ATTACHMENT 6

NON-PROPRIETARY VERSION OF THE PPL LICENSING TOPICAL REPORT NE-2000-001N, REVISION 1

PPL LICENSING TOPICAL REPORT

NE-2000-001N

Susquehanna Steam Electric Station

Units 1 and 2

Licensing Topical Report for

POWER UPRATE

RESULTING FROM INCREASED FEEDWATER

FLOW MEASUREMENT ACCURACY

September, 2000

Susquehanna Steam Electric Station
Units 1 and 2

Licensing Topical Report for

POWER UPRATE RESULTING FROM INCREASED

FEEDWATER MEASUREMENT ACCURACY

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NOTICE

This report was derived from information developed during PPL's nuclear design and licensing analysis activities and from safety and licensing information provided to PPL by Siemens Power Corporation, Bechtel Power Corporation, General Electric Company and Kraftwerk Union A.G. This report is being submitted by PPL to the United States Nuclear Regulatory Commission specifically in support of future proposed amendments to each of the Susquehanna Steam Electric Station Unit operating licenses to permit operation at increased core thermal power, which results from a reduction in uncertainty realized by replacing the feedwater flow elements used for feedwater flow measurement. The information in this report is true and correct to the best of PPL's knowledge, information and belief.

ABSTRACT

This licensing topical report supports a 1.4 percent increase in reactor operating thermal power, and an increased feedwater and steam flow of 1.4 percent, based on the installation of the Caldon Leading Edge Flow Meter (LEFM✓™). The LEFM✓™ measures feedwater flow more accurately than the currently installed venturi flow meters, therefore, the increased reactor operating power level is achieved without exceeding the power level currently used for design bases analysis.

Detailed evaluations of the reactor and engineered safety features; of power conversion, emergency power, support systems; of environmental issues; of design basis analyses; and of previous licensing evaluations were performed. The report demonstrates that both Units 1 and 2 of the Susquehanna Steam Electric Station will operate safely with up to a 1.4 percent increase in operating thermal power, a corresponding 1.4 percent increase in turbine inlet steam flow; the increased capacities of the supporting systems and components at these uprated conditions.

This report follows the NRC-approved generic format and content for BWR power uprate licensing reports developed by General Electric and reported in NEDC-31897-A, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate" (Licensing Topical Report 1, or LTR1). Review and approval of this report by the NRC will provide the basis for PPL license amendments supporting implementation of this power uprate.

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NOMENCLATURE

Term	Definition
ADS	Automatic Depressurization System
ALARA	As Low As Reasonably Achievable (radiation dose)
AC	Alternating Current
ANSI	American National Standards Institute
APRM	Average Power Range Monitor
ARI	Alternate Rod Insertion
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transients Without Scram
ATWS RPT	Anticipated Transients Without Scram – Recirculation Pump Trip
BHP	Brake Horse Power
BIIT	Boron Injection Initiation Temperature
BOP	Balance of Plant
BTU	British Thermal Unit
BWR	Boiling Water Reactor
BWRSAR	Boiling Water Reactor Severe Accident Response (Computer code)
CC	Collapse Criteria or Criterion
CFR	Code of Federal Regulations
CFS	Condensate Filtration System
CO	Condensation Oscillation
COTTAP	COmpartment Transient Temperature Analysis Program (Computer code)
CPR	Critical Power Ratio
CRD	Control Rod Drive
CRDA	Control Rod Drive Accident
CS	Core Spray
CS-HVAC	Control Structure – Heating, Ventilating and Air Conditioning
DAR	Design Assessment Report
DBA	Design Basis Accident
DBA LOCA	Design Basis Loss-of-Coolant Accident
DC	Direct Current
DG	Diesel Generator
DGB	Diesel Generator Building
DGB H&V	Diesel Generator Building Heating and Ventilation
ECCS	Emergency Core Cooling System

EDG	**Emergency Diesel Generator**
EFPY	**Effective Full-Power Year**
EHC	**Electrohydraulic Control**
ELLLA	**Extended Load Line Limit Analysis**
EOC	**End-of-Cycle**
EOC RPT	**End-of-Cycle Recirculation Pump Trip**
EOP	**Emergency Operating Procedure**
EOF	**Emergency Operating Facility**
EPG	**Emergency Procedure Guideline**
EPP	**Environmental Protection Plan**
EQ	**Environmental Qualification**
ER-OL	**Environmental Report – Operating License Stage**
ESF	**Engineered Safety Feature**
ESSW	**Engineered Safeguards Service Water (ESW plus RHRSW)**
ESSWP H&V	**Engineered Safeguards Service Water Pump house Heating and Ventilation**
ESW	**Emergency Service Water**
FES	**Final Environmental Statement**
FHA	**Fuel Handling Accident**
FSAR	**Final Safety Analysis Report**
FWCF	**Feedwater Controller Failure**
GE	**General Electric Company, GE Nuclear Energy**
GLRWOB	**Generator Load Reject Without Bypass**
GRRCCW	**Gaseous Radwaste Closed Cooling Water**
HCU	**Hydraulic Control Unit**
HELB	**High Energy Line Break**
HEPA	**High-Efficiency Particle Absolute (filter)**
HPCI	**High Pressure Coolant Injection**
HVAC	**Heating, Ventilating and Air Conditioning**
H&V	**Heating and Ventilation**
ICF	**Increased Core Flow**
I&C	**Instrumentation and Control**
IE	**(NRC Division of) Inspection and Enforcement**
IEB	**Inspection and Enforcement Bulletin**
IEEE	**Institute of Electrical and Electronic Engineers**
INPO	**Institute of Nuclear Power Operation**
IPE	**Individual Plant Evaluation**
IPEEE	**Individual Plant Evaluation – External Events**
IRM	**Intermediate Range Monitor**
KWU	**Kraftwerk Union A. G.**
LCS	**Leakage Control System**

LHGR	**Linear Heat Generation Rate**
LOCA	**Loss-of-Coolant Accident**
LOOP	**Loss-of-Offsite Power**
LPCI	**Low Pressure Coolant Injection**
LPRM	**Local Power Range Monitor**
LPZ	**Low Population Zone**
LTR1	**Licensing Topical Report No. 1**
LTR2	**Licensing Topical Report No. 2**
MAPLHGR	**Maximum Average Planar Linear Heat Generation Rate**
MCC	**Motor Control Center**
MCPR	**Minimum Critical Power Ratio**
MELB	**Moderate Energy Line Break**
MG	**Motor Generator**
MHT	**Minimum Heat Transfer (ultimate heat sink analysis)**
MSIV	**Main Steam Isolation Valve**
MSIV-LCS	**Main Steam Isolation Valve – Leakage Control System**
MSLB	**Main Steam Line Break**
MSLBA	**Main Steam Line Break Accident**
MSLHR	**Main Steam Line High Radiation**
MWD/T	**Megawatt Days/(Metric) Tonne**
MWL	**Maximum Water Loss (ultimate heat sink analysis)**
MWt	**Megawatt thermal**
NBR	**Nuclear Boiler Rating**
NIST	**National Institute of Standards and Testing**
NPSH	**Net Positive Suction Head**
NPSHa	**Net Positive Suction Head available**
NPSHr	**Net Positive Suction Head required**
NRC	**Nuclear Regulatory Commission**
NSSS	**Nuclear Steam Supply System**
NUMARC	**Nuclear Management and Resource Council**
NUREG	**Nuclear Regulatory Commission (publication series)**
OLMCPR	**Operating Limit Minimum Critical Power ratio**
PCS	**Pressure Control System**
PCT	**Peak Cladding Temperature**
PPL	**PPL Susquehanna LLC**
PSAM	**Pool Swell Analytical Model**
RBCCW	**Reactor Building Closed Cooling Water**
RBCW	**Reactor Building Chilled Water**
RBM	**Rod Block Monitor**
RCIC	**Reactor Core Isolation Cooling**
RFCF	**Recirculation Flow Controller Failure**

RHR	**Residual Heat Removal**
RHRSW	**Residual Heat Removal Service Water**
RIPD	**Reactor Internals Pressure Differential**
RMA	**Risk Management Associates**
RPS	**Reactor Protection System**
RPT	**Recirculation Pump Trip**
RPV	**Reactor Pressure Vessel**
RT_{NDT}	**Null-Ductility Transition Reference Temperature**
RWCU	**Reactor Water Cleanup**
RWE	**Rod Withdrawal Error**
SBO	**Station Blackout**
SGTS	**Standby Gas Treatment System**
SIL	**GE Service Information Letter**
SJAE	**Steam Jet Air Ejector**
SLCS	**Standby Liquid Control System**
SLO	**Single-Loop Operation**
SLMCPR	**Safety Limit Minimum Critical Power Ratio**
SPC	**Siemens Power Corporation (formally SPC (Siemens Nuclear Power))**
SRM	**Source Range Monitor**
SRV	**Safety Relief Valve**
SRVDL	**Safety Relief Valve Discharge Line**
SSES	**Susquehanna Steam Electric Station**
STP	**Simulated Thermal Power**
STPM	**Simulated Thermal Power Monitor**
SW	**Service Water**
TBCCW	**Turbine Building Closed Cooling Water**
TBCW	**Turbine Building Chilled Water**
TB-HVAC	**Turbine Building – Heating, Ventilating and Air Conditioning**
TCV	**Turbine Control Valve**
TIP	**Traversing In-Core Probe**
TRM	**Technical Requirements Manual**
TS1	**Technical Specifications, Unit 1**
TS2	**Technical Specifications, Unit 2**
TSC	**Technical Support System**
TTWOB	**Turbine Trip Without Bypass**
UHS	**Ultimate Heat Sink**
VWO	**Valves Wide Open**

Susquehanna Steam Electric Station
Units 1 and 2

Licensing Topical Report for
POWER UPRATE RESULTING FROM INCREASED FEEDWATER FLOW MEASUREMENT ACCURACY

1.0 INTRODUCTION

1.1 PURPOSE

This licensing topical report supports a 1.4% increase in rated thermal power (RTP). A 1.4% increase in RTP will result in a like increase in feedwater and steam flow. The RTP increase will result in an increase of approximately 1 psi in reactor operating pressure (from 1048 to 1049 psia) and a small increase in saturated temperature corresponding to the pressure increase. The operating pressure of 1050 psia evaluated as part of the previous power uprate (Reference 1.6) bounds the expected, operating pressure of the reactor.

The 1.4% increase in RTP will be achieved by increasing feedwater flow to the reactor vessel and steam flow from the reactor vessel. The increase in steam flow will be achieved by adjusting the turbine control valve position to reduce the main steam line flow resistance. A corresponding increase in the thermal power assumed in 10CFR50 Appendix K accident analyses will not be made. The Caldon, Inc. Leading Edge Flow Meter✓™ system will be installed to increase the feedwater flow measurement accuracy, obviating the need for the 2% power uncertainty required by 10CFR50 Appendix K. The Topical Report for the LEFM✓™ system is attached to this Licensing Topical Report as Attachment 1. Attachment 1 details the applicability of the LEFM✓™ measurement system for boiling water reactors (BWRs) such as Susquehanna. The documentation details the stated claim that the LEFM✓™ flow measuring system has an accuracy of less than ±0.6% (all accuracy values quoted in this report are based on standard, 95% confidence interval methodology) of full reactor power for two feedwater loop BWRs. Since SSES has three feedwater loops feeding the reactor vessel and the uncertainty in measurement statistically decreases with the number of measurements, installation of the LEFM✓™ system at SSES will result in uncertainties less than quoted in the Attachment.

Regulatory Guide 1.49 (Reference 1.7) requires that all licensing basis analyses assume an initial power level of 1.02 times the licensed power level, to allow for possible error resulting from the measurement of reactor thermal power. PPL conducted a thorough review of the calorimetric calculation for core thermal power. In addition, PPL completed a determination of the current and projected uncertainty in the calculation of core thermal power level (Reference 1.8). The methodology detailed in Reference 1.9 was applied to both the current system of core thermal power measurement, using the venturi flow meters to determine feedwater flow, and the proposed LEFM✓ flow measurement system as input to the core thermal power calculation. The results of this calculation showed that the accuracy of the current core thermal power measurement is approximately ± 1%, not including allowance for instrument drift, for the current, venturi based, calorimetric calculation. The same methodology applied to the LEFM✓ based calculation showed that the core calorimetric calculation results in an error band of less than ± 0.55% of rated thermal power. Therefore, PPL concludes that, with the installation of the LEFM✓ system, SSES can operate at a licensed thermal power level of 3489 MWt, with no greater probability of exceeding the analysis power level of 3510 MWt than exists with the current calorimetric calculation.

Regulatory Guide 1.49 states that the factor of 1.02 should be applied to the initial power level for all analyses performed in support of the license application. This requirement applies to the initial power level used for analysis of (a) normal operating conditions, (b) anticipated transient conditions, (c) design basis accident conditions and (d) conditions related to potential off-site radiological dose consequences from postulated accidents. Current analysis in support of the license application for SSES is performed at initial power levels of 3510 MWt or greater, and the power level of 3510 MWt is 1.02 times the current licensed power level of 3441 MWt. PPL concludes that analysis performed at power levels of 3510 MWt or greater meets the intent of Regulatory Guide 1.49, since the probability of exceeding 3510 MWt, operating at the licensed thermal power level of 3489 MWt, with the addition of the LEFM✓system, is unchanged from the previous analysis. Therefore, any analysis in support of the licensed application performed at an initial power level of 3510 MWt or greater remains bounding and needs no revision, even though the value of 3510 MWt does not meet the explicit guidelines of Reference 1.7. (That is, the analyzed power level of 3510 MWt is less than the 1.02 factor of the proposed licensed thermal power level of 3489 MWt).

The operating power level of nuclear generating plants can be increased safely if the installed systems and equipment are capable of performing required functions at the uprated conditions. The uprate in power discussed herein is obtained by allowing power operation closer to the power level analyzed in accident analyses, by virtue of the installation of the Leading Edge Flow Meter™, (LEFM™) manufactured by Caldon, Inc. The increase in power takes advantage of the increased accuracy of the LEFM™ instrument versus the currently installed venturi feedwater flow instrument.

This report follows the NRC-approved generic format and content for BWR power uprate licensing reports and follows the same format as the previous Licensing Topical Report submitted by PPL (see Reference 1.6). This report addresses the same issues addressed in the previous PPL Power Uprate LTR, with the exception that, since the power level for design and licensing calculations do not change in the present case, these calculations are cited from the previous evaluation.

Siemens Power Corporation (SPC) furnishes fuel and certain analyses for reload license amendment requests. The operating cycles for which this power uprate applies will contain all SPC fuel in both Units 1 and 2. Therefore, the applicable LOCA analyses for this power uprate will be furnished by SPC, as part of the reload licensing process.

Descriptions of the plant and its systems, and of safety analysis methods, may be found in the SSES FSAR or in the references cited.

1.2 EVALUATION APPROACH

The generating capacity of a BWR plant can be increased by increasing steam flow through the turbine, up to the limits imposed by the licensed reactor power and the turbine and generator designs. The power uprate discussed herein, proposes to increase the RTP by decreasing the variability (that is, increasing the accuracy) of the feedwater flow measurement. The power level used for licensing bases and LOCA analyses is not changed. Therefore, any design bases analyses performed using the current licensing analysis basis power level of 3510 MWt is not affected and does not require any further review.

This report describes the evaluations that show that both Units 1 and 2 of the Susquehanna Steam Electric Station (SSES) will operate safely with the requested increase in RTP. Evaluations include those performed on the turbine-generator system to assure that the increased steam flow can be accommodated, and on the balance of plant and support systems to assure that these systems remain within their design basis.

The approach, scope and detail of these evaluations are based on the generic BWR power uprate guidelines presented in LTR1 and LTR2, and the specific design features of the SSES units. Fuel design information and analysis is based on the use of SPC ATRIUM-10™ fuel bundles in both units and specifics of the fuel design and analyses are contained in documented calculation packages and the updated Susquehanna FSAR (Reference 1.1).

The plant-unique evaluations included reviews of design documents, operating data and specific studies prepared for the previous power uprate submittal for the SSES units. These studies and data have been evaluated to confirm that the SSES design is adequate for the uprated conditions; however, in addition to the installation of the Caldon LEFM✓™ system, two further modifications may be required to allow full usage of the total increase in RTP:

1. The turbine steam passing capability, as it presently exists, may have to be increased to allow full usage of the total increase in RTP. PPL plans to increase the turbine steam flow rate to the maximum allowed, within the limits of the RTP for the core, and evaluate alternatives for increasing steam flow for a later outage. The limit on steam passing capability does not affect the safety or licensing basis of the plant. The steam capacity limit affects only the thermal and electrical power level at which the station operates economically. Thus, PPL is requesting the total 1.4% RTP increase, and will limit the core power to the licensed power level or less, depending on economics.

2. The expected increase in total MW(e) for the station makes case N-10 in Reference 1.1, Table 8.2-1 go from a stable to an unstable transient on the bulk power system. A modification to return this fault to stable has been identified and is discussed in Section 6.1 of this Topical. The evaluations indicate that the modification to the bulk transmission system discussed in Section 6.1 are not required until the second unit at the station (that is, Unit 1 in spring 2002) completes this power uprate.

1.2.1 Bounding Analyses for BWRs

Certain features and issues are common to all BWRs, and bounding evaluations have been completed for many of the issues associated with the current increase in licensed power level for the previous power uprate. These bounding analyses are described and cited in Reference 1.6.

1.2.2 Reactor Core and Fuel Performance

Core and fuel performance are evaluated for each fuel cycle and reported separately for each reload of each unit in documented calculation packages and the updated Susquehanna FSAR (Reference 1.1). The cycle chosen for the initial implementation of the reactor operating power increase resulting from increased feedwater flow measurement accuracy is the first SSES cycle with a core made up completely with SPC ATRIUM-10™ fuel bundles. Therefore, the LOCA analysis of record will be the analysis performed by SPC.

The general effects of this power uprate on these analyses are described in Chapter 2.0.

1.2.3 NSSS and ECCS Systems and Components

The NSSS systems and the emergency core cooling systems (ECCS) and components were reviewed to verify that the ECCS analysis was performed at an assumed initial power level of 3510 MWt. Chapters 3.0, 4.0 and 5.0 describe these evaluations and their conclusions. These analyses confirm that safety-related components will perform adequately and with adequate control margins. Because there is no change to the assumed licensing analysis power level, no changes to safety setpoints or other design features are required to ensure adequate system and component performance.

Components which are not safety-related but whose performance may be affected by this power uprate, and that may thereby compromise the ability of the plant to operate satisfactorily at the uprated conditions, were also evaluated.

1.2.4 Power Conversion, Electrical and Auxiliary Systems

Power conversion, electrical and auxiliary systems possibly affected by this power uprate were evaluated at the uprated conditions. Chapters 6.0 and 7.0 describe these evaluations and their conclusions. The

evaluations confirm that these systems and components will perform adequately, with adequate control margins. No setpoint changes or changes to any other design features in the power conversion, electrical and auxiliary system regime are required, with the exception of the transformer breaker modification discussed in Section 6.1. The effect of the turbine steam passing capability is not discussed further, because it is an economic consideration and has no effect on the safety or licensing basis of the plant.

1.2.5 Containment Capability

Section 4.1 summarizes the evaluation of the effects of the power uprate on containment analyses. This power uprate has no affect on the licensing analyses because the licensing analyses were performed at the bounding value of 3510 MWt (102% of current RTP level). Therefore, no further analyses were required.

1.2.6 Feedwater Flow Measurement

The increase in rated thermal power (RTP) for the reactor core requested and justified herein in based upon the increased accuracy in feedwater flow measurement. The LEFM✓™ system will be installed as part of the justification for operation at increased RTP. The evaluation of increased accuracy, the plan of installation and plant operation under the condition of operability and inoperability of the LEFM✓™ flow measurement system is discussed in Section 5.2.1

1.2.7 High- and Moderate-Energy Line Breaks

Analyses of high- and moderate-energy line break effects were reviewed to evaluate possible effects of operating at the uprated power level. Since the analyzed pressure conditions (that is, all high energy line breaks were analyzed using an initial pressure of 1053 psia, corresponding to the expected steam dome pressure at 3510 MWt) do not change, the effects of these breaks do not change from previously evaluated. This review is described in Sections 10.1 and 10.2.

1.2.8 Piping Systems

The evaluation of reactor coolant pressure boundary piping and balance of plant piping is described in Sections 3.5.1 and Section 3.5, respectively. Since the reactor system pressure used for analysis purposes (1050 psia) parameters did not change from previously licensed conditions, the assurance that the piping

design conditions were met under the previous conditions was sufficient to assure that the piping would meet the currently uprated conditions in an acceptable manner. Loads from the containment events and high energy line breaks described in Sections 4.1, 10.1 and 10.2, respectively, also did not change as a result of the current power uprate conditions. Therefore, the current power uprate has no effect on either the reactor coolant pressure boundary piping or the balance of plant piping.

1.2.9 Radiological Analyses

Radiological analyses for events ranging from normal operating conditions through FSAR Chapter 15 events were evaluated to show that the regulatory limits for on-site dose, off-site dose and control room habitability will be met. These evaluations included effects of higher reactor power on source terms and are discussed in detail in Chapters 8.0 and 9.0.

1.2.10 Reactor Safety Analyses

Event and cycle specific calculations performed for each reload cycle confirm that the analyses of Final Safety Analysis Report Chapter 15 limiting events remain within regulatory limits. Chapter 15 limiting events are analyzed with the initial conditions up to 102% of previous rated power (3510 MWt) and the assumed initial conditions are not changed for this power uprate. Therefore, cycle specific analysis methods and assumptions are unaffected by the increase in rated thermal power level.

1.2.11 General Issues

Section 9.3 describes the effects of the increased rated core thermal power level on special transient events, such as station blackout (SBO) and anticipated transients without scram (ATWS).

The ATWS analysis is completed for each reload. The initial conditions assumed for the ATWS analysis is the RTP level, not the bounding value of 3510 MWt. The ATWS analysis for each operating cycle in which power is uprated as a result of the feedwater flow instrumentation will be performed using the uprated RTP level.

Station blackout analysis also assumes the normal RTP level as its initial condition. The effect of operating at the uprated power level is the increased decay heat level that must be accounted for in the analysis. The result and conclusion of the review is discussed in Section 9.3.

Chapter 10.0 describes the review of the plant design for possible effects of the uprated RTP conditions on environmental qualification, the individual plant evaluation (IPE), the individual plant evaluation-external events (IPEEE) report and the emergency operating procedures (EOP's).

