operating at 2568 MWt. This negligible increase in water temperature will have no impact on the NPDES limits. FPC expects that future improvements in cycle efficiency and turbine performance will reduce the discharge temperature further. Therefore, no changes are required in the Environmental Protection Plan because of the power level increase to 2568 MWt.

SHOLLY EVALUATION OF REQUEST:

Florida Power Corporation has reviewed the requirements of 10 CFR 50.92 as they relate to the proposed change in core power level and considers the proposed change not to involve a significant hazards consideration. In support of this conclusion, the following analysis is provided:

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated. The thermal-hydraulic and nuclear characteristics of the reactor core were originally designed for a rated thermal power of 2568 MWt or higher. Therefore, the proposed thermal power increase to the reference power level of 2568 MWt does not change the original design assumptions and analyses for the reactor core. Most of the design basis accidents and transients were originally evaluated at the proposed power level. As described more fully in this submittal, those transients and accidents that were not originally evaluated at 2568 MWt were re-evaluated using CR-3 FSAR Chapter 14 accident sequence of events, reactor protection criteria, and approved calculational methods. Based on this evaluation and initial plant design evaluations, FPC has determined that the probability and consequences of design basis transients and accidents are not significantly increased and that the radiological consequences from the design basis transients and accidents remain well below 10 CFR 100 limits.

   FPC has also reviewed CR-3 balance of plant and safety related systems to determine which systems and components could be affected by the proposed power increase. The changes to the reactor coolant system and secondary conditions and parameters are discussed in this submittal. These changes are minor in nature. The only Technical Specification change is to revise the reference power to 2568 MWt. No facility modifications will be required. FPC evaluated the systems and components and concluded that these systems and components will continue to perform within their design parameters with the unit operating at 2568 MWt.

   Based on the foregoing, the proposed amendment does not significantly increase the probability or consequences of an accident previously evaluated.

2. The proposed thermal power increase does not create the possibility of a new or different kind of accident from previously evaluated accidents. As noted above, the thermal-hydraulic and nuclear characteristics of the

reactor core were originally designed for operation at the proposed thermal power. Therefore, operation at the proposed power level does not introduce new or different performance characteristics that create the possibility of a new or different kind of accident.

FPC has also reviewed CR-3 safety-related systems and balance of plant systems to determine which systems could be affected by the proposed power increase and the resultant minor changes in plant parameters and operating conditions. Systems that could be affected were evaluated using the licensing basis criteria described in the CR-3 FSAR to assure their adequacy at the increased power level. Included in these evaluations were plant features that are not power level related or directly affected by an increase in power level, as well as, associated issues such as environmental considerations. Equipment performance and plant operation were evaluated with respect to actual performance versus projected operating conditions to identify any hardware modifications required to achieve the upgraded power. Based on this evaluation, FPC has determined that all systems will continue to perform within their design parameters at 2568 MWt and that no physical modifications to these systems will be necessary to accommodate a 2568 MWt rating. Only minor re-calibration of plant instrumentation to reflect the increase power will be needed. The proposed power level does not introduce any new performance characteristics or modes of operation for plant systems and components, and does not introduce any new failure modes.

Based on the foregoing, the proposed amendment does not create the possibility of a new or different kind of accident.

3. The proposed amendment does not involve a significant reduction in a margin of safety. The thermal-hydraulic and nuclear characteristics of the reactor core were originally designed for operation at the proposed power level. Most of the design basis transients and accidents were originally analyzed assuming a power level of 2568 MWt or higher. As described more fully in this submittal, those transients and accidents that were not originally analyzed at 2568 MWt were re-evaluated using CR-3 FSAR Chapter 14 accident sequence of events, reactor protection criteria, and approved calculational methods. FPC has determined that operation with the proposed thermal power will be bounded by the original analyses. In addition, FPC's evaluation of affected plant systems and components revealed that plant systems and components will continue to operate within their design parameters with no significant change in a margin of safety.

   Based on the foregoing, the proposed amendment does not involve a reduction in the margin of safety.

