

Attachment 2

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 10

Extended Power Uprate

Response to Request for Additional Information

REDACTED AND NON-PROPRIETARY INFORMATION

Total number of pages in Attachment 2
(excluding this cover sheet) is 39.

NON-PROPRIETARY INFORMATION

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
RELATED TO EXTENDED POWER UPRATE REQUEST
VERMONT YANKEE NUCLEAR POWER STATION

PREFACE

The following information is provided in response to NRC's request for additional information (RAI) dated July 6, 2004, regarding the application by Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (Entergy or Vermont Yankee) for a license amendment to increase the authorized thermal power level of the Vermont Yankee Nuclear Power Station. At this time, Entergy is responding to seven of the nine individual RAIs in NRC's letter of July 6, 2004 (responses to RAIs SPSB-C-32 and 33 will be provided at a later date). In addition to the responses to seven of the RAIs in Reference 1, Entergy is also updating its response to a previous RAI, RLEP-C-5.

The individual RAIs are repeated as provided in NRC's letter of July 6, 2004.

The subject RAIs have been discussed during conference calls held between the staffs of the NRC and Entergy to further clarify the information needs of the NRC staff. In certain instances the RAIs were modified based on clarifications and understandings reached during the telecons. The information provided herein is consistent with those understandings.

For convenience, a list of frequently used EPU acronyms is included.

PROPRIETARY INFORMATION

This Attachment 2 is identical to Attachment 1, except it has been edited to remove Proprietary Information. The removed information has been deemed to be proprietary to the General Electric Company. Instances where proprietary information was deleted from the text are identified by double square brackets. The basis for the proprietary information is contained within the affidavit provided as Attachment 3.

NON-PROPRIETARY INFORMATION

List of Frequently Used Acronyms – Extended Power Uprate

ΔP	Differential Pressure
AC	Alternating Current
ADS	Automatic Depressurization System
ADHR	Alternate Decay Heat Removal
AI	Alternate Injection
AL	Analytical Limit
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
AOV	Air Operated Valve
ASME	American Society of Mechanical Engineers
AST	Alternative Source Term
ATWS	Anticipated Transients Without Scram
BOP	Balance-of-Plant
BWR	Boiling Water Reactor
BWROG	BWR Owners Group
BWRVIP	BWR Vessel Internals Project
CDF	Core Damage Frequency
CFD	Computational Fluid Dynamics
CFR	Code of Federal Regulations
CLB	Current Licensing Basis
CLTP	Current Licensed Thermal Power
CLTR	CPPU Licensing Topical Report
COLR	Core Operating Limits Report
CPPU	Constant Pressure Power Uprate
CR	Control Rod Insertion (event)
CRTP	Current Rated Thermal Power
CS	Core Spray
CUF	Cumulative Usage Factor
DAS	Digital Acquisition System
DBA	Design Basis Accident
DC	Direct Current
DL	Dynamic Loading
DW	Drywell
EAC	Environmental Advisory Committee
ECCS	Emergency Core Cooling System
ELTR	Extended Power Uprate Licensing Topical Report
ENN	Entergy Nuclear Northeast
EOP	Emergency Operating Procedure
EOS	Emergency Overspeed
EPR	Electric Pressure Regulator
EPRI	Electric Power Research Institute
EPU	Extended Power Uprate
EQ	Environmental Qualification
ESF	Engineered Safety Feature
FAC	Flow-Accelerated Corrosion
FEA	Finite Element Analysis
FFT	Fast Fourier Transform
FIV	Flow Induced Vibration

NON-PROPRIETARY INFORMATION

List of Frequently Used Acronyms – Extended Power Uprate
(continued)

FPCDS	Fuel Pool Cleanup and Demineralizer System
FW	Feedwater
G-K	Gido-Koestel
GDC	General Design Criterion
GE	General Electric
GEES	GE Energy Services
GENE	GE Nuclear Engineering
GL	Generic Letter
GNF	Global Nuclear Fuel
GRMS	Gravity Root Mean Square
HAZ	Heat Affected Zone
HELB	High Energy Line Break
HEM	Homogeneous Equilibrium Critical Flow Model
HEP	Human Error Probability
HP	High Pressure
HPCI	High Pressure Coolant Injection
HRA	Human Reliability Analysis
HTC	Heat Transfer Coefficient
HVAC	Heating Ventilation and Air Conditioning
Hx	Heat Exchanger
ICF	Increased Core Flow
IGSCC	Intergranular Stress Corrosion Cracking
INPO	Institute of Nuclear Power Operations
IPE	Individual Plant Examination
IST	Inservice Testing
JCO	Justification for Continued Operation
LOCA	Loss-of-Coolant Accident
LP	Low Pressure; Low Pressure Coolant Injection (event)
LPCI	Low Pressure Coolant Injection
LTP	Long Term Program
LTT	Large Transient Testing
MAAP	Modular Accident Analysis Program
MAX	Maximum
MDLM	Mist Diffusion Layer Model
MIN	Minimum
MOV	Motor Operated Valve
MPR	Mechanical Pressure Regulator
MS	Main Steam
MSL	Main Steam Line
MSIV	Main Steam Isolation Valve
MSSS	Main Steam Supply System
MWt	Megawatts Thermal
N/A	Not Applicable
NAI	Numerical Applications, Inc.
NDE	Non-Destructive Examination
NFPCS	Normal Fuel Pool Cooling Subsystem
NOS	Normal Overspeed
NPSH	Net Positive Suction Head