1.2.12 Component Setpoints and Ranges

Instruments and safety valves were reviewed as part of each system evaluation. An additional systematic review was performed for safety valves and instruments with a ranges and/or setpoints that may be affected by the uprated RTP. The results of this review are discussed in Sections 5.1 and 5.2.

1.2.13 Licensing and Design Basis Issues

Applicable licensing commitments, IE Bulletins, circulars, notices, etc. were evaluated for possible effects with respect to operating at the uprated power level. Section 11.1 describes the review of items requiring specific evaluation for SSES.

Changes to the Technical Specifications are required for operation at increased RTP. These changes include the revised definition of rated thermal power in Section 1 and revisions to TS 5.6.5b, discussing the LEFM✓™ system and the design basis power level assumed for SSES operating and accident analysis. The consequences of the failure of the LEFM✓™ on plant operation are also discussed in TS 5.6.5b.

The LEFM✓™ has a self-verification feature, as discussed in Attachment 1. In addition, the LEFM✓™ system is designed with a completely redundant set of sensing instrumentation, therefore, it is highly unlikely that the LEFM✓™ will become unavailable. However, when operation of the LEFM✓™ cannot be verified, an alarm is generated and the core thermal power will be reduced to the previous RTP of 3441 MWt, as defined in TS 5.6.5b.

1.2.14 Environmental Assessment

The SSES environmental protection plans (References 1.4 and 1.5) were reviewed to confirm that the effects of operating at the increased power level will be within the limits established by these plans. Section 11.4 discusses the results of the review.

1.3 UPRATED CONDITIONS

The proposed increase RTP corresponds to a 1.4% increase in main steam flow to the turbine-generator. The resulting steam flow (14.4 MLb_m/hr) is less than the valve wide open (VWO) steam flow (14.63 MLb_m/hr). The uprated operating power level is within the turbine-generator design basis. This increased steam flow is consistent with a 1.4% increase in licensed, RTP level from 3441 MWt to 3489 MWt.

Increasing core flow along the rod/flow control lines, as shown on the power/flow map (Figure 2-1) will increase reactor power level. The power/flow map has been expanded (in that 100% power corresponds to the new RTP level) to account for the increased RTP level.

SSES is currently licensed to operate with extended load line limit analysis (ELLLA), along the APRM rod block monitor line. The analyses contained in this report were performed to support a 1.4% increase in operating rated thermal power (RTP). The ELLLA line has been retained and the power rescaled so that the increased RTP is the new 100% power level.

The 1.4% increase in turbine steam flow will be accomplished by reducing the resistance in the steam lines (that is, opening the turbine control valves slightly). An inconsequential increase in actual reactor operating pressure is expected, but the expected operating pressure of 1049 psia remains less than the previously evaluated value of 1050 psia.

1.3.1 Reactor Heat Balance

The thermal power, pressure, total core flow and coolant enthalpy determine the thermal hydraulic performance of a BWR. These parameters, detailed in Table 1-1, are used to define the steady state operating conditions and the initial and boundary conditions for safety analyses. They are determined by energy balance calculations ("heat balances').

Reactor heat balances for the 101% and 101.4% steam flow uprate were performed, and the results are displayed in Figures 1-1 and 1-2, respectively, for a core flow of 100 MLb_m/hr and in Figures 1-3 and 1-4, respectively for 108 MLb_m/hr. The 101.4% heat balance (or 3489 MWt) corresponds to the maximum increase in rated thermal power justified by the increase in measurement accuracy with the newly installed CALDON LEFM✓™ feedwater flow measurement system. The heat balance at 101% of current licensed power level (or 3475 MWt) is provided because that power level is the expected operating power level based on the turbine steam passing capacity (see Section 1.2). The heat balance for 102% of current licensing power level (3510 MWt) is shown in Figure 1-5 and has not changed (see Reference 1.6).

Current licensed thermal power (Nuclear Boiler Rating, NBR)	3441 MWt
Proposed uprated licensed thermal power (101% uprate)	3475 MWt
Proposed uprated licensed thermal power (101.4% uprate)	3489 MWt
3475 MWt/3441 MWt	1.01 (101%)
3489 MWt/3441 MWt	1.014 (101.4%)

The proposed uprate in licensed thermal power level results from increased accuracy of the feedwater flow measurement and, consequently, the increased accuracy of the reactor thermal power calculation. Therefore, the current ECCS power level of 3510 MWt, or 102% of current RTP is not affected.

The reactor heat balances are coordinated with the turbine heat balances, which are supplied by the turbine-generator vendor. Tables 1-1 (for 3475 MWt) and 1-2 (for 3489 MWt) summarize the changes is reactor operating parameters.

1.3.2 Uprate Analysis Basis

Reactor: Most reactor design, transient and accident analysis for the current licensing basis were performed up to 102% of licensed power level or 3510 MWt, as specified by Regulatory Guide 1.49

(Reference 1.7) for power-dependent safety analyses. (Anticipated transients without scram (ATWS) and station blackout (SBO) are the exceptions to the above statement because both ATWS and SBO transients are initiated from RTP.) This proposed change in licensed power level does not change this analysis value; however, the licensed power increase is obtained by installing a feedwater flow measurement system that reduces the uncertainty of the feedwater flow and temperature measurements. The feedwater flow and temperature measurements comprise approximately 99% of the reactor thermal power calculation (see Reference 1.8), consequently, increasing the accuracy and decreasing the variability of the feedwater flow determination increases the accuracy and decreases the variability of the core thermal power calculation. Figure 1-5, which is unchanged from the current licensing basis, shows the reactor heat balance at the conditions of 3510 MWt (the licensing basis power level) and 100 MLb_m/hr.

Containment: Containment analyses in Section 6.2 of the FSAR (Reference 1.1) as well as all containment design bases analyses (NUREG-0783 analyses) were performed at an initial power level of 3510 MWt. Therefore, these analyses are not affected. In addition, the power level used for the LOCA analyses for Siemens Power Corporation (SPC) ATRIUM™-10 fuel, as documented in Table 6.3-2 of the FSAR also assumes an initial power level of 3510 MWt. These analyses will not be revised as a result of this submittal. LOCA analyses are, and will continue to be, redone as required to support each operating cycle.

Emergency Core Cooling Systems (ECCS) and Engineered Safety Features (ESF): The bases for evaluation of ECCS and ESF systems and components at the revised, uprated, licensed thermal power level are consistent with those used for evaluations of the reactor and containment.

Radiological Consequences of Accidents: The radiological consequences of design basis accidents, discussed in Section 9.2, were determined using a power level equal to 105% of current licensing basis core power. This power level is greater than the 3510 MWt (102% of the previous licensed power level), therefore, no radiological reanalysis is required to support this submittal.

Margin: The basis for evaluation of the reactor, containment, ECCS and ESF systems and design basis accidents at the uprated reactor thermal power level assure that the power- and flow-dependent margins required by federal regulations and by design codes, for design and design basis evaluations, will be maintained.

1.4 SUMMARY OF REQUESTED LICENSE AND TECHNICAL SPECIFICATION CHANGES

The increase in licensed reactor operating power will require changes to the SSES Operating License. Section 11.2 describes the changes required to support this increase in rated thermal power.

1.5 CONCLUSION

This report evaluates the effects of a 1.4% increase in licensed reactor operating power. The evaluations conclude that the Susquehanna Steam Electric Station Units 1 and 2 will operate safely at the increased reactor operating power level.

REFERENCES

1.1 Pennsylvania Power & Light Company, "Susquehanna Steam Electric Station Units 1 & 2 Final Safety Analysis Report (FSAR)," Dockets 50-387 and 50-388.

1.2 Pennsylvania Power & Light Company, "Susquehanna Steam Electric Station Unit 1 Technical Specifications (TSI)," Docket No. 50-387. Appendix A to Facility Operating License No. NPF-14.

1.3 Pennsylvania Power & Light Company, "Susquehanna Steam Electric Station Unit 2 Technical Specifications (TSI)," Docket No. 50-388. Appendix A to Facility Operating License No. NPF-22.

1.4 Pennsylvania Power & Light Company, "Susquehanna Steam Electric Station, Units 1 and 2, Docket Nos. 50-387 and 50-388, Environmental Protection Plan (Non-Radiological)" (Unit 1 EPP), July 17, 1982. Appendix B to Facility Operating License NPF-14.

1.5 Pennsylvania Power & Light Company, "Susquehanna Steam Electric Station, Units 1 and 2, Docket Nos. 50-387 and 50-388, Environmental Protection Plan (Non-Radiological)" (Unit 2 EPP), March 1984. Appendix B to facility Operating License NPF-22.

1.6 Pennsylvania Power & Light Company, "Susquehanna Steam Electric Station, Units 1 and 2, Licensing Topical Report for Power Uprate With Increased Core Flow," Licensing Topical Report NE-92-001, July 1992.

1.7 United States Atomic Energy Commission, Regulatory Guide 1.49, "Power Levels of Nuclear Power Plants,", Rev. 1, December 1973.

1.8 Pennsylvania Power & Light Company, "Determination of Error Band on Reactor Thermal Power Calculation – Before and After FWFE Upgrade," EC-031-1010, May 2000.

1.9 American Society of Mechanical Engineers Standard ASME-PTC-19.1 (1985).

Table 1-1:

Original and Uprated Operating Conditions (3475 MWt)

Parameter	Original Rated Value at 1.0×10^8 lb_m/hr Core Flow	Original Rated Value at 1.08×10^8 lb_m/hr Core Flow	Uprated Value at 1.0×10^8 lb_m/hr Core Flow	Uprated Value at 1.08×10^8 lb_m/hr Core Flow
Thermal Power (Nuclear Boiler Rating, NBR)				
MWt	3441	3441	3475	3475
% Original Rated Power	(100.0)	(100.0)	(101.0)	(101.0)
Full Power Core Flow Range				
10^6 lb_m/hr	87 to 108	87 to 108	87 to 108	87 to 108
Core Inlet Enthalpy (Btu/lb_m)	525.0	526.9	524.3	526.4
Vessel Dome Pressure (psia)				
Design Operating	1050	1050	1050	1050
(Expected Operating)	(1047)	(1047)	(1049)	(1049)
Vessel Steam Dome Temperature (°F)	550.2	550.2	550.3	550.3
Vessel Steam Flow				
10^6 lb_m/hr	14.183	14.184	14.339	14.348
% Rated Steam Flow	(100)	(100)	(101.1)	(101.2)
Turbine Steam Flow				
10^6 lb_m/hr	14.138	14.139	14.294	14.303
% Rated Steam Flow	(100)	(100)	(101.1)	(101.1)
Turbine Inlet Pressure (psia)				
Design Operating	997	997	997	997
(Expected Operating	(997)	(997)	(997)	(997)
Feedwater Flow (10^6 lb_m/hr)	14.151	14.152	14.307	14.316
Full Power Feedwater Temperature (°F)	387.1	387.1	388.2	388.3

Table 1-2:

Original and Uprated Operating Conditions (3489 MWt)

Parameter	Original Rated Value at 1.0×10^8 lb_m/hr Core Flow	Original Rated Value at 1.08×10^8 lb_m/hr Core Flow	Uprated Value at 1.0×10^8 lb_m/hr Core Flow	Uprated Value at 1.08×10^8 lb_m/hr Core Flow
Thermal Power (Nuclear Boiler Rating, NBR)				
MWt	3441	3441	3489	3489
% Original Rated Power	(100.0)	(100.0)	(101.4)	(101.4)
Full Power Core Flow Range				
10^6 lb_m/hr	87 to 108	87 to 108	87 to 108	87 to 108
Core Inlet Enthalpy (Btu/lb_m)	525.0	526.9	524.4	526.4
Vessel Dome Pressure (psia)				
Design Operating	1050	1050	1050	1050
(Expected Operating)	(1047)	(1047)	(1049)	(1049)
Vessel Steam Dome Temperature (°F)	550.2	550.2	550.35	550.35
Vessel Steam Flow				
10^6 lb_m/hr	14.183	14.184	14.406	14.415
% Rated Steam Flow	(100)	(100)	(101.6)	(101.6)
Turbine Steam Flow				
10^6 lb_m/hr	14.138	14.139	14.361	14.370
% Rated Steam Flow	(100)	(100)	(101.6)	(101.6)
Turbine Inlet Pressure (psia)				
Design Operating	997	997	997	997
(Expected Operating	(997)	(997)	(997)	(997)
Feedwater Flow (10^6 lb_m/hr)	14.151	14.152	14.374	14.383
Full Power Feedwater Temperature (°F)	387.1	387.1	388.7	388.7

SSES - LEFM Power Uprate Heat Balance

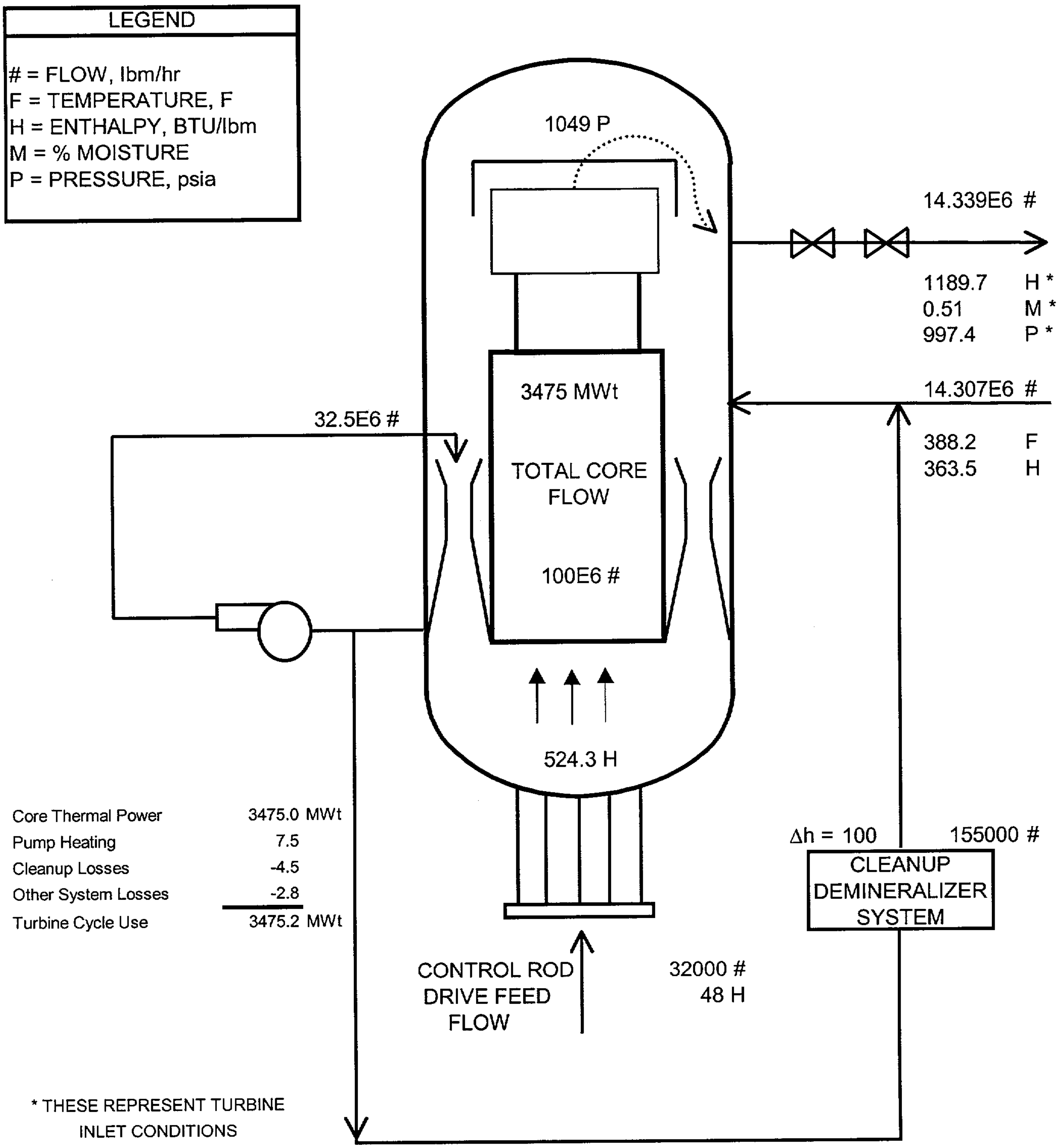


Figure: 1-1 Reactor Heat Balance at 3475 MWt/100 Mlbm/hr for LEFM Power Uprate
Reactor Heat Balance at 101.0% for LEFM Power Uprate

SSES - LEFM Power Uprate Heat Balance

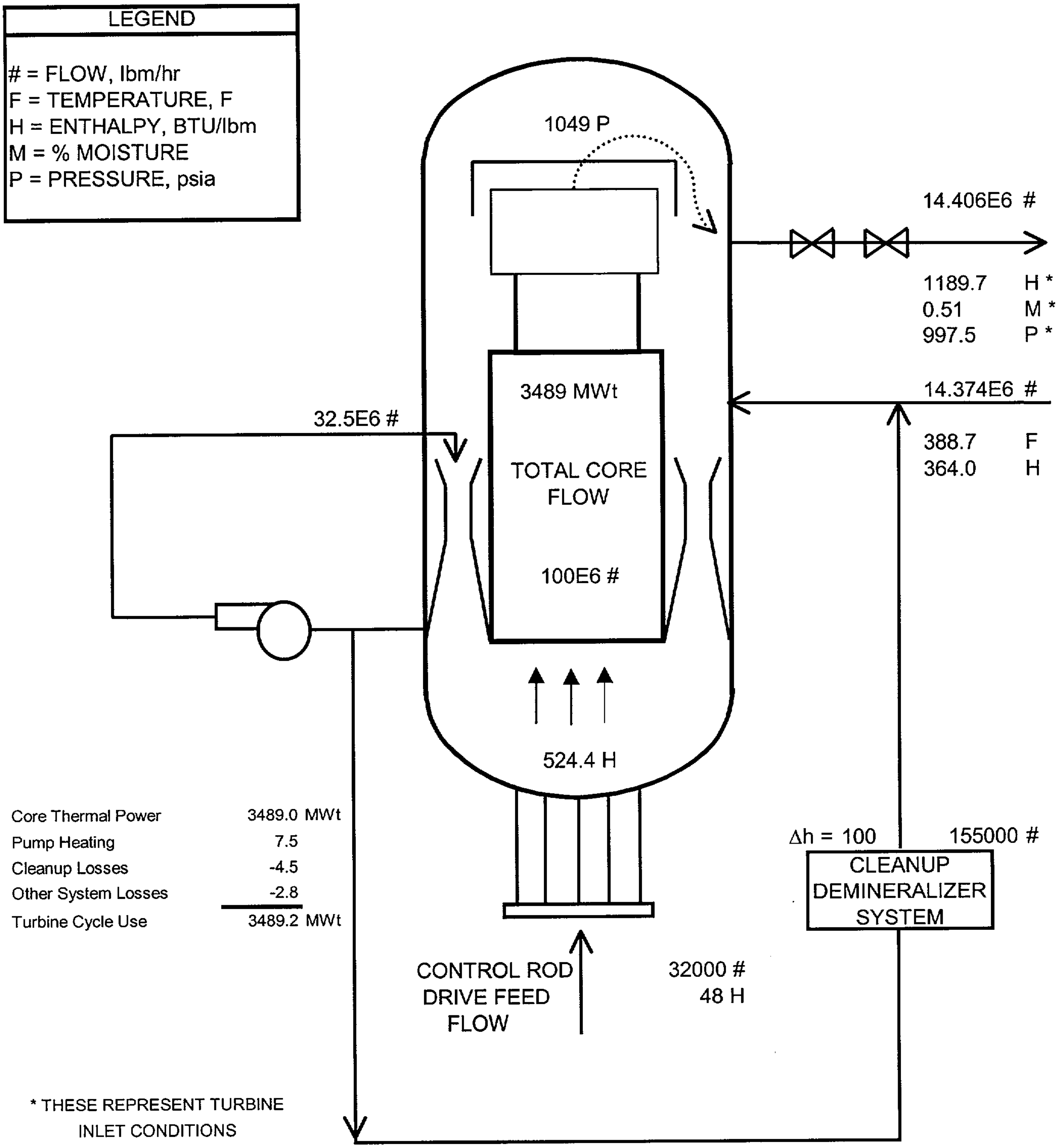


FIGURE: 1-2 Reactor Heat Balance at 3489 MWt/100 Mlbm/hr for LEFM Power Uprate
Reactor Heat Balance at 101.4% for LEFM Power Uprate

SSES - LEFM Power Uprate Heat Balance

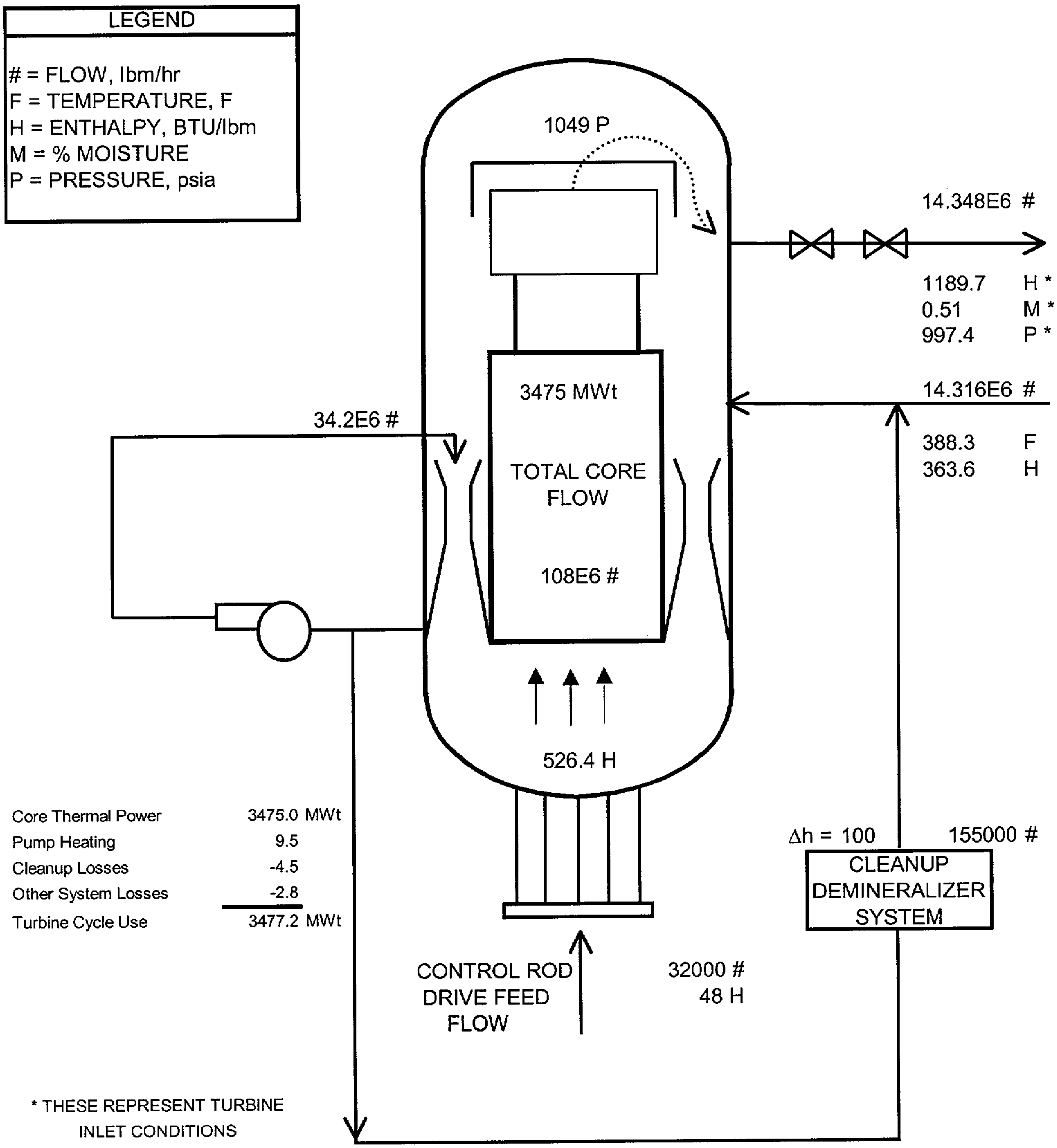


Figure: 1-3 Reactor Heat Balance at 3475 MWt/108Mlbm/hr for LEFM Power Uprate
Reactor Heat Balance at 101.0% for LEFM Power Uprate