TABLE 1 RCS AND SECONDARY CONDITIONS FOR CR-3				
Parameter	Current Value	Predicted Value	Change at 2568 MWt	Percent Change (%)
Core Power	2544 MWt	2568 MWt	+24 MWt	0.934%
T_{HOT}	601.2°F	601.4°F	+ 0.2°F	0.03%
T_{COLD}	558.0°F	557.8°F	- 0.2°F	0.04%
m_{RCS}	149×10^6 lbm/hr	149×10^6 lbm/hr	None	0.0%
T_{STEAM}	594.7°F	594.5°F	- 0.2°F	0.03%
T_{FW}	456.1°F	457.0°F	+ 0.9°F	0.2%
m_{FW}	10,711,100 lbm/hr	10,839,325 lbm/hr	+128,225 lbm/hr	1.2%
$Level_{SGA}$	82.7%	83.6%	+ [illegible]	1.1%
$Level_{SGB}$	78.9%	79.9%	+1.0%	1.3%

TABLE 2

SAFETY-RELATED PLANT FEATURES EVALUATED FOR OPERATION AT 2568 MWt

NUCLEAR STEAM SUPPLY SYSTEM

Reactor Coolant System (RCS)
Decay Heat Removal (DH)
Makeup and Purification (MUP)

PRINCIPAL SAFETY SYSTEMS

High Pressure Injection (HPI)
Low Pressure Injection (LPI)
Core Flooding System (CF)
Emergency Feedwater (EF) and Dedicated Condensate Inventory
Reactor Building Spray System (BS)
Reactor Building Cooling System (AH-XB)

OTHER ITEMS IMPORTANT FOR SAFETY

Containment Integrity
Equipment Qualification
High Energy Line Break (HELB)
Control Room Habitability
Internal Plant Flooding
Radioactive Effluents
Water Chemistry

OTHER SYSTEMS IMPORTANT FOR SAFETY

Nuclear Services Closed Cycle Cooling System (SW)
Decay Heat Closed Cycle Cooling System (DC)
Nuclear Services and Decay Heat Sea Water System (RW)
Spent Fuel Cooling System (SF)

ELECTRICAL SYSTEMS

Diesel Generators (EG)
DC Power System (Station Batteries) (DP)

I&C SYSTEM REVIEW

Emergency Feedwater Initiation and Control System (EFIC)
Engineered Safeguards Actuation System (ES)
Nuclear Instrumentation (NI)
Reactor Protection System (RPS)

TABLE 3

NON-SAFETY RELATED PLANT FEATURES EVALUATED FOR OPERATION AT 2568 MWt

MECHANICAL SYSTEMS

Main Steam
Condensate
Feedwater
Extraction Steam
Circulating Water
Main Steam Reheaters
Main Generator
Heater Drains

I&C SYSTEMS

Integrated Control System
Non Nuclear Instrumentation
ATWS Systems (AMSAC/DSS)

ELECTRICAL SYSTEMS

Main Generator
Step-up and Auxiliary Transformers
500kV switchyard
Isolated Phase Bus Duct

TABLE 4

DESIGN BASIS SAFETY ANALYSES

Originally analyzed for 2568 MWt or higher:

Uncompensated Operating Reactivity Changes
Startup Accident
Rod Withdrawal at Rated Power Operation Accident
Cold Water Accident
Loss-of-Coolant Flow Accident
Stuck-Out, Stuck-In, or Dropped Control Rod Accident
Load Rejection Accident
Station Blackout Accident
Steam Line Failure Accident
Steam Generator Tube Rupture
Rod Ejection Accident
Loss-of-Coolant-Accident
Fuel Handling Accident
Main Feedwater Line Break Accident
Waste Gas Decay Tank Rupture Accident

Analyzed for increase to 2568 MWt:

Moderator Dilution Accident[2]
Letdown Line Failure
Loss of Feedwater Event
SBLOCA HPI Line Pinch Break Accident

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2 CR-3 has been modified to prevent dilution through the Decay Heat Removal System. Moderator dilution is possible only through the Makeup & Purification System.