NON-PROPRIETARY INFORMATION

List of Frequently Used Acronyms – Extended Power Uprate
(continued)

NRC	Nuclear Regulatory Commission
NTU	Number of Transfer Units
OFS	Orificed Fuel Support
OLTP	Original Licensed Thermal Power
OOS	Out-of-Service
OPL	Operating Parameter List
PCIS	Primary Containment Isolation System
PCPL	Primary Containment Pressure Limit
PI	Project Instruction
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Analysis
PUSAR	Power Uprate Safety Analysis Report
QA	Quality Assurance
QAP	Quality Assurance Program
QAPM	Quality Assurance Program Manual
QC2	Quad Cities Unit 2
RAI	Request for Additional Information
RAW	Risk Achievement Worth
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Coolant System
RG	Regulatory Guide
RHR	Residual Heat Removal
RHRHX	Residual Heat Removal Heat Exchanger
RHRSW	Residual Heat Removal Service Water
RIPD	Reactor Internal Pressure Difference
RLA	Reload Licensing Analysis
RMS	Root Mean Square
RPV	Reactor Pressure Vessel
RPV-ED	Reactor Pressure Vessel Emergency Depressurization
RRU	Reactor Recirculation Unit
RTP	Rated Thermal Power
SAFDL	Specified Acceptable Fuel Design Limit
SBO	Station Blackout
SE	Safety Evaluation
SFP	Spent Fuel Pool
SFPCS	Standby Fuel Pool Cooling System
SGTS	Standby Gas Treatment System
SIL	Service Information Letter
SORV	Stuck Open Relief Valve
SQA	Software Quality Assurance
SRLR	Supplemental Reload Licensing Report
SRP	Standard Review Plan
SRSS	Square Root Sum of Squares
SRV	Safety/Relief Valve
SSC	Structure, System, and Component
SSE	Safe Shutdown Earthquake
SW	Service Water
TBCCW	Turbine Building Closed Cooling Water

NON-PROPRIETARY INFORMATION

List of Frequently Used Acronyms – Extended Power Uprate
(continued)

TC	Torus Cooling
TEF	Top of the Enriched Fuel
TRU	Turbine Recirculation Unit
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
VOQAM	Vermont Yankee Operational Quality Assurance Manual
VS	Vapor Suppression
VT	Containment Venting (event)
VY	Vermont Yankee
VYNPS	Vermont Yankee Nuclear Power Station
WW	Wetwell

NON-PROPRIETARY INFORMATION

License Renewal and Environmental Impacts Branch (RLEP)

RAI RLEP-C-5

How many full time employees and contractors work at VYNPS? Will the EPU affect the size of the labor force? Will the EPU have an affect on the labor force required for future outages? How many additional people are required for current outages?

Response to RAI RLEP-C-5

(Note: This RAI was originally answered in Entergy's letter of January 31, 2004, BVY 04-008. This response supplements the prior response based upon actual spring 2004 refueling outage statistics.)

Entergy completed the major EPU modifications during the spring 2004 refueling outage, which required approximately 425 more workers and supervisors than typical refueling outages. Normally, approximately 700 additional personnel are required for refueling outages; the spring 2004 refueling outage employed approximately 1,125 additional personnel.

Additional EPU-related modifications will be made during the next refueling outage. These remaining EPU modifications are less significant than those already implemented and are expected to require less than 100 additional workers to supplement typical outage staffing levels. Operation at EPU conditions is not expected to have any significant impact on future refueling outage staffing levels.

NON-PROPRIETARY INFORMATION

Plant Systems Branch (SPLB)

RAI SPLB-A-7

Spent Fuel Pool Cooling and Cleanup System:
(Reference 1, Attachment 6, [PUSAR] Section 6.3)

a) Spent Fuel Pool (SFP) Cooling Capacity:

Please describe the analyses that were performed and assumptions and input parameters that were used for the proposed EPU to address the following review criteria in NRC Review Standard, RS-001, Attachment 2 to Matrix 5, "Supplemental Spent Fuel Pool Cooling Review Criteria," Section 3.1.1.1:

- i) heat removal capability is based on bounding estimates of ultimate heat sink temperature, cooling system flow rates, and heat exchanger performance (e.g., fouling and tube plugging).
- ii) alternate heat removal paths (e.g., evaporative cooling) should be appropriately validated and based on bounding input parameter values (e.g., air temperature, relative humidity, and ventilation flow rate).

b) Heat Removal Capability and Limiting Case for Core Offload:

Table 6-3 in Attachment 6 to Reference 1 provides five SFP cooling/core offload configurations. Please update this table to include the following configurations discussed in the VYNPS Updated Safety Analysis Report (UFSAR), Section 10.5.5, page 10.5-9, third paragraph:

- i) Limiting Normal Batch (one-third core) Offload: One train (one heat exchanger and one pump) of Standby Fuel Pool Cooling Subsystem (SFPCS) in service, and
- ii) Limiting Full Core Offload: Both trains (two heat exchangers and two pumps) of SFPCS in service.