SSES - LEFM Power Uprate Heat Balance

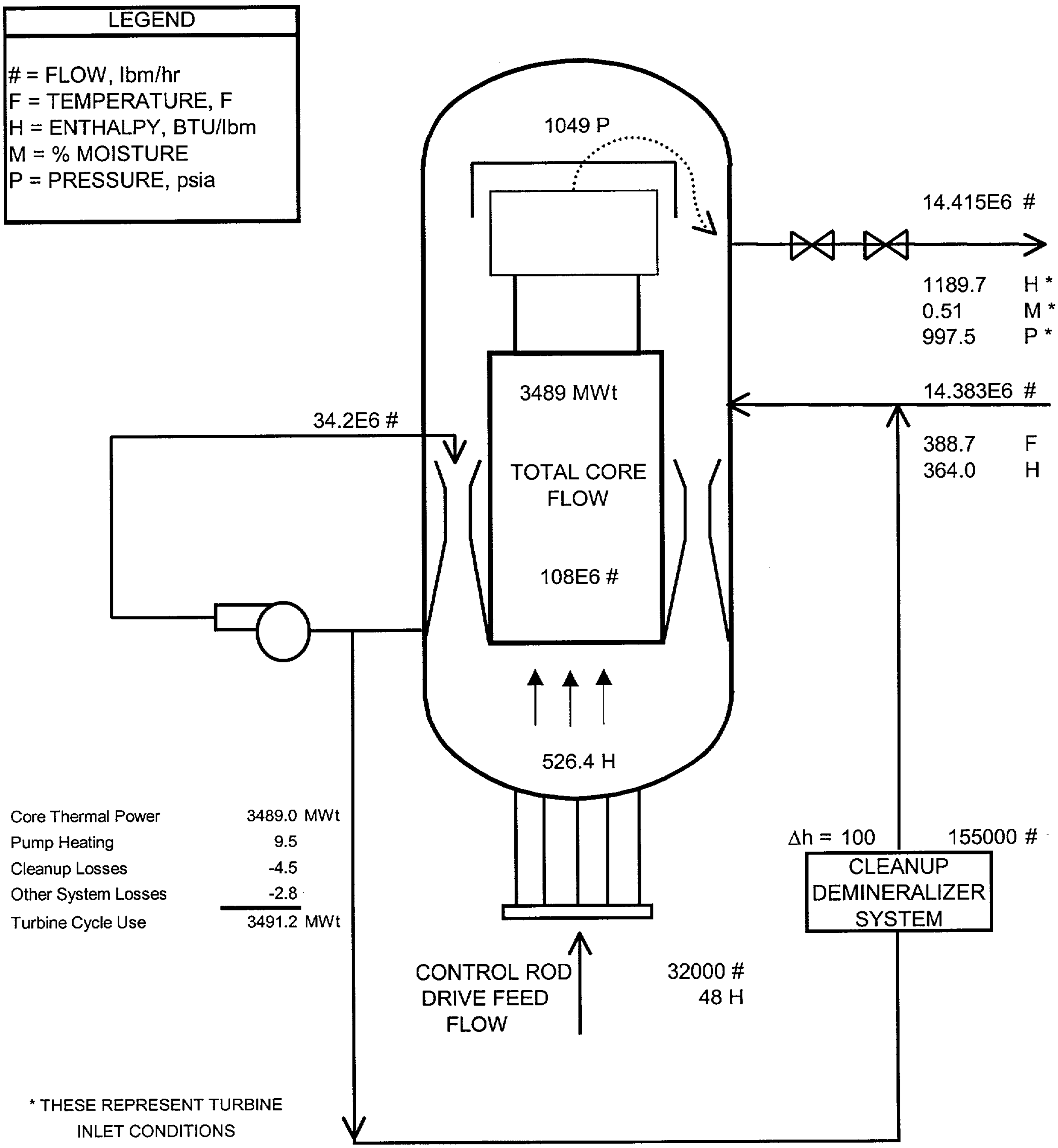
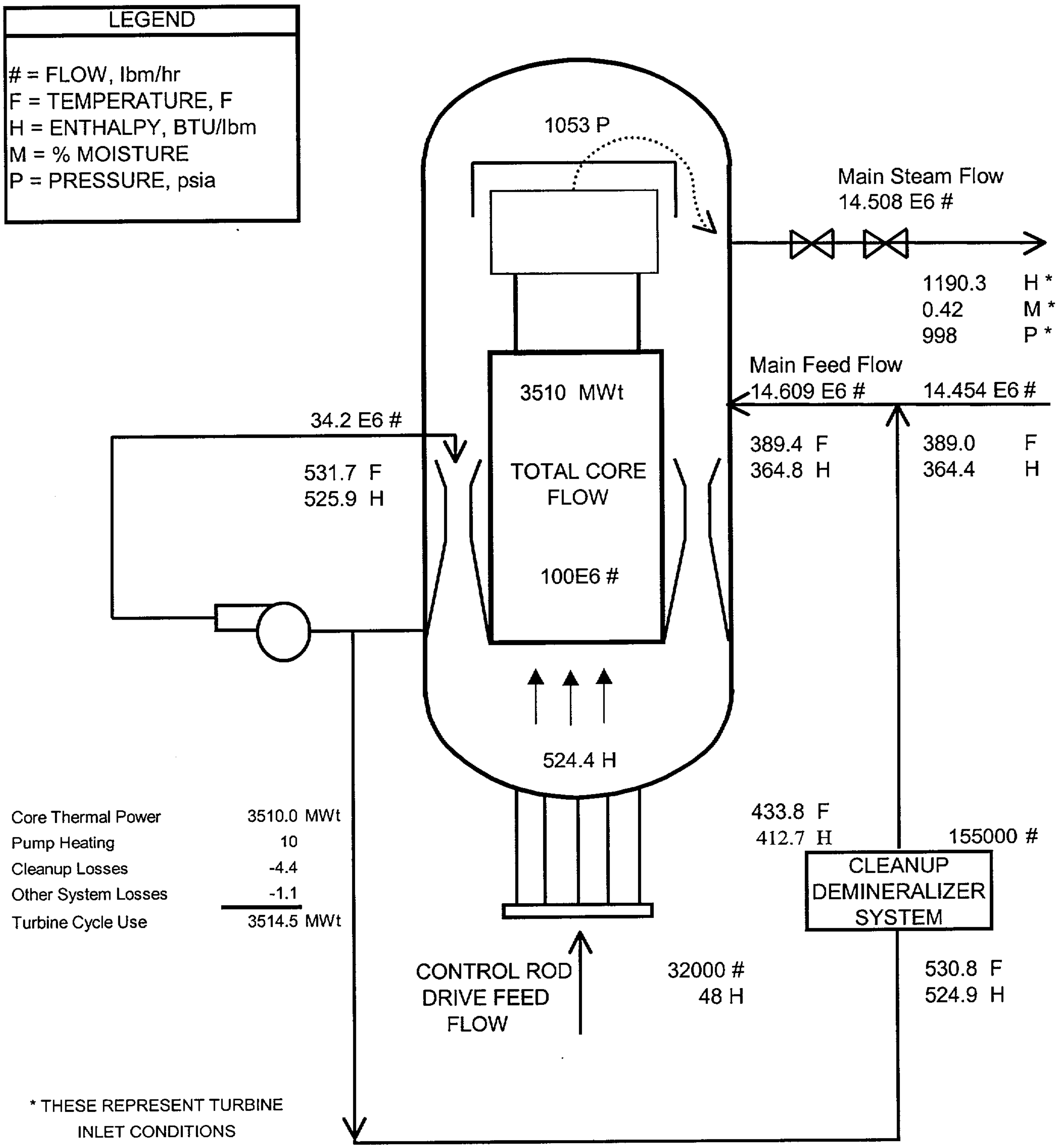


Figure: 1-4 Reactor Heat Balance at 3489 MWt/108 Mlbm/hr for LEFM Power Uprate
Reactor Heat Balance at 101.4% for LEFM Power Uprate

SSES - LEFM Power Uprate Heat Balance



Core Thermal Power	3510.0	MWt
Pump Heating	10	
Cleanup Losses	-4.4	
Other System Losses	-1.1	
Turbine Cycle Use	3514.5	MWt

* THESE REPRESENT TURBINE INLET CONDITIONS

Figure 1-5: Reactor Heat Balance at 102% of Current Licensed Power Level (3510 MWt/100 Mlbm/hr)
Reactor Heat Balance at 101.4% for LEFM Power Uprate

2.0 REACTOR CORE AND FUEL PERFORMANCE

This chapter primarily includes information requested by Regulatory Guide 1.70, Chapter 4, as it applies to increased, rated core thermal power.

2.1 FUEL DESIGN AND OPERATION

At current licensed power level or increased thermal power level, all fuel and core design limits will continue to be met by control rod pattern adjustments or other changes in reactor operation. New mechanical fuel designs are not needed for an increase to the licensed operating power level to assure adequate safety. Although they are not currently being proposed for SSES, new fuel designs may be used in the future to provide additional operating flexibility and maintain fuel cycle length.

Thermal-hydraulic design and operating limits assure an acceptably low probability of boiling transition-induced fuel cladding failure occurring in the core at any time, even for the most severe operational transients. Limits are also placed on the peak linear heat generation rates in order to meet both peak fuel cladding temperature limits for the limiting loss-of-coolant accident (LOCA) and fuel mechanical design bases.

The subsequent reload core designs for operation at the increased licensed power level will take into account the effect of the increased power level on the above limits to assure acceptable differences between the licensing limits and their corresponding operating values. The licensed power increase will increase the plant power density; however, this power density is still within the current operating power density range of other BWRs. The licensed power increase will have some effects on operating flexibility, reactivity characteristics and energy requirements. These issues are discussed in the following sections.

2.2 THERMAL LIMITS ASSESSMENT

Operating limits are established to assure that regulatory safety limits are not exceeded for a range of postulated events (transients, LOCAs, etc.). This section addresses the effects of increased RTP on thermal limits. Cycle specific analyses will be performed to establish or confirm cycle-specific limits, as is currently the practice.

2.2.1 Minimum Critical Power Ratio (MCPR)

The operating limit minimum critical power ratio (OLMCPR) is determined on a cycle-specific basis from the results of the reload transient analysis. This approach will not change. The results of the transient analysis for RTP, as discussed in Section 9.1, will demonstrate that adequate operating minimum critical power ratio (MCPR) margins continue to exist at the increased, licensed power level, and that operation will not be unduly restricted. As is pointed out in Section 9.1, the initial power level assumed in the limiting MCPR analyses is a maximum of 3510 MWt, which is the same maximum value currently used in the MCPR evaluation (see Reference 1.1, Chapter 15). It should be noted, however, that the peak ΔCPR is not always determined by operation at 100% power, but is more a function of the power/core flow ratio. Cycle specific analyses will be performed prior to each operating cycle.

2.2.2 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and Maximum Linear Heat Generation Rate (LHGR)

Operation within the maximum average planar linear heat generation rate (MAPLHGR) and maximum linear heat generation rate (LHGR) limits will be maintained. LOCA and fuel mechanical design analyses determine the MAPLHGR and LHGR limits. These limits are not affected.

2.3 REACTIVITY CHARACTERISTICS

Operation at higher power could reduce the excess reactivity during the cycle. This loss of reactivity is not expected to significantly degrade the ability to manage the power distribution through the cycle to achieve the target power level. However, the lower reactivity does result in an earlier all-rods-out condition. Sufficient excess reactivity is, however obtained through uprated cycle design to achieve the desired cycle energy. The increase in hot reactivity may result in less hot-to-cold reactivity difference and therefore smaller cold shutdown margins. However, this loss in margin is also accommodated through the uprate core design. If needed, a bundle design with improved shutdown margin characteristics can be used to preserve the flexibility between hot and cold reactivity requirements for future cycles. Reload analyses for each cycle will assure that minimum shutdown margin requirements will be maintained.

2.4 POWER/FLOW OPERATING MAP

The power/flow operating map (Figure 2-1) for operation at increased, licensed power level is based on the increased RTP and the current licensing basis core flow rate. The ELLLA line from the previous power uprate has been maintained through analyses to the new 100% operating power level and the power has been rescaled so that the increased, licensed power level is the new 100% power level.

2.5 STABILITY

The Oscillation Power Range Monitor (OPRM) system has been installed at SSES and is currently undergoing proof testing. The OPRM system was installed to meet the NRC requirements specified in Bulletin No. 88-07 and Supplement 1 to that bulletin (Reference 2.1). Technical Specifications and associated implementing procedures have been incorporated which restrict plant operation in the high power, low core flow region of the BWR power/flow operating map. Specific operator actions have been established to provide clear instructions for the possibility that a reactor inadvertently (or under controlled conditions) enters any of the defined regions. Neither the regions of concern nor the required operator actions are affected by the increase in RTP.

2.6 REACTIVITY CONTROL

2.6.1 Control Rod Drives and CRD Hydraulic System

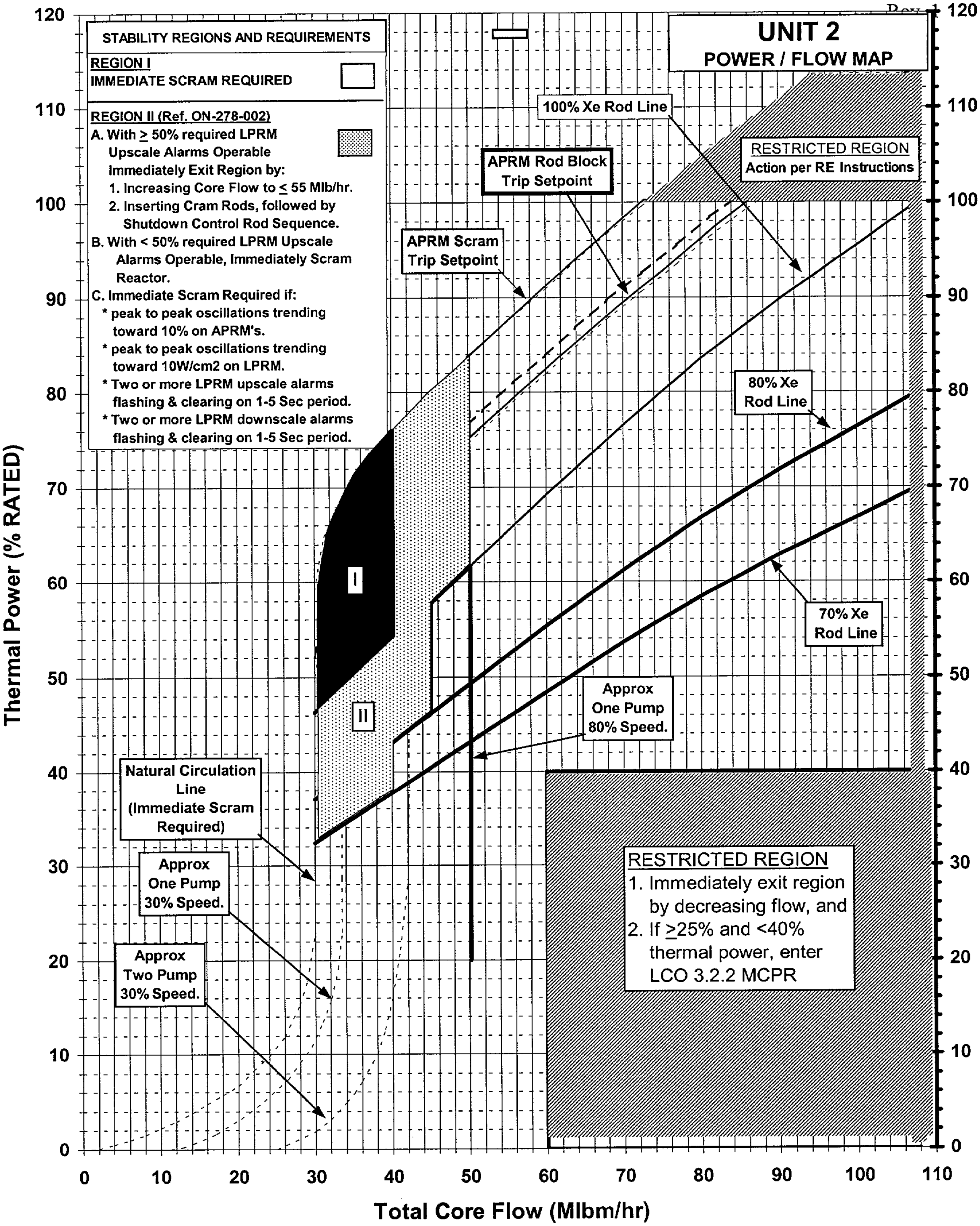
The control rod drive (CRD) system controls gross changes in core reactivity by positioning neutron absorbing control rods within the reactor core. The CRD system is also required to scram the reactor by rapidly inserting withdrawn rods into the core. The CRD system was evaluated to determine what effects the increased licensing basis power level would have on the system.

2.6.2 Reactor Recirculation System

Reactivity aspects of the recirculation system are included in Section 3.4.

REFERENCES

2.1 USNRC Bulletin 88-07, Supplement 1, "Power Oscillations in Boiling Water Reactors (BWRs), December 30, 1988.


UNIT 2
POWER / FLOW MAP
STABILITY REGIONS AND REQUIREMENTS
REGION I
IMMEDIATE SCRAM REQUIRED
REGION II (Ref. ON-278-002)
A. With ≥ 50% required LPRM Upscale Alarms Operable Immediately Exit Region by:
1. Increasing Core Flow to ≤ 55 Mlb/hr.
2. Inserting Cram Rods, followed by Shutdown Control Rod Sequence.
B. With < 50% required LPRM Upscale Alarms Operable, Immediately Scram Reactor.
C. Immediate Scram Required if:
* peak to peak oscillations trending toward 10% on APRM's.
* peak to peak oscillations trending toward 10W/cm2 on LPRM.
* Two or more LPRM upscale alarms flashing & clearing on 1-5 Sec period.
* Two or more LPRM downscale alarms flashing & clearing on 1-5 Sec period.
100% Xe Rod Line
APRM Rod Block Trip Setpoint
APRM Scram Trip Setpoint
RESTRICTED REGION
Action per RE Instructions
80% Xe Rod Line
70% Xe Rod Line
I
II
Approx One Pump 80% Speed.
Natural Circulation Line (Immediate Scram Required)
Approx One Pump 30% Speed.
Approx Two Pump 30% Speed.
RESTRICTED REGION
1. Immediately exit region by decreasing flow, and
2. If ≥25% and <40% thermal power, enter LCO 3.2.2 MCPR
Thermal Power (% RATED)
Total Core Flow (Mlbm/hr)
0
10
20
30
40
50
60
70
80
90
100
110
120

3.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

This chapter primarily includes information requested by Regulatory Guide 1.70 Chapter 5, and limited parts of Chapter 3, as they apply to increased RTP.

3.1 NUCLEAR SYSTEM PRESSURE RELIEF

The nuclear boiler pressure relief system prevents overpressurization of the nuclear system during abnormal operating transients. The plant safety/relief valves (SRVs) and the high pressure reactor scram provide this protection. For increased, licensed thermal power operation, the reactor steam dome pressure will not increase above the design pressure referenced in the previous power uprate submittal (Reference 1.6).

The operating steam dome pressure is defined to achieve good control characteristics for the turbine control valves (TCVs) at the higher steam flow conditions corresponding to increased, licensing basis thermal power level.

The automatic depressurization system (ADS) is discussed in Section 4.2.4.

3.2 CODE OVERPRESSURE PROTECTION

The results of the overpressure protection analysis are contained in each cycle-specific reload safety analysis. The design pressure of the reactor pressure vessel (RPV) remains at 1250 psig. The ASME code allowable peak pressure for the reactor vessel is 1375 psig (110% of design value), which is the acceptance limit for pressurization events. The limiting pressurization event is an MSIV closure with a failure of the valve position scram, which is described in Section 3.5 of Reference 9.1. The MSIV closure event is analyzed using NRC approved methods described in Reference 9.1, with the following exceptions: (1) The MSIV closure event will be analyzed at an initial power level of 3510 MWt and 108 million lb_m/hr core flow and (2) The maximum initial reactor pressure will be assumed to be the Technical Specification maximum value. The conditions are the same as used for previous cycle specific transient overpressure analyses.

The number of SRV's assumed to be out of service is based on the maximum allowed by Technical Specifications. The increased operating power level will produce a higher peak RPV pressure, but the reload analysis is expected to show that the peak RPV pressure in the transient overpressure event remains below the 1375 psig ASME code limit. Therefore, there is no decrease in the margin of safety.

3.3 REACTOR VESSEL AND INTERNALS

A comprehensive review has assessed the effects of the increased RTP on the reactor pressure vessel (RPV) and reactor vessel internals. The review was based primarily on the review performed for the previous power uprate, since the previous power uprate evaluated the RPV and reactor vessel internals at 108 Mlb_m/hr and 3510 MWt core power. These conditions bound the proposed conditions resulting from increased RTP. The RPV and reactor vessel internals are unaffected.

3.3.1 Reactor Vessel Fracture Toughness

RPV embrittlement is caused by the impingement of high energy neutrons on the RPV wall in the region adjacent to the core (the "beltline" region). Operation at the increased, licensed thermal power level is expected to result in a higher neutron flux, which will increase the integrated fluence over the plant life.

Therefore, the pressure versus temperature curves for Units 1 and 2, shown in Figures 3-1 and 3-2 of Reference 1.6, respectively, are unchanged and remain bounding.

3.3.2

3.3.2.1 Flow and Pressure Differential Loads

Increasing the RTP does not require any increase in evaluated reactor pressure or reactor core flow from values previously analyzed (see Reference 1.6). Since there is no design pressure increase, there is no increase in SRV and seismic loads. The LOCA loads of annulus pressurization and jet reaction were evaluated for the conditions of 108 Mlb_m/hr and 3510 MWt (Reference 1.6) and found to be within the design basis. Since these conditions are not changed in operation at increased RTP, no further analysis on these effects is required. Fuel lift loads were also evaluated at the current and proposed maximum core flow of 108 Mlb_m/hr and found to be acceptable. Therefore, the reactor internal pressure difference increase due to increased RTP is the only issue that needs further evaluation here.

3.3.2.2 Flow-Induced Vibration

Flow-induced vibration was evaluated for increased power level and maximum core flow conditions for the previous power uprate licensing submittal and found to be within design limits. The

magnitude of the flow-induced vibration loads are related to pump speed (pump vane-passing frequency) and total core flow. PPL has found indications of flow induced vibration on the recirculation piping, and the small-bore instrumentation lines (one-inch diameter) are the most susceptible to the vane passing frequency of the pump. PPL has repaired each line as leaks have developed and modifications are in place to correct the frequency response of all lines that have characteristic frequencies of concern. The modifications will be in place prior to operation at increased RTP, but is not related to increased RTP operation, because operation at increased RTP changes neither maximum pump speed nor maximum core flow rate.

3.3.3 Reactor Vessel Integrity (Code Stress Analysis)

The effect of operating at an increased RTP on RPV and components was evaluated to assure that all components continue to comply with the structural requirements of the ASME Boiler and Pressure Vessel Code. The analyses were performed for the design, normal, upset, emergency and faulted conditions.

3.3.3.1 Design Conditions

There is no effect on the reactor pressure vessel and internals design conditions. Therefore, the design stresses are unchanged and the code requirements are still met.

3.3.3.2

3.3.3.3 Emergency and Faulted Conditions

The RPV and RPV component stresses due to emergency and faulted conditions remain unchanged for the increased, licensed core thermal power increase; therefore, the ASME code requirements are still met.

3.4 REACTOR RECIRCULATION SYSTEM

Operation at increased RTP will be accomplished by operating within the bounds of the core power/flow map (Figure 2-1). The percent power on the map corresponds to the percent of the new rated core megawatts thermal power. The cycle-specific reload analyses will consider the full range of the core power and flow operating region. No increase in total core flow is required to support the increase in RTP, however, the increase in steam flow will slightly reduce the recirculation pump inlet temperature and enthalpy, as noted in Tables 1.1 and 1.2.

Since reactor pressure and core flow rate used for evaluation do not change as a result of the increase in RTP, the recirculation pump drive flow stops and the power required by the recirculation pump motors are also unchanged.

3.5 PIPING

3.5.1 Reactor Coolant Piping

Reviews of systems affected by the increased core thermal operating power level have been completed. These systems are a subset of the systems reviewed for the previous power uprate submittal. The piping systems that are part of the reactor coolant pressure boundary and are affected by the increased power/increased flow are the feedwater piping and the main steam piping. The increase in flow rates cause minor increases in transient loads on the piping, but the increase is so small that further evaluation is not warranted.

3.5.2 Balance of Plant Piping

3.5.2.1 Piping Analysis Review

The effects of operation at increased RTP on balance of plant systems were evaluated. Because system pressure and temperature used for evaluation are not increased as a result of the increased operating power level, the effects of increased operating power are confined to the piping systems which see increased flow rate as a result of the uprate. The balance of plant piping systems experiencing an increased flow rate are the condensate system and the turbine extraction steam/feedwater heater piping.

3.5.2.2 Erosion - Corrosion

The increase in RTP and consequent increase in flow rate will cause a minor increase in the erosion/corrosion wear rates in the affected systems. The condensate, condensate cleanup, feedwater, feedwater vent and drain, main steam, extraction steam, moisture separator, main condenser and bypass

steam piping systems will all see slight increases in erosion/corrosion wear rates. These systems are all covered by the erosion/corrosion evaluation and monitoring program at SSES, and all pertinent high wear locations in the piping systems are inspected at an appropriate frequency. Because the expected increase in wear rates is very small in the affected systems and because the systems are already inspected, no change to the SSES erosion/corrosion program is necessary.

3.6 MAIN STEAM LINE FLOW RESTRICTORS

The main steam line flow restrictors limit the loss of coolant from the RPV following a steam line rupture outside containment. This restriction limits the radiological release outside of the drywell prior to the closure of the MSIVs. The restrictors also provide a differential pressure to the main steam flow monitoring system to detect a high steam flow condition and close the MSIVs. The restrictors must withstand the maximum pressure difference expected across the restrictor, following a complete severance of a main steam line (Reference 1.1, Section 5.4.4)

The flow trip is set high enough to permit testing of the MSIVs at rated power without causing an automatic isolation. This requires a setting above 133% of normal flow. The trip must also be set low enough to permit early detection of a steam line break; less than 140% of normal flow. Revised values for the pressure differential limits have been evaluated for the expected 1.4% increase in steam flow, and the current setpoints remain within the acceptable range.

The restrictors were originally designed for a maximum (choked) flow pressure differential of 1375 psig. This is greater than the required flow expected after implementation of increased RTP since the vessel design pressure does not change. Therefore, the restrictors are adequate for use at increased thermal power conditions.

Since the steam flow rate through the flow restrictors increases slightly, there will be a slight but negligible increase in erosion in the flow restrictors. There is no increase in analyzed reactor pressure used for evaluation; therefore, the conditions inside the steam lines are unchanged and the steam line break radiological effects are also unchanged.

3.7 MAIN STEAM ISOLATION VALVES (MSIVs)

The main steam isolation valves (MSIVs) were originally evaluated for steam conditions of 1250 psig at 575°F. Steam conditions for increased, licensed core thermal power operation are not changed from the 1050 psia, 550°F, conditions from previous operation. The MSIVs were also designed for a flow rate across the valve of 3.72 Mlb_m/hr, and the expected flow rate across the valve under the proposed, increased core thermal power conditions is 3.6 Mlb_m/hr. Therefore, the design conditions for the MSIVs bound the expected proposed operating conditions and no further MSIV evaluation is required.

3.8 REACTOR CORE ISOLATION COOLING SYSTEM (RCIC)

The reactor core isolation cooling system (RCIC) provides core cooling when the reactor pressure vessel (RPV) is isolated from the main condenser, and the RPV pressure is greater than the maximum allowable for the initiation of low pressure core cooling systems. Based on a review of the material presented in Section 3.8 of Reference 1.6, the design and performance characteristics of the RCIC system are a function of reactor operating pressure only. Since the plant will continue to operate at a maximum 1050 psia at increased RTP, the RCIC system remains within its analyzed performance basis.

3.9 RESIDUAL HEAT REMOVAL SYSTEM (RHR)

The residual heat removal system (RHR) is designed to restore and maintain the coolant inventory in the reactor vessel and to provide primary system decay heat removal following reactor shutdown for both normal and post-accident conditions. The RHR system is designed to operate in the low pressure coolant injection (LPCI) mode, shutdown cooling mode, suppression pool cooling mode and containment spray cooling mode. The LPCI mode is discussed in Section 4.2.2. The effects of increased, licensed core thermal power operation on the remaining modes is discussed in the following paragraphs.

3.9.1

3.9.2 Suppression Pool Cooling Mode

The functional design basis for the suppression pool cooling mode (SPCM) is to ensure that the suppression pool temperature does not exceed its maximum temperature limit immediately after blow down (Reference 1.1, Section 6.2.2). This objective is met for operation at increased RTP, since the suppression pool temperature analysis (Section 4.1.1) confirms that the suppression pool temperature will remain below its design limit.

3.9.3 Containment Spray Cooling Mode

The containment spray cooling mode of RHR provides water from the suppression pool to spray headers in the drywell and suppression chambers to reduce containment pressure and temperature during post-accident conditions. Since the containment spray function was evaluated at the 3510 MWt initial power conditions, the increased, licensed core thermal power operation has no effect on the containment spray cooling function of the RHR system.

3.10 REACTOR WATER CLEANUP SYSTEM (RWCU)

The reactor water cleanup (RWCU) system maintains high reactor water quality by removing fission products, corrosion products and other soluble and insoluble impurities from the reactor coolant. In addition, the RWCU system provides a means for water removal from the primary system during periods of increasing water volume and is designed to be operated during startup, shutdown and refueling operations, as well as during normal operation.

REFERENCES

3.1 T. A. Caine, "Implementation of Regulatory Guide1.99, Revision 2 for Susquehanna Steam Electric Station Units 1 and 2," General Electric Report SASR 89-11, May 1989.

3.2 Pennsylvania Power & Light Co., Nuclear Procedure NEPM-QA-0901, "Plant Transient and Fatigue Monitoring System, Rev. 0.

4.0 ENGINEERED SAFETY FEATURES

This chapter primarily includes information requested by Regulatory Guide 1.70, Chapter 6 as it applies to operation at increased RTP

4.1 CONTAINMENT SYSTEMS

4.2 EMERGENCY CORE COOLING SYSTEMS (ECCS)

The effect of operating at increased RTP on each of the ECC systems is addressed in this Section. The actual ECCS performance is discussed in Section 4.3.

The ECCS NPSH requirements are discussed in the SSES FSAR (Reference 1.1, Section 6.3.2) and the individual ECCS pump NPSH requirements are conservatively based on a containment pressure of 0 psig and the maximum expected temperature of the pumped fluid. The individual pump is assumed to be operating at the maximum runout flow with the suppression pool temperature at its NPSH limit. As shown in Section 4.1.1 of Reference 1.6, the suppression pool temperature will remain below its NPSH limit. Therefore, operation at increased RTP will not affect compliance with the ECCS pump NPSH requirements.

4.2.1 <u>High Pressure Coolant Injection System (HPCI)</u>

The HPCI system has been evaluated by Siemens Power Corp. (SPC) (References 4.2 and 4.3). The initial conditions for these evaluations were 1050 psia operating reactor pressure and a core power level

of 3510 MWt. Since these initial conditions are not changed as a result of operation at increased RTP, the conclusion that the performance of the HPCI system meets its design criteria is also not changed.

4.2.2 Low Pressure Coolant Injection System (LPCI Mode of RHR)

The LPCI mode of the RHR system has been evaluated by SPC (References 4.2 and 4.3). The initial conditions for these evaluations were the same as the conditions discussed above. Since these initial conditions are unchanged for the operation at increased, licensed core thermal power, the conclusion that the LPCI mode of the RHR system meets its design criteria is also unchanged.

4.2.3 Core Spray System (CS)

The core spray system was evaluated under similar conditions and circumstances as discussed in Section 4.2.1 above. The conclusion that the CS system performance continues to meet its design basis remains unchanged.

4.2.4

4.3

4.4 STANDBY GAS TREATMENT SYSTEM (SGTS)

The SGTS was evaluated for the previous increase in licensed power level (from 3293 to 3441 MWt) and it was found that the SGTS would maintain its design basis performance under those conditions (Reference 1.1, Section 6.5.1). Since this submittal supports a licensed power increase of 1.4% (from 3441 to 3489 MWt), the accident radiological source term, which must be handled by SGTS, would also increase by 1.4%. In addition, the analysis for radioactivity release from the station was completed at an initial power level of 3616 MWt, a value of just over 105% of the previous licensed power level. The previous evaluation concluded that the SGTS has a 500% excess capacity. Therefore, operation at increased RTP would have no impact on the ability of the SGTS to meet its intended design function.

4.5 OTHER ENGINEERED SAFETY FUNCTION (ESF) SYSTEMS

4.5.1 Main Steam Isolation Valve Leakage Control System (MSIV-LCS)

The MSIV-LCS was eliminated from both SSES Units 1 and 2 (Reference 4.4).

4.5.2 Post-LOCA Combustible Gas Control

The SSES units have nitrogen-inerted containments. A combustible gas mixture is prevented in containment by limiting either the hydrogen or oxygen concentration. Hydrogen recombiners are used to control post-LOCA hydrogen and oxygen concentrations, using the recombiners to cause hydrogen and oxygen to react, forming water until either the available hydrogen or oxygen is consumed. The containment atmosphere is also mixed by the drywell unit-cooler fans and recirculation fans to prevent excess local hydrogen concentrations, as discussed in Section 6.6.1.

The inerted containment design was retained and included in the design evaluations even though worst-case hydrogen concentrations for the original design did not require inerting. The worst-case concentration of hydrogen is 3.5 volume percent (Reference 1.1, Section 6.2.5.3). The inerted containment provides additional conservatism against combustible gas concentration issues.

Radiolysis is proportional to reactor power and decay heat. However, it should also be noted that the analysis basis used in the determination of hydrogen generated by radiolysis was a power level of

3510 MWt (Reference 1.1, Table 6.2-13); therefore, the volume of hydrogen used for analysis purposes will not change as a result of operation at increased, licensed core thermal power. Thus, the evaluation of the hydrogen recombiners and other support systems is not impacted by this licensing submittal.

4.5.3 Emergency Cooling Water Systems

The Susquehanna emergency cooling water systems are the emergency service water (ESW) system and the residual heat removal service water system (RHRSW). These systems are discussed in Sections 6.4.1.1.1 and 6.4.1.1.2, respectively.

4.5.4 Emergency Core Cooling Auxiliary Systems

Emergency core cooling auxiliary systems are discussed in Chapter 6.

4.5.5 Main Control Room Atmosphere Control System

The main control room atmosphere control system is a subsystem of the control structure HVAC, which is discussed in Section 6.6.3.

4.5.6 Standby Power System

The SSES standby power system includes the 4 kV safety-related busses and the emergency diesel generators, which are discussed in Sections 6.1.2 and 6.1.3.

REFERENCES

4.1 U. S. Nuclear Regulatory Commission, "Suppression Pool Temperature Limits for BWR Containments," NUREG-0783, November 1981.

4.2 EMF-96-160 (P), Rev. 0, "LOCA Break Spectrum Analysis for Susquehanna Units 1 and 2 Using 1993 EXEM BWR Evaluation Model," April 1997.

4.3 EMF-96-161 (P), Rev. 0, "Susquehanna LOCA-ECCS Analysis MAPLHGR Limits for SQB-8 ATRIUM™-10 Fuel Using 1993 EXEM BWR Evaluation Model," April 1997.

4.4 Letter, C. Poslusny, USNRC to R. G. Byram, PPL, Susquehanna Steam Electric Station, Units 1 and 2, (TAC Nos. M91013 and M91014), August 1995.

5.0 INSTRUMENTATION AND CONTROL

This chapter primarily includes information requested by Regulatory Guide 1.70; Chapter 7 as it applies to operation at increased RTP.

5.1 NUCLEAR STEAM SUPPLY SYSTEM (NSSS) I&C

Changes in process variables and their effects on instrument setpoints were evaluated for operation at increased RTP to determine any related changes.

5.1.1 Neutron Monitoring

The average power range monitor (APRM) power signals will be rescaled to the uprated power and the percentage setpoints will not change.

5.1.2 NSSS Instrument and Valve Setpoints

Instrument setpoints are based on both plant operating experience and conservative licensing analyses to preclude inadvertent initiation while assuring adequate allowances for instrument accuracy, process effects, calibrations and drift relative to the safety analysis analytical limit. For operation at increased RTP, no changes are necessary to either the reactor pressure used for evaluation or the maximum core flow rate. Therefore, the setpoints that are based on RPV steam dome pressure or maximum core flow rate will not change.

5.1.2.1

5.1.2.2 High Pressure ATWS Recirculation Pump Trip and ARI

The anticipated transient without scram recirculation pump trip (ATWS RPT) and alternate rod injection (ARI) are provided to trip the reactor recirculation pumps and insert control rods during plant transients associated with increases in RPV dome pressure and low RPV water level. The ATWS RPT is designed to provide negative reactivity by reducing core flow during the initial portion of an ATWS, and by inserting control rods.

The major consideration for the ATWS RPT pressure setpoint is the increase in peak pressure during a hypothesized ATWS event because of the higher initial power level. Since the reactor pressure used for evaluation and the safety/relief valve setpoints are not changed in operating at increased RTP, it is expected that the cycle specific ATWS analysis will show that the peak, calculated steam dome pressure in an ATWS event will remain under the 1500 psig limit for ATWS. The cycle specific ATWS analysis is performed as part of the reload licensing process for each reload cycle. Therefore, the current ATWS RPT analytical limit is acceptable for these conditions.

The ARI function is not affected by operation at increased, licensed core thermal power.

5.1.2.3

5.1.2.4

5.1.2.5

5.1.2.6

5.1.2.7

5.1.2.8 Turbine Stop Valve Closure and Turbine Control Valve Fast Closure Scram Bypass

The first stage turbine pressure setpoint is chosen to allow operational margin so that scrams can be avoided by transferring steam to the turbine bypass system during turbine-generator trips at low power. An evaluation is required to assure that, when the reactor scrams on turbine stop valve and turbine control valve closure are bypassed, applicable licensing limits are not exceeded (for example, the MCPR safety limit is not violated). The evaluation for operation at increased RTP will be performed as part of the reload licensing analyses that support the increased core thermal power operation.

5.2 BALANCE OF PLANT I&C

Operation at the increased RTP has little effect on the balance-of-plant (BOP) instrumentation and control devices. The control valves and instrumentation have sufficient range at uprated conditions, and no setpoint changes are anticipated.

5.2.1

5.2.1.1

5.2.1.2

5.2.1.3

5.2.2 Pressure Control System

The objective of the pressure control system (PCS) is to provide fast and stable response to pressure and steam flow disturbances so that the reactor pressure is controlled within its allowed high and low limits. The PCS consists of the pressure regulation system, turbine control valve system and steam bypass system.

5.2.3 Electrohydraulic Turbine Control System (EHC)

No modifications to the turbine control valves or the turbine bypass valves are required for operation at the increased RTP conditions. Normal manual operator setpoint control is used to establish the new pressure setpoint using modified operating procedures. No control valve stability problems associated with the licensed core thermal power conditions are expected.

5.2.4 Systematic Survey of Instrument and Valve Setpoints

The previous sections of this chapter describe the potential effects on instrument and safety valve setpoints in operating at the increased RTP level that resulted from evaluations of the reactor, turbine and NSS systems. Instruments and safety valves were also examined in each of the other system evaluations summarized throughout this report; and the appropriate changes were made to the setpoints of engineered safeguards, power conversions and other BOP systems.

A review of the list of system instruments with either a range of a setpoint which may be affected by the operation at increased, licensed core thermal power, plus the system safety valves, was conducted. These lists were produced from plant instrument and valve lists, for each system possibly affected by increased core thermal power operation, in order to assure an exhaustive review of safety valves and instruments.

REFERENCES

5.1 Letter from Victor Nerses, NRC to R. G. Byram, PP&L, "Susquehanna Steam Electric Station, Units 1 and 2 – NRC Staff Evaluation and Issuance of Amendment Regarding Main Steam Line Radiation Monitor Setpoint Increase and Offgas System Design Basis Change (TAC Nos. MA1366, MA1367, MA1014 and MA1015)," October 13, 1998.

5.2 Caldon, Inc. LEFM✓™ System Procurement Specification, "Feedwater Flow Measurement System Suitable for Use in Conjunction With a 1.4% Thermal Power Uprate, Specification SP31, May, 2000.

6.0 ELECTRICAL POWER AND AUXILIARY SYSTEMS

This chapter primarily includes information requested by Regulatory Guide 1.70; Chapters 8 and 9, as they apply to operation at increased RTP.

6.1 AC POWER

6.1.1 Offsite Power System

The offsite power system includes bulk transmission facilities to customers and interconnections to other utilities. Unit 2 supplies the 500 kV transmission system and Unit 1 supplies the 230 kV transmission system. The offsite power system also includes two independent connections to provide reliable power sources for plant auxiliary loads and the engineered safety features of both units (the "offsite power source"). The offsite power source is supplied through two startup transformers that are common to both units, and is designed so that no single disturbance in the bulk power grid will cause complete loss of offsite power. Neither the bulk power transmission function nor the offsite power source function is affected by operation at increased RTP.

A feasibility study of the bulk power transmission system has determined that all switchyard equipment, with the exception of the Transformer #21 230 kV line 3000 amp gang-operated circuit breakers in the Susquehanna 230 kV switchyard, can support operation at increased RTP. Transformer #21 interconnects the 500 kV bulk power transmission system with the 230 kV bulk power transmission system. Operation at increased RTP will not impact thermal loading or voltage, but will affect the stability of the grid under a particular faulted condition discussed in the FSAR (Reference 1.1, Table 8.2-1). The particular Fault Test Case is Case N-10:

> "3 phase fault at Susquehanna 230 kV on the 500/230 kV transformer with stuck west bus breaker. Primary clearing at remote terminal (Susquehanna 500 kV Switchyard). Delayed clearing at Susquehanna 230 (lose Stanton-Susquehanna #2 230 kV line)."

The grid is not required to be stable for this particular fault, per the design basis Mid Atlantic Area reliability Council (MAAC) standards, as noted in the table. However, increasing the power level such that this case is now unstable constitutes a change to the SSES stability results as previously presented and reviewed, and represented in the table. The ability of the bulk power transmission system to withstand a three phase fault with delayed clearing is not explicitly required for compliance with the

MAAC reliability criteria. Section IV of the MAAC criteria defines that the system must be able to withstand a three phase fault with normal clearing and a single phase to ground fault with delayed clearing. A three phase fault with delayed clearing is considered to be a maximum credible disturbance (MCD) and tests of the bulk power transmission system to assess the effect of MCDs are required under Section V of the MAAC criteria. MCD condition tests are performed to assess the robustness of the bulk power system, and may or may not be mitigated depending on the severity of the consequences and the cost to address the condition. Based on the magnitude of the consequences and minimal costs to mitigate this particular MCD (Case N-10), the involved circuit breakers will be replaced to minimize the probability and consequences of such an event.

The remedy for this situation is to replace the Transformer #21 230 kV line 3000 amp gang-operated circuit breakers in the Susquehanna 230 kV switchyard with 3000 amp independent pole operated circuit breakers. When this replacement is made, the Case N-10 Fault Test results in a stable transient and the evaluation in Table 8.2-1 of the FSAR remains unchanged from the existing FSAR. This replacement of the two circuit breakers will be performed prior to the implementation of increased RTP operation for Unit 1.

6.1.2 Onsite Power Distribution

The onsite AC power produced by the unit main generator supplies:

1) Auxiliary Transformers for onsite power distribution
2) Main Step-Up Transformers for offsite distribution

During plant startup, shutdown and post-shutdown, offsite sources supply the onsite distribution systems through the two startup transformers. The review in this section includes all transformers, busses, breakers, components and instrumentation that support the onsite power system, isolated phase bus ducts, main step-up transformers and startup transformers. The main generator study is addressed in Section 7.1 of this report. The diesel generators are reviewed in Section 6.1.3 and the offsite power source is reviewed in Section 6.1.1.

Station loads under normal operation and distribution conditions are computed based on equipment nameplate data. Operation at increased RTP is achieved by using existing equipment operating at or below the nameplate rating. Therefore, under normal conditions, the electrical supply and distribution components (switchgear, MCCs, cables, etc.) are adequate.

Station loads under emergency operation and distribution conditions using emergency diesel generators (EDGs) are based on equipment nameplate data except for emergency service water (ESW) and residual heat removal (RHR) pumps where the operating point is used, and core spray (CS) where high-flow brake horsepower (BHP) is used. Operation at increased RTP is achieved by using existing equipment operating at or below the nameplate rating and within the calculated BHP for the stated pumps. Therefore, under emergency conditions, the electrical supply and distribution components are adequate.

Safety-related load changes associated with increased RTP operation have been reviewed against the existing plant voltage model and have been found to be bounded. Ongoing work to upgrade the plant voltage model will incorporate these plant load changes.

The unit main generator supplies power to the auxiliary transformers for the onsite AC distribution system. The 13.8 kV startup busses take power from the auxiliary transformers or the startup transformers for onsite distribution to class 1E and non-class 1E loads. The class 1E system distributes power from the startup busses through the engineered safeguard transformers to the safety-related loads. The non-class 1E system provides power from the startup busses for non-safety related loads. All electrical loads that increased power resulting from operating at increased licensed core thermal power were evaluated to determine the effects of the electrical distribution systems. The evaluation found that all electrical load increases can be supported by the present electrical distribution system configurations.

The unit main generator supplies power to the main step-up transformer for offsite distribution. The transformer steps up the 24 kV generator output to 230 kV for Unit 1 and 500 kV for Unit 2. PPL evaluated the Unit 1 transformers and found them to be capable to handle the increased power necessary to support operation at increased RTP. The Unit 2 transformers were replaced prior to operation at 3441 MWt, and will also support operation at increased RTP.

6.1.3 Diesel Generators and Auxiliaries

An evaluation confirmed that the loads on the 4kV 1E busses will remain within the original design capacity of the emergency diesel generators (EDGs) and their auxiliaries. The study also confirmed that bus voltages and under-voltage trip setpoints will not change and that effects on the load sequence will

not require changes in operating modes, control logic or EDG start timing. The EDGs and their auxiliaries will therefore continue to be adequate for operation at increased RTP.

The design operating load on each 4 kV 1E bus was determined from rated nameplate data for equipment other than 4 kV pump motors, and from pump brake horsepower based on design flows for 4 kV pump motor loads. The evaluation confirmed that, at increased, licensed core thermal power, the 4 kV pumps will operate at or below design flows and other equipment will operate at or below nameplate ratings. Increased, licensed core thermal power will not require additional equipment, earlier operation of equipment or operating durations longer than those previously analyzed. The EDG load profiles summarized in the FSAR (Reference 1.1; Tables 8.3-1 through 8.35a) will bound operation at increased, licensed core thermal power. In addition, FSAR Tables 8.3-2 through 8.3.5a show that the EDGs are loaded to less than their continuous ratings at all times following a design basis accident in one unit combined with a forced shutdown in the other unit.

Operation at increased RTP will not change EDG initiation logic. The operating voltage of the 4 kV 1E busses and the bus under-voltage trip setpoint for EDG initiation will not change. Low reactor vessel water level or high drywell pressure will still initiate the EDGs. Rated speed, start sequence and output voltage will not change. The accident analyses for operation at increased, licensed core thermal power do not assume an EDG start time of less than ten seconds. Therefore, the ten second start time of the EDGs will not be affected and the starting air system pressure and volume will not require change.

EDG building ventilation is discussed in Section 6.6.7. The evaluations performed for the current operating condition bound operation at increased RTP and the effects on the EDGs and auxiliary system components is not changed from those evaluations.

6.2 DC POWER

The DC systems for each generating unit consist of 4 125 V DC subsystems, two 250 V DC subsystems and two ± 24 V DC subsystems; and are divided into class 1E and non-class 1E systems. In addition, the E EDG building has a 125 V DC subsystem. The DC power systems consist of battery banks, battery chargers, load centers, distribution panels and motor control centers. The DC power system is safety related and is not affected by operation at increased RTP.

Operation at increased RTP will not increase the design loads or the operation of the DC systems, and these systems do not directly support any subsystem function which may be affected by increased RTP operation. The station blackout (SBO) evaluation for increased RTP (Section 9.3.2) includes the effect on the DC systems.

6.3 FUEL POOL COOLING AND CLEANUP SYSTEM

Fuel pool storage capacity will not be changed as a result of operation at increased RTP. In addition, increased RTP operation will not require modification of the fuel pool cooling and cleanup system, its filter demineralizer system, the service water system or the fuel pool cooling assist mode of RHR. Cycle-specific calculations assure that cooling loads on the normal pool cooling system and fuel pool cooling assist mode of RHR will remain within their design capacities. The emergency service water (ESW) system will provide the necessary makeup flow to the fuel to maintain level, if required, and normal makeup requirements are not significant. Increased, licensed core thermal power operation will not adversely affect fuel pool water chemistry. The fuel pool cooling and cleanup system will therefore be adequate for all required functions under increased, licensed core thermal power operation.

Operation at increased RTP will increase the decay heat loads required on the fuel pool cooling system and the RHR fuel pool cooling assist mode. Cycle-specific and bounding calculations verify that the decay heat in the fuel pool will remain within the design heat removal capacity of the heat removal systems for both normal and emergency loads. The fuel pool cooling system has a design capacity of 13.2×10^6 Btu/hr. The RHR fuel pool cooling assist mode has a design capacity of 42.4×10^6 Btu/hr, a margin of 6.7×10^6 Btu/hr over the increased design emergency heat load (35.201×10^6 Btu/hr, from Reference 1.1, Table 9.1-2f, multiplied by 1.014) of 35.7×10^6 Btu/hr. Therefore, a margin of ~ 16% remains on the fuel pool cooling system design with the implementation of operation at increased RTP.

The makeup required from emergency service water to replace evaporation and boil-off on loss of the fuel pool cooling system is less than the 35 gpm assumed for system design capacity. A flow analysis of the ESW system and the ultimate heat sink (UHS) confirmed that the required flow will be available for operation at increased RTP.

6.4 COOLING WATER SYSTEMS

6.4.1 Service Water Systems

6.4.1.1 Safety-Related Service Water Systems

6.4.1.1.1 Emergency Service Water System

The emergency service water (ESW) system removes heat from HVAC coolers, emergency diesel generators (EDGs), emergency core cooling (ECCS) and engineered safety feature (ESF) components and other equipment required to operate under normal and accidents conditions, including loss of offsite power (LOOP) and loss of coolant accident (LOCA) conditions. The system draws water from the ultimate heat sink (UHS) spray pond and returns the water, after being heated in the heat exchangers, by way of a spray network that dissipates the heat to the atmosphere. The system is divided into two loops, each sized for the combination of Unit 1 and Unit 2 design basis accident (DBA) cooling and flow requirements. The system is safety related.

The design requirements for the ESW system have been reviewed and found to be based on the 3510 MWt initial condition assumed for design basis accident evaluation. Since operation at increased, licensed core thermal power does not change the 3510 MWt design basis initial condition, the design temperature of 97°F and the ESW system flow rates are not affected by the increase in operating power level.

6.4.1.1.2 Residual Heat Removal Service Water (RHRSW) System

The RHRSW system provides a safety-related cooling water source for the RHR system under normal or post-accident conditions. The system pumps water from the UHS spray pond through the RHR heat exchangers and the water returns to the spray pond via the spray network. The system may also be used to flood the reactor core or primary containment following an accident, if necessary.

Operation at increased, licensed core thermal power increases the normal loads on the RHRSW system (that is, for normal shutdown conditions) in an amount proportional to the operating power increase. However, the design heat loads for the RHRSW system are based on transient and accident situations when the initial power is assumed to be 3510 MWt. Since operation at increased, licensed core thermal power does not change the operating power level assumed at transient or accident situations, the increased operating core thermal power level has no effect on the RHRSW system design.

6.4.1.1.3 Ultimate Heat Sink (UHS)

The UHS provides a safety-related cooling water source for the ESW system and the RHRSW system during testing, normal shutdown and accident conditions (see Sections 6.4.1.1.1 and 6.4.1.1.2). The UHS consists of an 8 acre, 25 million gallon concrete-lined spray pond. Following a design accident, the UHS provides enough cooling water at or below the ESW and RHRSW design temperature for a minimum of 30 days, without makeup.

The UHS has been evaluated for both the design basis minimum heat transfer case (MHT, Reference 1.1, Section 9.2.7) and the design basis maximum water loss case (MWL). These analyses were performed using the design basis decay heat load consistent with core power operation at 3510 MWt for a period of several years. The analyses confirm that the UHS can meet its design basis function of providing a source of water, at 97°F or less, for a period of thirty days without makeup. In addition, the analyses confirm that the Technical Specifications for normal pond level and temperature remain adequate to assure operability of the UHS. Operation at increased RTP has no effect on these analyses or the conclusions of these analyses.

6.4.1.2 Non-Safety-Related Service Water System

The service water (SW) system has no safety-related function and is designed to continuously supply cooling water to various heat exchangers in the turbine, reactor and radwaste buildings during normal plant operation.

An evaluation confirmed that the service water system will support increase RTP operation with no equipment or setpoint changes. The design service water heat load bounds the proposed operating conditions. Therefore, the cooling tower is able to dissipate the service water heat load at increased, licensed core thermal power conditions without affecting the existing design service water temperature of 95°F. Increased core thermal power operation will require a slight increase in flow that is well within the capability of the service water pumps.

6.4.2 Reactor Building Closed Cooling Water (RBCCW) System

The RBCCW system has no safety-related function. The purpose of the RBCCW system is to cool various auxiliary plant components in the reactor and radwaste buildings during normal and loss-of-offsite-power (LOOP) conditions.

During normal operation the RBCCW heat exchanger is cooled by the service water system (Section 6.4.1.2). The components cooled by the RBCCW system during normal operation will remain within their design conditions following the implementation of increased RTP and, therefore, will not increase their heat loads on this system. The RBCCW system will therefore be adequate for normal operation at the increased core thermal power level.

Upon loss-of-offsite-power (without a coincident LOCA), the RBCCW heat exchangers can be manually switched from the service water system to the ESW system. The RBCCW pumps, powered from the emergency diesel generators (EDGs), provide cooling water to the reactor recirculation pump coolers, RWCU pump coolers and the drywell coolers. This function of the RBCCW system was analyzed under previous power uprate conditions and found to be acceptable. The design heat loads for the systems cooled by RBCCW under LOOP conditions do not change under increased RTP conditions, therefore the conclusion that the RBCCW system can accommodate post-LOOP operation is unchanged under increased RTP operation.

6.4.3 Turbine Building Closed Cooling Water (TBCCW) System

The TBCCW system supplies cooling water to auxiliary plant equipment in the turbine building. Neither the system nor any of its loads are safety-related. An evaluation confirmed that the system will support operation of these components following the implementation of increased RTP operation with no equipment or setpoint changes.

The system loads include the control rod drive (CRD) pump bearing and gear oil coolers and the condensate pump motor bearing coolers. Since reactor pressure used for evaluation does not change under increased RTP operation, the CRD pump bearing and gear oil cooler heat loads are unchanged, but the increased feedwater flow required will slightly increase the heat loads on the condensate pump motor bearing cooler. The increase in condensate pump motor bearing cooler heat load is small and well within the capacity of the cooler.

During normal operation the TBCCW heat exchanger is cooled by the service water system (Section 6.4.1.2). The components supplied by the TBCCW system during normal operation are either unaffected by operation at increased, licensed core thermal power or will remain within the design capabilities after the implementation of increased, licensed core thermal power. The design heat load on the TBCCW system will remain unchanged and well within the design capability of the TBCCW heat exchangers. The TBCCW system will therefore be adequate for normal operation at increased, RTP.

After a LOOP, the TBCCW pump can be loaded onto an EDG supplied bus and the TBCCW heat exchanger can be cooled by the ESW system to permit cooling of the CRD pump bearing and gear oil coolers. This is the only design heat load on the TBCCW system following a LOOP. This function is not required, but is provided to prevent CRD pump damage. This function will not be affected by operation at increased, licensed core thermal power because the TBCCW heat exchanger has sufficient excess capacity to handle the heat load.

6.4.4 Gaseous Radwaste Recombiner Closed Cooling Water (GRRCCW) System

The gaseous radwaste recombiner closed cooling water system (GRRCCW) system provides cooling for the off-gas recombiner, condensate cooler and motive steam jet condenser of the off-gas system. The GRRCCW system is required to operate only during normal plant operation and has no safety related function.

Operation at increased RTP will increase the heat loads from the offgas recombiner condenser, motive steam jet condenser and condensate of the offgas system by approximately the same percentage. An evaluation of the offgas system found that the offgas system will remain within its original design capacities (see Sections 7.8 and 8.2). The increase in offgas system heat load on the GRRCCW system is also within the original GRRCCW capacity, which will be adequate for operation at the increased power level. The increased heat load will increase expected GRRCCW temperatures by much less than 1°F and will maintain adequate margin to design limits. The design heat load on the GRRCCW system will remain the same, which is well within the capacity of the GRRCCW heat exchangers and the service water system.

System chemistry is unaffected by the increased operating power since system design volume, flow, temperature and reactor pressure used for evaluation will not be changed.

6.4.5 Ultimate Heat Sink (UHS)

The UHS for safety-related cooling systems is described in Section 6.4.1.1.3.

6.4.6 River Water Makeup

The river water makeup system consists of four river water pumps and their screens; the intake structure and pump house; along with piping, valves and controls. It supplies raw water to compensate for cooling tower and spray pond blowdown and evaporation, and for makeup to the plant water treatment and storage systems and has no safety-related function.

Operation at increased RTP will increase the cooling tower heat rejection rate and thereby the cooling tower evaporative losses. This increase in cooling tower heat rejection rate will also require increased blowdown to maintain circulating water quality. The other makeup and incidental uses will not change significantly. The nominal design flow will increase from the present value of 37,925 gpm to 38,500 gpm.

The original system design capacity for the river water makeup system was 40,500 gpm, with three of the four system pumps running. Therefore, the expected, nominal design flow at increased power conditions remains well within the design basis, except under infrequent, maximum meteorological conditions. During these peak demand periods increased power operation will increase the maximum system design flow from the current value of 40,695 gpm to a value of 41,265 gpm. Previous evaluations have concluded that the fourth river water makeup pump can be operated to maintain margin during these infrequent periods, without adversely affecting HVAC performance, electrical distribution, system piping or travelling screen operation. The need for four-pump operation will be quantified after the implementation of operation at increased RTP and a complete evaluation, with testing, will be performed, if required. Therefore, the river water makeup system will support plant operation at increased RTP.

6.4.7 Chilled Water Systems

6.4.7.1 Reactor Building Chilled Water System

During normal plant operation the reactor building chilled water system (RBCW) supplies chilled water to various reactor building HVAC, drywell HVAC and equipment loads. The system has no safety-related functions, except its containment isolation valves.

The original calculated design load on the RBCW system was less than 600 tons and the system design provided two 100% capacity (600 ton) chillers. However, actual peak loads during hot weather conditions often exceed 600 tons, and a single chiller is operated above its design rating.

Operation at increased RTP is not expected to increase the drywell cooling system load on the RBCW system over the previous evaluation. Neither the reactor operating pressure nor the maximum recirculation drive flow (and consequent recirculation pump motor cooler loads) are expected to increase over values previously analyzed. System operating strategies, including tandem chiller operation, have been developed to meet peak heat load demands and will be implemented as necessary to support operation at increased RTP. Therefore, adequate margin exists in the RBCW to maintain current drywell and reactor building conditions following the implementation of increased, RTP.

6.4.7.2 Control Structure Chilled Water System

The control structure chilled water system is included in the description of the control structure HVAC systems in Section 6.6.3.

6.4.7.3 Radwaste Building Chilled Water System

The radwaste building chilled water system is included in the description of the radwaste building HVAC system in Section 6.6.4.

6.4.7.4 Turbine Building Chilled Water System

The turbine building chilled water system is included in the description of the turbine building HVAC system in Section 6.6.5.

6.5 STANDBY LIQUID CONTROL SYSTEM (SLCS)

The ability of the SLCS boron solution to be delivered to the reactor is not a direct function of core thermal power, and therefore is not affected by operation at increased RTP. SLCS shutdown capability is evaluated for each reload cycle.

6.6 HVAC SYSTEMS

6.6.1 Drywell Cooling System

The drywell cooling system will support operation at increased RTP without exceeding the operating and design temperature limits for equipment in the drywell.

Selected cooler and control rod drive area fans provide post-LOCA mixing of the drywell atmosphere to prevent explosive concentrations of hydrogen. This non-cooling function is safety-related. Accident conditions after the implementation of increased, licensed core thermal power operation may produce slightly higher concentrations of hydrogen in the drywell. The drywell cooling system mixing capacity remains adequate for increased core thermal power operation.

Drywell cooling during normal, transient and shutdown operation is not safety-related, but is required to maintain drywell temperatures below the limits, specified by the Technical Specifications. Operation at increased RTP will not have a significant impact on drywell heat load because the reactor pressure used for evaluation (and corresponding RPV saturated temperature, the driving force for heat transfer to the drywell airspace) and recirculation pump motor heat loads will not change significantly. Therefore, the capacity of the drywell cooling system continues to be adequate, with margin, to support plant operation at the increased power level.

Chilled water for the drywell cooling system is supplied by the RBCW system. The peak-load operating strategies planned for the RBCW system, discussed in Section 6.4.7.1, will ensure that adequate chilled water capacity will be available to meet drywell cooling loads without exceeding operating and design temperature limits in the drywell.

6.6.2 Reactor Building HVAC

The reactor building HVAC system will support operation at increased RTP without exceeding any of the design limits for the safety-related equipment in the reactor building, for both normal and accident conditions.

Heat transmission from the drywell is a significant load on the reactor building during normal operation. Since the heat load on the drywell is not significantly changed under increased, licensed core thermal power operation and the drywell cooling system has been evaluated as adequate for the increased core thermal power conditions, the Technical Specification limit on drywell temperature will not require change (see Section 6.6.1). Since the maximum drywell temperature will not change, the heat load on the reactor building HVAC resulting from drywell heat transmission will not change. Therefore, operation at increased RTP does not affect the reactor building HVAC system.

Since the reactor pressure used for evaluation (and corresponding RPV saturated temperature) does not increase with the implementation of operation at increased RTP, conditions in other areas of the reactor building, including the HPCI and RCIC rooms, the main steam and feedwater pipe tunnels, and the RHR penetration area. Therefore, there is no increase in maximum design temperatures in these areas.

Maximum temperatures inside primary containment and the reactor building were evaluated at the design power level of 3510 MWt (see Section 4.1.1). Therefore, operation at increased, licensed core thermal power does not affect any post-accident temperature analysis bounds.

6.6.3 Control Structure HVAC System

Evaluations have confirmed that operation at increased RTP has a negligible effect on control structure HVAC. Control structure HVAC includes various heating, ventilating, isolating and exhaust systems along with the control structure chilled water system.

Operation at increased RTP is evaluated to have a negligible effect on reactor building temperature (see Section 6.6.2); therefore, there is no effect on control structure HVAC from heat transmission through walls in the control structure common to the reactor building. The heat load on the control structure chilled water system from the emergency switchgear room in Unit 1 may increase slightly as a result of increased power operation, but such an expectedly small increase will not have an effect on the control structure HVAC system.

The ESW system cools the control structure chillers during loss-of-offsite-power conditions. The ESW evaluation (Section 6.4.1.1.1) shows that the emergency heat loading on the control structure chillers are within the capability of the ESW system and are essentially unchanged from the evaluation for the previous power uprate submittal.

The maximum chilled water cooling load is expected to occur under post-accident conditions. The evaluations performed for the previous power uprate conditions used the accident power level of 3510 MWt for this evaluation. Consequently, operating at increased RTP has no effect on the accident heat loads or the system response to the accident.

The turbine building chilled water system (TBCW) supplies the control structure HVAC access control and lab area unit coolers. These coolers are not affected by core power level, and consequently, these coolers will not impose any additional heat loads on the TBCW system as a result of operation at the increased RTP.

The Unit 2 reactor building emergency switchgear and load center is cooled by a separate direct exchange (DX) cooling system. The heat load on the DX system is dependent on the surrounding reactor building airspace temperature. The reactor building airspace temperature is not changed, therefore, there is no effect on DX unit performance resulting from increased RTP.

6.6.4 Radwaste Building HVAC System

The radwaste building HVAC system serves the radwaste building, which is common to both Units 1 and 2. The radwaste building HVAC system maintains a controlled environment for equipment and personnel, contains particulate contamination and gaseous effluents from radwaste processing and controls their release.

An evaluation confirmed that operation at the increased RTP has no significant effects on the radwaste building HVAC system, including the radwaste building chilled water system. The evaluation included possible effects from increased processing requirements of the liquid and solid radwaste systems and the offgas system (Sections 8.1, 8.2 and 8.3); increased spent resin burden from the condensate demineralizers (Section 7.6.3); control of discharges via the turbine building ventilation system (Section 6.6.5); and effects on supporting systems.

6.6.5 Turbine Building Ventilation System

The ventilation system in each turbine building consists of ventilation and chilled water systems to provide a controlled environment for equipment and personnel and to contain and control particulate contamination and gaseous effluents from the turbine cycle. The filtered exhaust subsystems discharge turbine building, radwaste building, service and administration shop and control building exhausts to the respective turbine building vent stacks. Exhaust from potentially contaminated areas is processed through HEPA and charcoal filters prior to discharge.

An evaluation confirmed that operation at the increased RTP has no significant effects on the turbine building HVAC system, including the turbine building chilled water system. In addition, operation at the increased RTP will not require any changes to system capacities, because the heat loads on the turbine building HVAC system and its interfacing systems will remain within original system design capacities. The evaluation included possible effects from increased heat loads in the turbine building; effects from the turbine building, radwaste building, service and administration shop and control building exhaust and effects on supporting systems.

6.6.6 Engineered Safeguards Service Water (ESSW) Pump House Heating and Ventilation System

Each compartment in the ESSW pump house contains two ESW and two RHRSW pumps. Each pump is provided with a separate ventilation system, powered from the same division of the Class 1E power system which serves the ESW and RHRSW pumps in that compartment, to assure operability of the pumps during loss-of-offsite-power.

The engineered safeguards service water pump house ventilation system (ESSWP H&V system) maintains the environment of the pump house within limits required by personnel and equipment, for all modes of operation, including conditions of design basis accidents, loss-of-offsite-power and seismic events.

An evaluation confirmed that the ESSWP H&V system functions and performance will not be affected by operation at increased RTP. In addition, the ESSWP H&V system will not be affected by its interfacing systems nor will the system have any effect on its supporting systems as a result of operation at the

increased core thermal power level. The design temperature limits of the building (104°F maximum and 60° minimum) will not change as a result of increased core thermal power operation and the heat loads are bounded by the original design heat load calculations.

6.6.7 Emergency Diesel Generator (EDG) Building Ventilation System

The heating and ventilation (H&V) subsystem for each EDG is powered from the EDG whose enclosure it serves, to assure operability of the system and the affected EDG during the loss-of-offsite-power event. The system is safety-related and must satisfy the design environment requirements of the EDG building during all modes of operation; including design basis accidents, seismic events and loss-of-offsite-power. The ventilation system also prevents the accumulation of an explosive concentration of fuel oil vapors, and of hydrogen in the battery room in the "E" EDG building basement.

An evaluation confirmed that the EDG building H&V system functions and performance will not be affected by operation at the increased, licensed core thermal power level. In addition, the EDG building H&V system will neither affect or be affected by any of its supporting systems as a result of operating at increased RTP.

The design temperature limits for the EDG buildings have not been changed by operation at the increased, licensed core thermal power level. The heat loads within the EDG buildings are affected only by the heat loads coming from the EDGs themselves and not by core thermal power level. The rate of hydrogen and fuel oil vapor generation are not influenced by core thermal power level. Therefore, operation at the increased, licensed thermal power level has no effect on these parameters.

6.7 FIRE PROTECTION SYSTEM

Fire protection systems are not affected by changes in core operating power level. Operation at increased RTP will not change suppression or detection systems, the physical plant configuration or combustible loads. Systems, equipment and operator actions required to respond to a fire and achieve and maintain safe cold shutdown are not affected by core thermal power level and will not change with the implementation of operation at increased RTP.

6.8 BALANCE OF PLANT PIPING

Balance of plant piping is addressed in Section 3.5.2.

7.0 POWER CONVERSION SYSTEMS

This chapter primarily includes information requested by Regulatory Guide 1.70; Chapter 10 as it applies to operation at increased RTP.

7.1 TURBINE GENERATOR

The turbine generator was designed to operate at valve wide open (VWO) steam flow conditions. This VWO steam flow rate is established by the turbine vendor as 14.6 Mlb_m/hr and this flow rate is not affected by operation at increased RTP. Startup testing following the installation of the LEFM✓™ will assure that adequate turbine control valve margin is maintained at increased RTP conditions.

7.1.1. Generator Stator Cooling System

The stator cooling system was designed for VWO conditions discussed above. The evaluation determined that no modifications are necessary for operations at increased RTP conditions, and since the design conditions are unchanged, the conclusions are also unchanged.

7.1.2 Generator Hydrogen Cooling System

The hydrogen cooling system was designed to handle the VWO conditions discussed above. PPL performed an evaluation on the hydrogen cooling system and determined that no modifications are required, therefore, the generator hydrogen cooling system will support operation at increased RTP.

7.2 MAIN AND BYPASS STEAM SYSTEMS

The safety-related sections of the main steam and bypass systems are discussed in Chapter 3.0 and the MSIVs are discussed in Section 3.7. This section discusses the non-safety-related sections of the main steam and bypass systems, downstream of the MSIVs.

An evaluation confirmed that the main steam system will support operation at increased RTP. The main steam system was designed and evaluated for the VWO conditions discussed above. The steam flow at the 1.4% increased RTP operating conditions will be 14.4 Mlb_m/hr, while the design VWO steam flow is

14.6 Mlb_m/hr. The turbine vendor evaluation determined that the turbine nozzles will only pass approximately 14.3 Mlb_m/hr (1% more steam flow) in their current configuration, maintaining the 3% control valve margin required for normal operation. It is expected that the turbine nozzles will require modification to increase their flow area to accommodate increased flow requirements beyond a 1% flow increase. However, the design steam flow requirements will still be maintained under either the 1% flow increase or the 1.4% flow increase, because the VWO conditions are not approached by the increased flow. The effects of the higher steam flow on main steam piping erosion-corrosion are described in section 3.5.2.2.

The steam seal evaporator supply flow will increase slightly, to 31,300 lb_m/hr from 31,250 lb_m/hr and the reactor feed pump turbine flow will remain about the same (Reference 1.6 section 7.2), since reactor pressure used for evaluation does not increase. The increase in steam seal evaporator supply flow will have no significant effect on the design margins of the system.

The bypass steam system is not affected by core thermal power level. Since the reactor pressure used for evaluation and the turbine inlet pressure remain the same, and the operating steam flow increases by 1.4%, the fraction of bypass flow to steam flow reduces from 0.258 to 0.254, which is still greater than the design requirement for this ratio of 0.25. The bypass flow corresponds to approximately 30% of the new core thermal power level, below which the steam bypass system can accommodate the pressure transient that occurs from a turbine trip, without a reactor scram. The control functions of the turbine that affect bypass valve operation require no changes (Reference 1.6, Section 7.3).

7.3 EXTRACTION STEAM

The extraction steam system supplies steam from intermediate stages of the main turbine cycle to the feedwater heaters. The system is designed to avoid main turbine overspeed by preventing water from flashing to steam and reversing flow into the turbine on a turbine trip; and to avoid water induction damage by preventing water from being drawn into the turbine in the case of a feedwater heater tube failure. The system is not safety-related, but is affected by increased RTP.

The higher reactor core thermal power, and the associated increase in turbine generator outlet power, will affect extraction steam flows, pressures and temperatures. The design conditions for system piping bound the increased steam flow conditions. The protection provided by the bleeder trip valves, extraction steam

isolation valves and associated drain valves will adequately prevent turbine damage from overspeed and water induction after the implementation of operation at increased RTP, and the system will supply sufficient steam to the feedwater heaters. The increased flow through the bleeder trip valves may exceed their original design capacity, but these valves have been evaluated for the increased flow conditions with satisfactory results. Any effects of the higher steam flow on extraction piping erosion-corrosion rates will be detected and tracked by the SSES piping erosion-corrosion program (see Section 3.5.2.2). Therefore, it is concluded that the extraction steam system will support operation at increased, licensed core thermal power operation.

7.4 MOISTURE SEPARATORS

Each SSES main turbine has two horizontal moisture separators, located on each side of the turbine, in the crossaround piping loop between the high pressure and low pressure turbine cylinders. They remove moisture formed during energy extraction. They are not safety-related but are affected by the increased power level.

The physical layout of the moisture separators for the two units is different, but the conclusions of the evaluations at increased RTP are the same. The moisture separators were designed for the VWO conditions, as discussed above. The evaluation confirmed that the main turbine moisture separators and their drain control and turbine water induction prevention features will support operation at the increased power and steam flow level. The separator moisture removal effectiveness will decrease slightly as a result of the increased steam flow; however, the increase in turbine blade and nozzle erosion will be negligible.

7.5 STEAM SEALS AND DRAINS

The main turbine steam seals and drains supply clean (nonradioactive) steam to the main turbine shaft seals; the stem packing of the turbine stop valves, control valves, combined intermediate valves and bypass valves; the shaft seals of the reactor feed pump turbines and the stem packing of the reactor feed pump turbine stop and control valves. The systems are not safety-related but are affected by operation at increased RTP.

An evaluation confirmed that the main turbine steam seals and drains will support operation at increased, licensed core thermal power. The seals and drains were designed to operate at the VWO conditions discussed above, which encompasses the proposed power level increase.

7.6 CONDENSATE AND FEEDWATER SYSTEMS

The condensate and feedwater systems are designed to draw condensate from the main condenser hotwell and supply feedwater to the reactor, at the flow, quality, temperature and pressure required by the reactor, at all operating conditions. Evaluations at the increased RTP confirmed that the condensate and feedwater systems will meet these requirements with no equipment changes and with only minor setpoint changes. Setpoint changes are discussed in Section 5.2.3.

7.6.1 Normal Operation

The condensate and feedwater systems were originally designed for VWO conditions, which include the proposed normal operating conditions. The evaluation concluded that condensate and feedwater pump capacity, motor and turbine power and support functions are adequate for increased, licensed core thermal power operation and adequate design margins are maintained. Condensate and feedwater pump NPSH available is adequate at the increased core thermal power conditions.

Normal feedwater temperature will increase by 1.0 to 1.1°F but the existing system design pressures and temperature will not require change.

The feedwater control system will continue to maintain reactor level within the required range at all normal operating conditions at the increased core thermal power level. The hotwell level control system will continue to control condenser hotwell level within the required range at normal operating conditions at the increased power level. Feedwater heaters, drains, drain control valves and heater steam supply are adequate for operation at increased RTP. See Section 7.3, "Extraction Steam."

The increase in RTP and associated increase in feedwater and steam flow will increase the rate of erosion-corrosion slightly in the piping and the feedwater heaters. Predictive methods for determining inspection locations will continue to be used, validated and upgraded as required, as in the current erosion-corrosion program. See the Erosion-Corrosion discussion in Section 3.5.2.2.

7.6.2 Transient Operation

The plant response to a single feedwater pump trip will not change, since the reactor power will run back along the currently analyzed load line to the same reduced-flow operating condition, and the evaluation concluded that the feedwater pump capacity margin and the feedwater control system response are adequate for this transient.

The feedwater controller receives inputs from reactor vessel water level, steam mass flow and feedwater mass flow. The reactor vessel operating water level will not be changed in operating at increased RTP. The ranges of the steam and feedwater flow transmitters and flow elements are adequate for the increased flow rates. The signal range from the feedwater controller to the reactor feed pump governor has the required margin for transient responses at increased RTP.

Operation at the increased power and flow level will cause small changes in the heater steam and drain flows. Heater drain level control response to transients will remain adequate.

With six of seven condensate demineralizer vessels in service, the design maximum flow through the vessels is adequate to accommodate isolation of one vessel at the increased flow.

The hotwell level control system will accommodate a transient of 10% of the increased feedwater flow.

The feedwater heaters will continue to limit feedwater temperature reduction to 100°F or less for any single failure or operator action. The calculated reduction is less than 60°F for the limiting event.

Calculations at 110% of original design feedwater flow with two or three condensate pumps operating showed that adequate condensate pump NPSH would be available during transients.

Condensate low-load operation uses a recirculation line and control valve, designed for operation at the system design pressure. This function is not affected because neither the control valve logic nor the system design pressure will be changed as a result of operation at increased RTP.

Previously, PPL performed an evaluation of the feedwater check valves to verify their structural integrity following the postulated rupture of the feedwater line upstream of the outermost containment isolation valve (HV-F032A, B). An evaluation of the feedwater check valves for the same transient at the LOCA

design temperature and pressure was performed as a part of the previous power uprate analysis (Reference 1.6, Section 7.6.2). The conditions used in the analysis (102% of uprated power (3510 MWt) and the corresponding reactor pressure) bound the conditions at increased, licensed core thermal power. This analysis was redone in 1998 because the feedwater check valve design was changed, but the analysis remains applicable to the current plant design. Since the plant LOCA design conditions do not change in the implementation of operation at increased RTP, the feedwater check valve design is adequate for the increased power condition.

7.6.3 Condensate Demineralizers

Reactor water quality will remain within existing Technical Specification limits with six demineralizers in service. The increased condensate and feedwater flow will remain within the deep-bed condensate demineralizer design flow of 4800 gpm per bed with all seven beds in operation. The addition of the condensate filtration system (CFS, Section 7.6.5), in conjunction with plant operation with hydrogen water chemistry (HWC, Section 7.6.4), will assure that operation with seven demineralizer beds in service is routine. There will be occasions when one deep bed demineralizer is removed from service for maintenance, but during these infrequent periods, the six remaining beds can process the full feedwater flow.

The condensate demineralizer resin has a design maximum temperature of 140°F. A review of the turbine heat balance provided by the turbine-generator vendor, General Electric Company, shows that the condensate temperature leaving the condenser at increased RTP operation is unaffected by increased power operation. Therefore, there is no effect on the condensate demineralizer design resulting from operation at increased RTP.

7.6.4 Hydrogen Water Chemistry (HWC) System

The purpose of the hydrogen water chemistry (HWC, Reference 1.1 Section 9.5.9) is to supply and inject hydrogen gas into the feedwater system. The required hydrogen injection rate is the rate necessary to mitigate the chemical conditions in the reactor pressure vessel that allow inter-granular stress corrosion cracking (IGSCC) in the lower reactor vessel internals. The HWC system has no safety related functions

The injection rate is a function of reactor power level and feedwater flow rate. Therefore, it may be necessary to increase the hydrogen injection rate at a rate proportional to the increase in power level and feedwater flow rate but this small increase is well within the system capacity.

The programmable logic controllers (PLC's) that control hydrogen injection rate will require investigation because the PLC's may get their 100% power value from either the feedwater flow rate or the reactor power level measurement. The PLC's may require reprogramming to assure they give the correct hydrogen injection as a function of power level. The HWC PLC's are not safety-related and the required programming is well within their capability.

7.6.5 <u>Condensate Filtration System</u>

The purpose of the condensate filtration system is to remove as many impurities from the reactor feedwater as possible, so that the condensate demineralizers (Section 7.6.3) can continue to perform their ion exchange function unhindered by the impurities.

The CFS was designed to handle a total of 29000 gpm condensate flow. This converts to a total of 14.3 Mlb_m/hr, which is slightly less than feedwater flow required to support the increased operating power level of 3489 MWt (14.4 Mlb_m/hr). The slight increase in flow rate can be handled by the CFS without changing the pressure drop requirements for backwash; however, the frequency of backwash may be slightly increased. Since the pressure loss requirements are not changed, the effect on condensate pump suction NPSH requirements will not be affected.

7.7 MAIN CONDENSER

The main condenser is a triple-pressure deaerating type with three separate shells that accept exhaust from the three low-pressure turbine shells. The main condenser also provides the function of containing leakage from the main steam isolation valves (MSIVs) previously performed by the MSIV- leakage control system (MSIV-LCS). The modification to remove the MSIV-LCS was conducted in accordance with the PPL modification process and, since neither the steam line pressure nor the MSIV leakage requirements are changed by plant operation at increased RTP, the use of the condenser for providing the MSIV-LCS function is also unaffected.

The condenser hotwell must retain condensate for at least two minutes at full load to allow radioactive decay, prior to returning the condensate to the cycle. An evaluation confirmed that the main condenser will meet these requirements after the implementation of increased RTP operation.

7.7.1 Normal Operation

The main condenser was originally designed for the valves wide open (VWO) condition at the original plant power level. The design heat load at that point was 7.92×10^9 Btu/hr and the power uprate submittal increased the normal condenser heat load to 7.93×10^9 Btu/hr (Reference 1.6; Section 7.7.1). Operation at increased RTP will increase the normal condenser heat load to 8.0×10^9 Btu/hr, as a result of the increase in main steam drain flow that occurs at the higher operating power level. The additional heat load will increase the condenser outlet circulating water temperature by approximately 0.2°F, but the condenser will continue to meet design requirements.

7.7.2 Transient Operation

Occasional loads from bypass steam, extraction steam dump lines, and turbine cycle relief valve discharges are not significantly changed from the original design. The original design included no additional condenser area to accommodate these discharges. The additional flow from these lines at the increased operating power level will result in slightly increased condenser pressures. The condenser evaluation performed by PPL concluded that these changes are acceptable.

7.8 MAIN CONDENSER EVACUATION SYSTEM

The main condenser evacuation system provides no safety-related functions. The evacuation system consists of two subsystems:

1. The mechanical vacuum pump evacuates the condenser during shutdown and startup conditions only, and is therefore not affected by the increased operating core power level. This subsystem discharges to the turbine building ventilation system (Section 6.6.5).
2. The steam jet air ejectors (SJAEs) operate at power, to remove noncondensible gases due to air inleakage, fission gas release from fuel rods and radiolysis of water in the reactor; and to condense steam removed with the noncondensibles and return the condensate to the condenser. This subsystem discharges to the gaseous radwaste system (Section 8.2).

An evaluation confirmed that the system will satisfactorily perform these functions at the increased RTP conditions.

Normal condenser air inleakage is 15-30 scfm, which is much less than the original design. Operation at the increased, licensed core thermal power level will cause a slight increase in normal condenser operating pressure, which will further reduce normal condenser inleakage. The rate of noncondensible gas generation resulting from radiolysis of water is proportional to operating power level. The radiolysis rate for 3441 MWt is 114 scfm hydrogen and 57 scfm oxygen. Increasing the operating power to 3489 MWt (a power increase of 1.4%) results in estimated noncondensible gas generation of 116 scfm hydrogen and 58 scfm oxygen. The condenser is designed to remove 160 scfm hydrogen and 80 scfm oxygen; therefore, the design limits for condenser evacuation are not challenged at the increased power level. Fission gases appear in trace amounts in the condenser, resulting from small fuel pin leaks and the fission of tramp uranium in the fuel cladding, and are not expected to increase significantly (see Section 8.5.3). Since the rate of noncondensible gas removal and the condenser pressure will not change significantly, the steam that is removed with and must be condensed from the noncondensible gases will not change significantly.

7.9 CIRCULATING WATER SYSTEM AND COOLING TOWERS

The circulating water system circulates water through a natural-draft cooling tower to remove the latent heat from the main condenser and the sensible heat from the service water system. The system is not safety-related.

The increase in main condenser heat load (see Section 7.7.1) will result in an uprated circulating water heat load beyond the previously evaluated value and will result in a condenser outlet circulating water temperature increase of approximately 0.2°F. The increased outlet temperature increases the cooling tower air flow rate, and that increase in air flow compensates for the temperature increase so that the cooling tower is able to dissipate the additional heat load without any perceptible change in cold water temperatures or overall cooling tower performance. An evaluation performed by PPL confirmed that the circulating water system and the cooling tower will support operation at the increased power level.

8.0 RADWASTE SYSTEMS AND RADIATION SOURCES

This chapter primarily includes information requested by Regulatory Guide 1.70; Chapters 11 and 12 as they apply to increased RTP.

8.1 LIQUID RADWASTE SYSTEM

A common liquid radwaste system serves both SSES units. The system is not safety-related but is affected by operation at increased RTP.

The higher condensate, feedwater and steam cycle flows resulting from the increased core thermal power operation will increase liquid radwaste influents by approximately 1.5%. The increase expected from increased power operation can be adequately handled by the system. Much of the previous liquid radwaste influent resulted from backwashing of the condensate demineralizers, and the condensate filtration system (CFS), now in place, has reduced the amount of condensate demineralizer backwash requirements considerably (see Section 7.6.4). Therefore, the liquid radwaste system will support operation at the increased core thermal power level.

8.2 GASEOUS RADWASTE SYSTEM

Gaseous radwaste treatment and release is performed by three systems, the standby gas treatment system (SGTS), the offgas recombiner system and the ambient temperature charcoal treatment system. The SGTS is an engineered safety feature (ESF) and is described in Section 4.4. This section describes the offgas recombiner system and the ambient temperature charcoal treatment system.

Noncondensible gases are extracted from the condenser by the steam jet air ejectors (Section 7.8). This flow is processed in the offgas recombiners, held up in delay piping for decay of radioactive products and filtered through the ambient temperature charcoal treatment system prior to release. This is the major source of gaseous radionuclide release during normal operation.

8.2.1 Offgas Recombiner System

The discussion of the required capacity of the main condenser evacuation system (Section 7.8) notes that actual condenser air inleakage is less than design, and that the increase in condenser pressure as a result of the increased core thermal power operation should further reduce the condenser inleakage.
PPL performed an evaluation on the offgas recombiner system and determined that the expected 1.4% increase in hydrogen and oxygen flow, proportional to the thermal increase from 3441 MWt to 3489 MWt, has a negligible effect on the performance of the system. Therefore, the offgas recombiner system will support plant operation at increased RTP.

8.2.2 Ambient Temperature Charcoal Treatment System

The offgas charcoal treatment system was designed for operation at 3440 MWt. Increasing the power to 3489 MWt will have a minor effect on the rate of noble gas and iodine generation, and will not challenge the limits of system performance. The effects of operation at increased RTP on 10 CFR 20 limits, 10 CFR 50 limits and ALARA guidelines is addressed in Section 8.6.3.

8.3 SOLID RADWASTE SYSTEM

The solid radwaste system is not safety-related. A common solid radwaste system serves both Units 1 and 2.

An evaluation performed by PPL confirmed that the solid radwaste system will perform its functions at the increased core thermal power conditions. Operation at the higher core thermal power level will increase solid waste generation by less than 3%. The maximum expected solid radwaste generation is within the capacity of the solid waste system for offsite shipment, or for over 5 years of onsite storage.

The increase in core thermal power will cause a minor increase in waste activity levels and may change the radionuclide distribution. Any required change in waste classification will be assessed through normal waste sampling and analysis.

8.4 DRY FUEL STORAGE FACILITY

PPL has added the Dry Fuel Storage Facility to assure that the current spent fuel pools continue to have the capability to accept a fuel core offload (References 8.1 and 1.1, Section 11.7). The Dry Fuel Storage facility is on site and operational. The facility accepts fuel assemblies that have been stored for a significant period of time (that is, the fuel assembly decay power is reduced considerably before removal to the Dry Fuel Storage Facility). Therefore, because transfer of a fuel assembly to the Dry Fuel Storage Facility is limited by fuel assembly heat load, the only effect that operation at increased RTP has on the facility is to slightly increase the amount of time spent fuel must be stored in the fuel pools. The fuel pools are already evaluated for the increased heat load (see Section 6.3).

8.5 RADIATION SOURCES IN THE REACTOR CORE

Radiation sources in the reactor core are a function of reactor operating power level.

8.5.1 Operation

During power operation the radiation sources in the core are directly related to the operating fission rate. These sources include radiation from the fission process, accumulated fission products and neutron reactions as a secondary result of fission. Historically, these sources have been defined in terms of energy released per unit of reactor power. Therefore, the increase in operating source terms is no greater than the increase in power level resulting from operation at increased RTP (Reference 1.6; Section 8.3.1).

8.5.2

8.6 RADIATION SOURCES IN THE REACTOR COOLANT

8.6.1 Coolant Activation Products

During reactor operation the coolant passing through the core region becomes radioactive as a result of the ongoing nuclear reactions. Since these sources are produced by interactions in the core region, their rates of production are proportional to fission power. However, while the magnitude of the source production increases in proportion to fission power, the concentration in the steam remains nearly constant. This is because the increase in activation production is balanced by the corresponding increase in steam flow. The activation products observed in the reactor water increase in approximate proportion to the increase in thermal power (1.4%). The activation products are bounded by the existing design basis concentration.

8.6.2 Activated Corrosion Products

The reactor coolant contains activated corrosion products, which are the result of metallic materials entering the water and being activated in the reactor region. Under increased core thermal power conditions, the feedwater flow increases with power and the activation rate in the reactor region increases with power. The net result is expected to be an approximately 3 percent increase in activated corrosion products over the conditions that existed prior to the implementation of increased, licensed core thermal power. However, the level of activation products is considerably less than previously analyzed, because of the implementation of hydrogen water chemistry operation coupled with the operation of the condensate filtration system. Since the dose increase resulting from the operation of hydrogen water chemistry has previously been evaluated and found to be within design basis, no change in the design basis activated corrosion product concentrations is required.

8.6.3

8.7 RADIATION LEVELS

Radiological effects on the environment are described in Section 11.4.

8.7.1 Normal Operation

An evaluation of the effects of the first SSES power uprate project concluded that the impact of increased power level on normal, calculated radiation dose rates was negligible because of the conservatism inherent in the calculational method. Since the time that study was performed, plant operation at SSES was converted to hydrogen water chemistry (HWC), and the condensate filtration system (CFS, see Section 7.6.4) was added to the condensate flow path. The effects of the addition of HWC and CFS on radiation source terms, in particular, the N^{16} source term, and shielding requirements were evaluated by PPL for the implementation of HWC/CFS. The evaluations performed for HWC/CFS operation concluded that the radiation levels in the plant would increase by a factor of five (5) to five and one-half (5.5). Increasing the normal, operating power level by 1.4% does not increase normal radiation levels from N^{16}, because both steam flow and feedwater flow increase proportionally with the increase in RTP. Since steam flow increases under increased core thermal power operation, the calculated transit times would decrease slightly, but not significantly with respect to the N^{16} decay half-life. Therefore, no change in the design projected dose rate above the current HWC/CFS assessment is expected.

8.7.2 Post-Operation and Accident

This section describes the effects of increased core thermal power operation on post-operation (post-shutdown) and post-accident radiation levels. Due to the increase in core inventory resulting from the higher operating power level, post-operation radiation levels change slightly. The slight increase in post-operation radiation levels will have no effect on the design of the plant, the Technical Support Center (TSC) or the Emergency Operations Facility (EOF).

Post-accident radiation levels are based on 105% of uprated power or 3616 MWt (Reference 1.1, Chapter 15). Therefore, increasing core thermal power level by 1.4% to 3489 MWt will have no effect on the post-accident radiation source term analysis.

8.7.3 Offsite Doses (Normal Operation)

Offsite doses due to design bases analyses are discussed on Section 9.2.

The normal offsite doses will not be significantly affected by operation at the increased core thermal power level. The review performed to evaluate normal conditions under HWC/CFS operation concluded that the radiological consequences, using conservative methods and assumptions, are well below any regulatory limits given in 10 CFR 20 and 10 CFR 50, Appendix I.

REFERENCES

8.1 Pennsylvania Power & Light Company, "Dry Fuel Storage," NDAP-QA-0658, August, 1999.'

9.0 REACTOR SAFETY PERFORMANCE FEATURES

This chapter primarily includes information requested by Regulatory Guide 1.70, Chapter 15 as it applies to operation at increased RTP.

9.1 REACTOR TRANSIENTS

Reload licensing analyses evaluate the limiting plant transients. Disturbances of the plant caused by a malfunction, a single failure of equipment or personnel error are investigated according to the type of initiating event (Regulatory Guide 1.70, Chapter 15). PPL will use its NRC approved licensing analysis methodology (Reference 9.1) to calculate the effects of the limiting reactor transients. The limiting events for the Susquehanna units were identified in Section 3.4 of Reference 9.1. The relatively small change in rated core thermal power will not affect the selection of limiting events as described in Section 3.4 of Reference 9.1. The events which will be explicitly evaluated for cycle specific reload analyses are:

1. Loss of Feedwater Heating
2. Feedwater Controller Failure (FWCF)
3. Generator Load Rejection Without Bypass (GLRWOB)
4. Turbine Trip Without Bypass (TTWOB)
5. Rod Withdrawal Error
6. Recirculation Flow Controller Failure, Increase (RFCF)
7. Fuel Loading Error

The limiting events that establish the minimum critical power ratio (MCPR) operating limits are currently GLRWOB, FWCF and RFCF. These seven events will continue to be evaluated for each reload, although it is expected that the GLRWOB, FWCF and RFCF events will remain limiting. The licensing analysis will continue to be performed up to the initial conditions of 3510 MWt, to account for the reduced level of power magnitude uncertainty (see Attachment 1).

Parametric studies were conducted as part of the development of PP&L's licensing methods (Reference 9.1, Section 3). These parametric studies lead to the following expectations. The GLRWOB ΔCPR (Δ critical power ratio) is determined based on a parametric analysis up to the maximum power level, and

the FWCF is analyzed as a function of power. Thus, the increased, licensed core thermal power level only changes the maximum power level considered. It is expected, based on the Reference 9.1 studies, that the increased operating core thermal power level for the GLRWOB and the FWCF will produce slightly higher ΔCPRs. This expectation will be confirmed as part of the reload licensing analyses. The RFCF event is analyzed as a function of core flow and lower power levels, as described in Reference 9.1, and is minimally affected by the increased rated core thermal power. The RFCF event will be analyzed without taking credit for the flow biased simulated thermal power trip. The RFCF event is analyzed for each reload cycle.

The safety limit minimum critical power ratio (SLMCPR) is calculated as part of the reload licensing analyzes using NRC-approved Siemens Power Corporation (SPC) methodology. No change will be made to the SPC methodology due to operation at increased RTP.

9.2

9.3 SPECIAL EVENTS

9.3.1 Anticipated Transients Without Scram (ATWS)

PPL performs plant specific ATWS analyses for each reload safety analysis. The reload safety analysis for the first fuel cycle with increased RTP will contain the appropriate ATWS analyses.

9.3.2 Station Blackout (SBO)

Per NUMARC 87-00 methodology, SSES is classified as a 4-hour-duration station blackout plant based on an offsite power design characteristic group of "P1," an emergency AC power configuration group of "D" and a target emergency diesel generator reliability of 0.975 (Reference 9.2). Operation at increased RTP will not affect this 4-hour-duration classification.

The limiting parameters for SBO events lasting longer than four hours are water inventory for decay heat removal, class 1E battery capacity, compressed air capacity and the effects of loss of ventilation. Operation at increased, licensed core thermal will result in more decay heat that will require a slightly larger water inventory. However, the current SBO analysis provides for adequate water inventory to meet the additional requirements of increased power operation.

Class 1E battery capacity and the compressed air system are unaffected by operation at the increased, licensed core thermal power level. The demand placed on the class 1E batteries and the compressed air system under SBO scenarios will not increase as a result of increased core thermal power operation; therefore, the capacity of these systems will remain adequate.

Operation at increased RTP will have a slight effect on loss of ventilation since slightly more heat will be transferred to the containment from the decay heat transmitted to the suppression pool from the reactor vessel during the SBO event. The remainder of the containment heat load results from piping and other temperature driven heat transmission and is, therefore, unchanged under increased, licensed core thermal power operation. PPL reviewed the results of SBO calculations performed for the previous power increase (Reference 1.6) and concluded that the temperatures predicted in SBO scenarios are well below the acceptance criteria given in the NUMARC methodology. In addition, the room heat loads used in this calculation were heat loads generated for the loss of offsite power (LOOP) event, and are very conservative with respect to SBO. Therefore, it is concluded that the current SBO analysis for compartment temperatures bounds the conditions expected for operation at increased RTP.

REFERENCES

9.1 Pennsylvania Power & Light Company, "Application of Reactor Transient Analysis Methods for BWR Design and Analysis," Report PL-NF-90-001, August 1990.

9.2 Nuclear Management and Resources Council, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," NUMARC 87-00, Rev. 1, August 1991.

10.0 EVALUATION OF GENERAL ISSUES

10.1 HIGH-ENERGY LINE BREAK

All of the existing high-energy line break evaluations were performed using the pressure and temperature corresponding to a power level of 3510 MWt (Reference 1.1, Section 3.6A). The affected compartment response to a high-energy line break is a function of mass-energy release rate and the mass-energy release is a function of the high-energy piping pressure and temperature. Since design pressure, reactor pressure used for evaluation and temperature are not changing in the implementation of increased RTP, the existing high-energy line break analysis supports the core thermal power increase and the discussion in Reference 1.6, Section 10.1 is unaffected.

10.2 MODERATE-ENERGY LINE BREAK AND INTERNAL FLOODING

Existing moderate-energy piping experiences no appreciable change in temperature or pressure as a result of operation at increased licensed core thermal power. The moderate-energy line break (MELB) water spray and flooding evaluation of the plant is, therefore, not affected by operation at increased RTP.

10.3 EQUIPMENT ENVIRONMENTAL QUALIFICATION

10.3.1 Environmental Qualification of Electrical Equipment

Safety-related electrical equipment is evaluated in accordance with IEEE 323, with the pressure and temperature in the area where the devices are determined from Reference 1.1, Section 3.6A. Since the high-energy line break calculations are unchanged for the implementation of increased, licensed core thermal power operation, the environmental conditions and the environmental qualification for the safety-related electrical devices are unchanged for operation at increased RTP.

10.3.1.1 Inside Containment

Qualification of safety-related electrical equipment located inside the containment is based on environmental conditions resulting from a main steam line break of a design basis loss-of-coolant accident (DBA-LOCA), as well as those conditions expected during normal operation. The analysis of the DBA-LOCA and main steam line break are not changed by the implementation of increased, licensed

core thermal power operation. The heat loads inside containment are driven by reactor coolant temperature, which is also unchanged from previous evaluations. Therefore, the normal operation and post-accident conditions inside the containment are not affected by increased core thermal power operation, and the existing environmental qualification for safety-related equipment inside the containment is not affected by operation at increased RTP.

10.3.1.2 Outside Containment

Temperature, pressure and humidity environments used for the qualification of equipment outside containment result from a main steam line break in the pipe tunnel, DBA-LOCA or other high-energy lines breaks, whichever is limiting for each plant area. Since the analysis of these defining events is not changed by the implementation of operation at increased RTP; the existing environmental qualification for safety-related equipment outside the containment is not changed.

10.3.2 Environmental Qualification of Mechanical Equipment with Non-Metallic Components

Operation at the increased, licensed core thermal power level is not expected to increase the normal area process temperatures by an appreciable amount. The radiation levels for normal and accident conditions were found to be essentially unchanged as evaluated in Section 8.6. Therefore, the environmental qualification of the mechanical equipment and components is unchanged from previous evaluations.

10.3.3 Mechanical Component Design Qualification

The mechanical design of equipment and components (pumps, heat exchangers, etc.) in certain balance-of-plant systems may be affected by operation at increased RTP, as a result of the minor increases in process temperatures and pressures in the power cycle. These changes in condition are small and well within the design parameters of affected components.

10.4 INDIVIDUAL PLANT EVALUATION (IPE)

The IPE (Reference 10.1) and the evaluation performed to assess the effect of the previous power uprate on the IPE were reviewed to determine the effect of increased, licensed core thermal power on the conclusions of the IPE. The net effect on the IPE of operation at increased RTP is to remove an amount of time proportional to the operating power increase from operator response and equipment repair times.

The previous power uprate of ~5% thermal power showed that the reduction in time for operator action is approximately equal to the increase in core thermal power and the corresponding increase in decay heat generation. Discussions with IPE analysts and Operations Staff indicate that the proposed, small increase in RTP will have a negligible impact on operator response time and the effect on other analysis documented in the IPE is so small as to be not quantifiable. Therefore, the additional 1.4% core thermal power increase does not have any significant impact on IPE results.

10.5 INDIVIDUAL PLANT EXAMINATION FOR EXTERNAL EVENTS (IPEEE)

The Individual Plant Examination for External Events (IPEEE, Reference 10.2) evaluation was reviewed for potential effects from operation at increased RTP. The IPEEE evaluates plant risk from external events, such as tornadoes, hurricanes and other storms, seismic events and fires. Risk is a function of event probability and event consequences. Since the IPEEE evaluates external events, the probability of these events is independent of operating power level. Event consequences for a nuclear power plant are concerned with the spread of radioactivity and the potential for large scale radioactivity release resulting from core melt and/or containment failure. Section 8.6 of this report documents that radiation levels in the plant are unaffected by operation at increased RTP and section 10.4 concludes that the risk of plant damage is essentially unaffected by operation at increased RTP. Therefore, since neither the probability nor the consequences of an evaluated external event are affected by operation at increased RTP, the IPEEE is also unaffected.

10.6 EMERGENCY PREPAREDNESS AND LICENSED OPERATOR PERFORMANCE

10.6.1 Emergency Operating Procedures

The Emergency Operating Procedures (EOP's) at SSES were completely revised and upgraded in 1998. Increasing the core thermal operating power results in some modifications to the curves referenced in the EOP's and the calculations that support the curves. The boron injection initiation time (BIIT) is a weak function of core thermal power. The BIIT curve is part of the ATWS analysis, and will be revised for each operating cycle, as discussed in Section 9.3.1. A review of the other calculations supporting the EOP's shows that the core thermal power level shows up as a parameter in several calculations, but the review also indicated that changes resulting from the increased power level are small and, with training, transparent to the operator.

10.6.2 Alarm Response

As discussed in Section 1.2.12 of this report, the LEFM✓™ feedwater flow measurement system contains a self-checking module, which alarms when the system diagnoses a failure in the flow measurement algorithm. As a result of this alarm, a new alarm response procedure is required to detail required actions for the Operations Staff. The alarm response will be written and the Operations Staff will be trained on the alarm response procedure prior to the implementation of operation at increased RTP.

10.6.3 Effect on the Safety Parameter Display System (SPDS)

The Safety Parameter Display System (SPDS) is insensitive to normal operating power level, therefore, operation at increased RTP has no effect on the SPDS.

10.6.4 Effects on the Operator Training Program and the Simulator

The Operation Staff will be trained on operation at increased RTP prior to actual operation at increased RTP. Since Unit 2 will be the lead unit for the implementation of increased RTP, the Operations Staff will be fully trained on the differences between the Units, as currently occurs in the operator training program.

The plant specific simulator is referenced to Unit 1, however, the simulator can be reprogrammed to operate at the increased RTP. The programmed simulator will accept the increased power level and adjust operating flows, etc. to be consistent with the increased power level. As discussed in Section 10.4 of this report, the major impact on Operations Staff is the slight decrease in timing associated with operator response. The Operations Staff will be trained on this change prior to the implementation of operation at increased RTP.

10.6.5 Effects on the SSES Emergency Plan

The SSES Emergency Plan and staffing requirements relating to Emergency Plan response are insensitive to operating power level and are therefore unaffected by operation at increased RTP.

10.7 SUMMARY OF STARTUP TESTS

Compared to the initial startup test program, operation at the increased licensed core thermal power level requires only limited startup tests. The testing for increased core thermal power operation will be conducted in accordance with Reference 1.6. The following tests will be performed to assure adequate performance at the increased core thermal power conditions:

1. Testing will be performed on any instrumentation that requires recalibration for increased core thermal power operation.
2. Steady-state data will be taken at points from 90% up to the previous rated thermal power, so that operating performance parameters can be projected for increased power operation before the previous licensed power level is exceeded. The LEFM✓™ will be the measurement source of record for these tests, with the previously used venturi flow meter serving a backup.
3. Power increases beyond the previous licensed power level will be made along an established flow control/rod line in at least one intermediary step between the previous licensed power level and the new licensed power level. Steady-state operating data will be taken and evaluated at each step, using the newly installed LEFM✓™ system as the source of record for feedwater flow measurement and core thermal power calculation. Since the licensed power increase is small compared with the previous power uprate, the two step process is deemed appropriate.
4. Control system tests will be performed for the feedwater reactor water level controls and pressure controls. These operational checks will be made at the previous licensed thermal power condition and at the new licensed thermal power condition to show acceptable adjustment and operational capacity. The small increase in licensed core thermal power makes performing the adjustments only at the end points reasonable.

The recommended startup test approach is a series of small increases in steam flow, roughly 0.5% steam flow increase per step, followed by a series of Test Procedures (TP's) completed at each plateau. The Test Procedures will examine turbine and feedwater control stability and assure that the plant is operating as expected. The test program will proceed, and the individual Test Procedures will be reviewed and approved according to established plant procedures

REFERENCES

10.1 Pennsylvania Power & Light Company, "Susquehanna Steam Electric Station Individual Plant Evaluation," Calculation NPE-91-001, December 1991.

10.2 Pennsylvania Power & Light Company, "Susquehanna Steam Electric Station Individual Pant Examination for External Events," NE-94-001, June 1994.

11.0 LICENSING EVALUATIONS

11.1 REVIEW OF OTHER LICENSING AND DESIGN BASES ISSUES

Nuclear industry events, issues and SSES experiences have required evaluation of the SSES design and operating practices. Those evaluations may have depended on the licensed power level, and may therefore require reevaluation at the increased, licensed core thermal power conditions.

An extensive review of past events, issues and experiences was performed as part of the previous power uprate submittal (see Reference 1.6, Section 11.1). The previous power uprate included a change to the design basis accident (DBA) power level and an operating reactor pressure vessel pressure increase as well as an increase in core thermal power level. The previous review found that most events and issues were not affected by core thermal power. Those few events and issues that were affected were evaluated at the design power level. Since the current request for operation at licensed RTP does not increase either reactor pressure used for evaluation or the design thermal power, the results of the previous evaluations are applicable for the current core thermal power increase.

Any events and issues that have arisen since the previous power uprate review have been reviewed as part of the individual affected system review. The systems affected by the increase in core thermal have been extensively reviewed as part of this topical and all open issues resolved.

Changes to the Technical Requirements Manual (TRM) will be completed to assure that the proper core thermal power level is used in the case when the LEFM✓ system is out of service for a period of time longer than allotted by the alarm response procedure. The alarm response procedure and the TRM change will be performed according to the current procedures for changing those documents

FSAR changes will be proposed and submitted as part of the established FSAR change process.

11.2 EFFECT ON THE LICENSE AND TECHNICAL SPECIFICATIONS

Increasing the licensed core thermal operating power requires a license revision to reflect the increased RTP and changes to the Technical Specifications. The affected pages are marked up and include in the proposed license amendment.

11.3 "NO SIGNIFICANT HAZARDS" EVALUATION (10 CFR 50.92)

This evaluation is included with the proposed license amendment.

11.4 ENVIRONMENTAL ASSESSMENT

Operation at increased RTP has the potential to increase the environmental impact of the plant. An environmental impact analysis was therefore performed to evaluate the effects of the increase in RTP on the environment and on the existing plant environmental analyses, and to identify actions that might be required to permit power uprate under existing licenses, permits and agreements (Reference 1.6, Section 11.1). The analysis included an "Unreviewed Environmental Question" review as required by in the "Environmental Protection Plan" (References 1.4 and 1.5), and reviewed the effects previously evaluated in the "Environmental Report – Operating License Stage" (Reference 11.1) and the "Final Environmental Statement" (Reference 11.2); or limited by applicable federal and state permits and regional agreements. Some effects will remain the same, while increased core thermal power operation may nominally increase others. PPL has evaluated the environmental impact of operation at a power level of 3510 MWt, and concluded that no undue environmental effects are expected. This power level covers operation at increased RTP.

Since operation at increased RTP will not significantly change the method of generating electricity, nor of handling any influents from the environment or effluents to it, no new or different environmental impacts are expected. The conservative models and methods used in the environmental assessments of the original design, confirmed by studies conducted during actual operation, show that more than adequate margin exists for the proposed core thermal power increase without exceeding the nonradiological environmental effects estimated in the original estimates and analyses and cited in the original permit applications and impact statements.

Adequate margin also exists for the proposed core thermal power increase without exceeding regulatory limits for radiological effects. Radiological effects of routine operations are described in Section 8.6.3. Current operating experience also indicates that actual releases and waste disposal after licensed core

thermal power increase will continue to be significantly less than the original estimates.[1] For these reasons, operation at increased RTP is not expected to have an adverse effect on the routine operation "dose commitment" estimated by previous radiological environmental analyses, and no revision of these analyses is required.

The environmental assessment includes an estimate of the potential exposure from all accident types combined. The evaluation of accident doses were performed at 3510 MWt[2], which bounds the proposed power level under increased, licensed core thermal power operation. Although direct comparison with the original analyses is not meaningful because of changes in methodology, a comparison on a consistent basis would show that the expected dose is approximately proportional to core thermal power. Since the accident analyses were performed at 3510 MWt, and proposed operation at increased, licensed core thermal power will not exceed, in the worst case, the 3510 MWt limit, no revision of these analyses is required.

An "Unreviewed Environmental Question" evaluation was conducted in accordance with each unit's "Environmental Protection Plan" (References 1.4 and 1.5) to determine if the previous power uprate could cause any significant environmental impacts. The previous review includes a review of the National Pollutant Discharge Elimination System (NPDES) Permit and other permits, and concluded that the previous power uprate should not contribute to any new noncompliances. No significant increase in generation of hazardous or nonhazardous waste is expected, except for a small increase in sediment removed from the cooling tower, which is roughly proportional to the increase in thermal power. No change is expected in either the load on the sewage treatment plant or well water usage and river water use will remain within the exiting agreement with the Susquehanna River Basin Commission.

The evaluation for the previous power uprate was extended to an operating power level of 3510 MWt and concluded that no significant changes would occur at the increased power level, except for an increase in both cooling tower makeup requirements and an increase in waste heat discharged. The cooling tower makeup and waste heat discharge were proportional to the increase in core thermal power operation and were evaluated at 3510 MWt. Since the 3510 MWt power level is the design "not to exceed" power level,

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1 See the semiannual effluent and waste disposal reports; and the discussion of the effects of increased RTP on the radwaste systems in Sections 8.1, 8.2 and 8.3, above.

2 Section 9.2, above, describes the calculation of accident doses for increased, licensed core thermal power operation, which were actually calculated at 3616 MWt.

that is, the maximum power level that the plant could be operating when the instruments indicate licensed power level, it is concluded that the environmental impact study in Reference 1.6, Section 11.1 bounds the conditions for increased RTP operation. Based on that evaluation, it is concluded that operation at the increased, licensed core thermal power level is not an "unreviewed environmental question."

The proposed operation at increased RTP therefore requires no change to the "Environmental Protection Plans" (References 1.4 and 1.5), since it does not involve:

a) A significant increase in any adverse environmental impact previously evaluated in the "Environmental Report – Operating License Stage" (Reference 11.1), or the "Final Environmental Statement" (Reference 11.2), or in any decision of the Atomic Safety and Licensing Board:

b) A significant change in effluents or power levels; or

c) A matter not previously reviewed and evaluated in the documents specified in paragraph (a) that may have some significant adverse environmental impact.

The Pennsylvania Department of Environmental Resources will be notified of the change in power level prior to operation at increased RTP.

REFERENCES

11.1 Pennsylvania Power & Light Company, "Susquehanna Steam Electric Station, Units 1 and 2, Environmental Report – Operating License Stage" (ER-OL), May 1978.

11.2 Pennsylvania Power & Light Company and Allegheny Electric Cooperative, Inc., "Final Environmental Statement Related to the Operation of Susquehanna Steam Electric Station, Units 1 and 2, Docket Nos. 50-387 and 50-388" (FES), NUREG-0564, June 1981.

ATTACHMENT 1

CALDON, Inc. Topical Report 80P, 'Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System,' Revision 0.

AND

CALDON, Inc. Engineering Report 160P Supplement to Topical Report ER-80P, 'Basis for a Power Uprate with the LEFM✓™ System,' Revision 0.

Note:

These two reports are not included herein since they contain proprietary information and have previously been provided to the NRC.

ATTACHMENT 7

NO SIGNIFICANT HAZARDS CONSIDERATIONS AND ENVIRONMENTAL ASSESSMENT

NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

PPL Susquehanna, LLC has evaluated the proposed amendment and determined that it involves no significant hazards consideration. According to 10CFR50.92 (c) a proposed amendment to an operating license involves a no significant hazards consideration if operation of the facility with the propose amendment would not:

- Involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated;

- Create the possibility of a new or different kind of accident from any previously analyzed; or

- Involve a significant reduction in a margin of safety.

PPL Susquehanna, LLC proposes to increase the Rated Thermal Power (RTP) level of Susquehanna Units 1 and 2. The (RTP) increase will be achieved by installation of the Leading Edge Flow Meter (LEFM✓™) supplied by Caldon, Inc. The LEFM✓™ provides improved feedwater flow measurement accuracy and thus improved operation power level certainty.

The determination that the criteria set forth in 10CFR50.92 are met for this amendment as indicated below:

1. Does the proposed change involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated?

   The comprehensive analytical efforts performed to support the proposed change included a review of the NSSS systems and components that could be affected by this change. All systems and components will function as designed and the applicable performance requirements have been evaluated and found to be acceptable.

   The primary loop components (reactor vessel, reactor internals, CRDMs, piping and supports, reactor coolant pumps, etc.) continue to comply with their applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components.

All of the NSSS systems will still perform their intended design functions during normal and accident conditions. The balance of plant systems and components continue to meet their applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components. All of the NSSS/BOP interface systems will continue to perform their intended design functions. The safety relief valves and containment isolation valves meet design sizing requirements at the uprated power level. The current LOCA hydraulic forcing functions are still bounding for the proposed 1.4% increase in power.

Because the integrity of the plant will not be affected by operation at the uprated condition and it can be concluded that all structures, systems, and components required to mitigate a transient remain capable of fulfilling their intended function, the effects on the remainder of the safety analyses can be assessed. The reduction in the uncertainty allowance provided for the reactor heat balance measurement allows current safety analyses to be used, without change, to support operation at a core power of 3489 MWt. As such, all FSAR Chapter 15 accident analyses continue to demonstrate compliance with the relevant event acceptance criteria. Those analyses performed to assess the effects of mass and energy releases remain valid. The source terms used to assess radiological consequences have been reviewed and determined to bound operation at the 1.4% uprated condition.

Therefore, this proposed amendment does not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any previously analyzed?

   No new accident scenarios, failure mechanisms or single failures are introduced as a result of the proposed change. All systems, structures, and components previously required for the mitigation of a transient remain capable of fulfilling their intended design function. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety related system.

   Therefore, this proposed amendment does not involve a possibility of a new or different kind of accident from any previously analyzed.

3. Does the proposed change involve a significant reduction in a margin of safety.

   Operation at the uprated power condition does not involve a significant reduction in a margin of safety. Extensive analyses of the primary fission product barriers have concluded that all relevant design criteria remain satisfied, both from the standpoint of the integrity of the primary fission product barrier and from the standpoint of compliance of the regulated acceptance criteria. As appropriate, all evaluations have been performed using methods that have either been reviewed and approved by the NRC in Section 5.6.5 of the SSES Technical Specifications, or that are in compliance with all regulatory review guidance and standards.

   Therefore these changes do not involve a significant reduction in margin of safety.

Based upon the above, the proposed amendment does not involve a significant hazards consideration.

ENVIRONMENTAL ASSESSMENT

An environmental assessment is not required for the proposed change because the requested change conforms to the criteria for actions eligible for categorical exclusion as specified in 10 CFR 51.22(c)(9). The requested change will have no impact on the environment. As discussed in the "No Significant Hazards Consideration Evaluation", the proposed change does not involve a significant hazard consideration. The proposed change does not involve a change in the types or increase in the amounts of effluents that may be released off-site. In addition, the proposed change does not involve an increase in the individual or cumulative occupational radiation exposure.

ATTACHMENT 8

TECHNICAL SPECIFICATION MARK-UPS

Note:

The changes shown to Section 5.6.5 for both Unit 1 and Unit 2 are marked up on the current NRC approved affected pages.

UNIT 1 TECHNICAL SPECIFICATION MARK-UPS

1.1 Definitions (continued)

RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~3441~~ MWt. 3489
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: a. The reactor is xenon free; b. The moderator temperature is 68°F; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
TURBINE BYPASS SYSTEM RESPONSE TIME	The TURBINE BYPASS SYSTEM RESPONSE TIME consists of the time from when the turbine bypass control unit generates a turbine bypass valve flow signal

(continued)

5.6 Reporting Requirements (continued)

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the main steam safety/relief valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. The Average Planar Linear Heat Generation Rate for Specification 3.2.1;

2. The Minimum Critical Power Ratio for Specification 3.2.2;

3. The Linear Heat Generation Rate for Specification 3.2.3;

4. The Average Power Range Monitor (APRM) Gain and Setpoints for Specification 3.2.4; and

5. The Shutdown Margin for Specification 3.1.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC~~, specifically those described in the following documents~~:

Insert 1

1. PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis," July, 1992.

(continued)

5.6 Reporting Requirements

5.6.5 COLR (continued)

11. PL-NF-90-001, Supplement 2-A, "Application of Reactor Analysis Methods for BWR Design and Analysis: CASMO-3G Code and ANFB Critical Power Correlation", July 1996.

12. ANF-89-98(P)(A) Revision 1 and Revision 1 Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation, May 1995.

13. ANF-91-048(P)(A), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model," January 1993.

14. XN-NF-80-19(P)(A), Volumes 2, 2A, 2B, and 2C "Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model," September 1982.

15. XN-NF-80-19(P)(A), Volumes 3 Revision 2 "Exxon Nuclear Methodology for Boiling Water Reactors Thermex: Thermal Limits Methodology Summary Description," January 1987.

16. XN-NF-79-71(P)(A) Revision 2, Supplements 1, 2, and 3, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors," March 1986.

17. EMF-1997(P)(A) Revision 0, "ANFB-10 Critical Power Correlation," July 1998, and EMF-1997(P)(A) Supplement 1 Revision 0, "ANFB-10 Critical Power Correlation : High Local Peaking Results," July 1998.

Insert 2

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

INSERT 1:

When an initial assumed power level of 102 percent of rated power is specified in a previously approved method, this refers to the power level associated with the design basis analyses, or 3510 MWt. The power level of 3510 MWt is 100.6 % of the rated thermal power level of 3489 MWt. The RTP of 3489 MWt may only be used when feedwater flow measurement (used as input to the reactor thermal power measurement) is provided by the Leading Edge Flow Meter (LEFM✓™) as described in the LEFM✓™ Topical Report and supplement referenced below. When feedwater flow measurements from the LEFM✓™ system are not available, the core thermal power level may not exceed the originally approved RTP of 3441 MWt, but the value of 3510 MWt (102% of 3441 MWt) remains the initial power level for the bounding licensing analysis.

Future revisions of approved analytical methods listed in this Technical Specification that are currently referenced to 102% of rated thermal power (3510 MWt) shall include reference that the licensed RTP is actually 3489 MWt. The revisions shall document that the licensing analysis performed at 3510 MWt bounds operation at the RTP of 3489 MWt so long as the LEFM✓™ system- is used as the feedwater flow measurement input into the core thermal power calculation.

The approved analytical methods are described in the following documents:

<u>INSERT 2</u>

18. Caldon, Inc., "TOPICAL REPORT: Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Engineering Report - 80P, March 1997.

19. Caldon, Inc., "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM✓™ or LEFM CheckPlus™ System, Revision 0," Engineering Report ER-160P, May 2000.

UNIT 2 TECHNICAL SPECIFICTION MARK-UPS

1.1 Definitions (continued)

RATED THERMAL POWER (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~3441~~ 3489 MWt.

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME

The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

SHUTDOWN MARGIN (SDM)

SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that:

a. The reactor is xenon free;

b. The moderator temperature is 68°F; and

c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn.

With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.

STAGGERED TEST BASIS

A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.

THERMAL POWER

THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TURBINE BYPASS SYSTEM RESPONSE TIME

The TURBINE BYPASS SYSTEM RESPONSE TIME consists of the time from when the turbine bypass control unit generates a turbine bypass valve flow signal

(continued)

5.6 Reporting Requirements (continued)

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the main steam safety/relief valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. The Average Planar Linear Heat Generation Rate for Specification 3.2.1;

2. The Minimum Critical Power Ratio for Specification 3.2.2;

3. The Linear Heat Generation Rate for Specification 3.2.3;

4 The Average Power Range Monitor (APRM) Gain and Setpoints for Specification 3.2.4; and

5. The Shutdown Margin for Specification 3.1.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. ~~specifically those described in the following documents~~:

INSERT 1

1. PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis," July, 1992.

2. XN-NF-80-19(P)(A), Volume 4, Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, Inc. June 1986.

(continued)

5.6 Reporting Requirements

5.6.5 COLR (continued)

14. ANF-89-98(P)(A) Revision 1 and Revision 1 Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation, May 1995.

15. ANF-91-048(P)(A), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model," January 1993.

16. XN-NF-80-19(P)(A), Volumes 2, 2A, 2B, and 2C "Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model," September 1982.

17. XN-NF-80-19(P)(A), Volumes 3 Revision 2 "Exxon Nuclear Methodology for Boiling Water Reactors Thermex: Thermal Limits Methodology Summary Description," January 1987.

18. XN-NF-79-71(P)(A) Revision 2, Supplements 1, 2, and 3, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors," March 1986.

19. EMF-1997 (P)(A) Revision 0, "ANFB-10 Critical Power Correlation," July 1998, and EMF-1997 (P)(A) Supplement 1 Revision 0, "ANFB-10 Critical Power Correlation : High Local Peaking Results," July 1998.

INSERT 2

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

INSERT 1:

When an initial assumed power level of 102 percent of rated power is specified in a previously approved method, this refers to the power level associated with the design basis analyses, or 3510 MWt. The power level of 3510 MWt is 100.6 % of the rated thermal power level of 3489 MWt. The RTP of 3489 MWt may only be used when feedwater flow measurement (used as input to the reactor thermal power measurement) is provided by the Leading Edge Flow Meter (LEFM✓™) as described in the LEFM✓™ Topical Report and supplement referenced below. When feedwater flow measurements from the LEFM✓™ system are not available, the core thermal power level may not exceed the originally approved RTP of 3441 MWt, but the value of 3510 MWt (102% of 3441 MWt) remains the initial power level for the bounding licensing analysis.

Future revisions of approved analytical methods listed in this Technical Specification that are currently referenced to 102% of rated thermal power (3510 MWt) shall include reference that the licensed RTP is actually 3489 MWt. The revisions shall document that the licensing analysis performed at 3510 MWt bounds operation at the RTP of 3489 MWt so long as the LEFM✓™ system- is used as the feedwater flow measurement input into the core thermal power calculation.

The approved analytical methods are described in the following documents:

INSERT 2

20. Caldon, Inc., "TOPICAL REPORT: Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Engineering Report - 80P, March 1997.

21. Caldon, Inc., "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM✓™ or LEFM CheckPlus™ System, Revision 0," Engineering Report ER-160P, May 2000.

ATTACHMENT 9

"CAMERA-READY" TECHNICAL SPECIFICATION PAGES

Note:

The pages provided herein include proposed changes to the Section 5.6.5 pages that NRC has yet to approve. That is, these "camera-ready" pages presume all previously submitted proposed changes that affect these pages are approved by the NRC prior to approval of the changes proposed herein.

UNIT 1 "CAMERA READY"

1.1 Definitions (continued)

RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3489 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: a. The reactor is xenon free; b. The moderator temperature is 68°F; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during η Surveillance Frequency intervals, where η is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
TURBINE BYPASS SYSTEM RESPONSE TIME	The TURBINE BYPASS SYSTEM RESPONSE TIME consists of the time from when the turbine bypass control unit generates a turbine bypass valve flow signal

(continued)

5.6 Reporting Requirements (continued)

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the main steam safety/relief valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. The Average Planar Linear Heat Generation Rate for Specification 3.2.1;

2. The Minimum Critical Power Ratio for Specification 3.2.2;

3. The Linear Heat Generation Rate for Specification 3.2.3;

4. The Average Power Range Monitor (APRM) Gain and Setpoints for Specification 3.2.4; and

5. The Shutdown Margin for Specification 3.1.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC.

When an initial assumed power level of 102 percent of rated power is specified in a previously approved method, this refers to the power level associated with the design basis analyses, or 3510 MWt. The power level of 3510 MWt is 100.6% of the rated thermal power level of 3489 MWt. The RTP of 3489 MWt may only be used when feedwater flow measurement (used as input to the reactor thermal power measurement) is provided by the Leading Edge Flow Meter (LEFM✓™) as described in the LEFM✓™ Topical Report and supplement referenced below. When feedwater flow measurements from the LEFM✓™ system are not available, the core thermal power level may not exceed the originally approved RTP of 3441 MWt, but the value of 3510 MWt

(continued)

5.6 Reporting Requirements (continued)

(102% of 3441 MWt) remains the initial power level for the bounding licensing analysis.

Future revisions of approved analytical methods listed in this Technical Specification that are currently referenced to 102% of rated thermal power (3510 MWt) shall include reference that the licensed RTP is actually 3489 MWt. The revisions shall document that the licensing analysis performed at 3510 MWt bounds operation at the RTP of 3489 MWt so long as the LEFM✓™ system is used as the feedwater flow measurement input into the core thermal power calculation.

The approved analytical methods are described in the following documents:

1. PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis," July, 1992.

2. XN-NF-80-19(P)(A), Volume 4, Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, Inc. June 1986.

3. XN-NF-85-67(P)(A), Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel, "Exxon Nuclear Company, Inc., September 1986.

4. XN-NF-80-19(A), Volume 1, and Volume 1 Supplements 1 and 2 (March 1983), and Volume 1 Supplement 3 (November 1990), "Exxon Nuclear Methodology for Boiling Water Reactors: Neutronic Methods for Design and Analysis," Exxon Nuclear Company, Inc.

5. ANF-524(P)(A), Revision 2 and Supplement 1, Revision 2, "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors", November 1990.

6. ANF-1125(P)(A) and ANF-1125(P)(A), Supplement 1, "ANFB Critical Power Correlation", April 1990.

7. NEDC-32071P, "SAFER/GESTR-LOCA Loss of Coolant Accident Analysis," GE Nuclear Energy, May 1992.

(continued)

5.6 Reporting Requirements

5.6.5 COLR (continued)

17. EMF-1997(P)(A) Revision 0, "ANFB-10 Critical Power Correlation," July 1998, and EMF-1997(P)(A) Supplement 1 Revision 0, "ANFB-10 Critical Power Correlation : High Local Peaking Results," July 1998.

18. Caldon, Inc., "TOPICAL REPORT: Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Engineering Report - 80P, March 1997.

19. Caldon, Inc., "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM✓™ or LEFM CheckPlus™ System, Revision 0, "Engineering Report ER-160P, May 2000.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

UNIT 2 "CAMERA READY"

1.1 Definitions (continued)

RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3489 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: a. The reactor is xenon free; b. The moderator temperature is 68°F; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
TURBINE BYPASS SYSTEM RESPONSE TIME	The TURBINE BYPASS SYSTEM RESPONSE TIME consists of the time from when the turbine bypass control unit generates a turbine bypass valve flow signal

(continued)

5.6 Reporting Requirements (continued)

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the main steam safety/relief valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. The Average Planar Linear Heat Generation Rate for Specification 3.2.1;

2. The Minimum Critical Power Ratio for Specification 3.2.2;

3. The Linear Heat Generation Rate for Specification 3.2.3;

4 The Average Power Range Monitor (APRM) Gain and Setpoints for Specification 3.2.4; and

5. The Shutdown Margin for Specification 3.1.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC.

When an initial assumed power level of 102 percent of rated power is specified in a previously approved method, this refers to the power level associated with the design basis analyses, or 3510 MWt. The power level of 3510 MWt is 100.6% of the rated thermal power level of 3489 MWt. The RTP of 3489 MWt may only be used when feedwater flow measurement (used as input to the reactor thermal power measurement) is provided by the Leading Edge Flow Meter (LEFM✓™) as described in the LEFM✓™ Topical Report and supplement referenced below. When feedwater flow measurements from the LEFM✓™ system are not available, the

(continued)

5.6 Reporting Requirements (continued)

core thermal power level may not exceed the originally approved RTP of 3441 MWt, but the value of 3510 MWt (102% of 3441 MWt) remains the initial power level for the bounding licensing analysis.

Future revisions of approved analytical methods listed in this Technical Specification that are currently referenced to 102% of rated thermal power (3510 MWt) shall include reference that the licensed RTP is actually 3489 MWt. The revisions shall document that the licensing analysis performed at 3510 MWt bounds operation at the RTP of 3489 MWt so long as the LEFM✓™ system is used as the feedwater flow measurement input into the core thermal power calculation.

The approved analytical methods are described in the following documents:

1. PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis," July, 1992.

2. XN-NF-80-19(P)(A), Volume 4, Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, Inc. June 1986.

3. XN-NF-85-67(P)(A), Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel, "Exxon Nuclear Company, Inc., September 1986.

4. XN-NF-80-19(A), Volume 1, and Volume 1 Supplements 1 and 2 (March 1983), and Volume 1, Supplement 3 (November 1990), "Exxon Nuclear Methodology for Boiling Water Reactors: Neutronic Methods for Design and Analysis," Exxon Nuclear Company, Inc.

5. ANF-524(P)(A), Revision 2 and Supplement 1, Revision 2, "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors", November 1990.

6. ANF-1125(P)(A) and ANF-1125(P)(A), Supplement 1, "ANFB Critical Power Correlation", April 1990.

(continued)

5.6 Reporting Requirements

5.6.5 COLR (continued)

18. XN-NF-79-71(P)(A) Revision 2, Supplements 1, 2, and 3, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors," March 1986.

19. EMF-1997 (P)(A) Revision 0, "ANFB-10 Critical Power Correlation," July 1998, and EMF-1997 (P)(A) Supplement 1 Revision 0, "ANFB-10 Critical Power Correlation : High Local Peaking Results," July 1998.

20. Caldon, Inc., "TOPICAL REPORT: Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Engineering Report - 80P, March 1997.

21. Caldon, Inc., "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM✓™ or LEFM CheckPlus™ System, Revision 0, "Engineering Report ER-160P, May 2000.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

ATTACHMENT 10

PPL SUSQUEHANNA, LLC UNIT 1 AND UNIT 2 LICENSE PAGE MARK-UPS

UNIT 1 LICENSE PAGE MARK-UPS

(3) ~~PP&L~~PPL Susquehanna, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed neutron sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) ~~PP&L~~PPL Susquehanna, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) ~~PP&L~~PPL Susquehanna, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

~~PP&L, Inc.~~PPL Susquehanna, LLC (~~PP&L~~) is authorized to operate the facility at reactor core power levels not in excess of ~~3441~~ 3489 megawatts thermal in accordance with the conditions specified herein and in Attachment 1 to this license. The preoperational tests, startup tests and other items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 185, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. ~~PP&L~~PPL Susquehanna, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

For Surveillance Requirements (SRs) that are new in Amendment

UNIT 2 LICENSE PAGE MARK-UPS

(3) ~~PP&L~~PPL Susquehanna, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed neutron sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) ~~PP&L~~PPL Susquehanna, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) ~~PP&L~~PPL Susquehanna, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u> 3489

~~PP&L, Inc.~~PPL Susquehanna, LLC ~~(PP&L)~~ is authorized to operate the facility at reactor core power levels not in excess of ~~3441~~ megawatts thermal (100% power) in accordance with the conditions specified herein and in Attachment 1 to this license. The preoperational test, startup tests and other items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 159, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. ~~PP&L~~PPL Susquehanna, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

For Surveillance Requirements (SRs) that are new in Amendment 151