Also, discuss the assumptions and input parameters that were used in the analyses for the two additional configurations discussed above and confirm that they are consistent with the existing plant licensing basis and that the worst-case ultimate heat sink temperatures were used.

NON-PROPRIETARY INFORMATION

Response to RAI SPLB-A-7

a) Spent Fuel Pool (SFP) Cooling Capacity:

- i) As stated in the RAI Reference 1, Attachment 6 (PUSAR), Section 6.3, the fuel pool cooling system at VYNPS consists of three independent systems: the Normal Fuel Pool Cooling Subsystem (NFPCS), the Standby Fuel Pool Cooling System (SFPCS) and the Residual Heat Removal (RHR) system Augmented Fuel Pool Cooling (FPC) mode. The EPU heat removal capability analyses were based on bounding estimates of ultimate heat sink temperature, cooling system flow rates, and heat exchanger performance based on fouling and tube plugging as identified in the following Table SPLB-A-7-1. Note that the configurations presented in RAI Reference 1, Attachment 6 (PUSAR), Table 6-3 conservatively assume a start of fuel transfer from the reactor to the spent fuel pool at 24 hours after plant shutdown. In addition, the fuel transfer rate from the reactor to the spent fuel pool is conservatively assumed as 12 bundles/hour. The normal bundle transfer rate at VYNPS is typically 8 bundles/hour. These conservative assumptions lead to a higher decay heat load in the spent fuel pool than the demonstration of the system performance presented in the VYNPS UFSAR, Section 10.5.6 and Table 10.5.3, i.e., batch offload at six days decay and full core offload at ten day decay.
- ii) The alternate heat removal path using evaporative cooling was validated based on the bounding values shown in the following Table SPLB-A-7-2.

b) Heat Removal Capability and Limiting Case for Core Offload:

The two configurations presented in the VYNPS UFSAR, Section 10.5, page 10.5-9, present representative configurations of the SFPCS where one train (one-pump and 1 heat exchanger) has sufficient heat removal capacity for a batch offload and two trains (two pumps and 2 heat exchangers) have sufficient capacity for a full core offload assuming sufficient delay time between reactor shutdown and the beginning of fuel transfer. For the VYNPS UFSAR batch offload representative configuration, the batch offload is completed at six days after reactor shutdown and the representative full core offload is completed at ten days following reactor shutdown. In addition, the configurations presented in the VYNPS UFSAR, Section 10.5, page 10.5-9, present scenarios more conservative than Standard Review Plan (SRP) Section 9.1.3 in that the batch offload configuration assumes more than a single failure (failure of both the NFPCS trains and the failure of one SFPCS train), and the full offload configuration assumes at least a single failure (failure of both of trains of NFPCS). Configuration 2 presented in Table 6-3 of the PUSAR presents a batch offload scenario using conservative assumptions with a single failure. Configurations 3 and 4 presented in Table 6-3 of the PUSAR present abnormal (full core) offload scenarios, using conservative assumptions. The configurations in the PUSAR are consistent with the VYNPS licensing basis presented in Section 10.5.6 of the VYNPS UFSAR.

Additionally, the NRC staff asked about the results of Configuration 5 presented in Table 6-3 of the PUSAR (RAI Reference 1, Attachment 6). As discussed with the NRC staff during the July 8, 2004, telecon, Configuration 5 consisting of *Full Core Offload: With RHR Augmented FPC mode* alone, and the heat load in the RPV cooled by natural

NON-PROPRIETARY INFORMATION

circulation assumes a fuel pool heat removal capacity more limiting than that required by the SRP Section 9.1.3. Per Section III.1.d of SRP 9.1.3, for the abnormal maximum heat load (full core offload) the temperature of the pool water should be kept below boiling and the liquid level maintained with the normal systems in operation. A single active failure need not be considered for the abnormal case. Entry into Configuration 5 would require a failure of both the NFPCS and the SFPCS, e.g., multiple failures of the available fuel pool cooling systems. This configuration was included in the PUSAR only as a hypothetical case for providing the results of a scenario where only RHR in augmented FPC mode was available for cooling.

NON-PROPRIETARY INFORMATION

TABLE SPLB-A-7-1

System	Parameter	Value	Basis
SFPCS	Ultimate Heat Sink Temperature (Service Water to Heat Exchangers). Note that service water is the cooling medium for SFPCS HX.	85°F	Service Water Temperature is at Design Temperature consistent with Section 10.6.5 of the VYNPS UFSAR.
SFPCS	SFPCS Heat Exchanger flow rate (SFP side)/ pump	700 gpm	Consistent with VYNPS UFSAR Table 10.5.4
SFPCS	SFPCS Heat Exchanger flow rate (Service Water side)/ HX	700 gpm	Consistent with VYNPS UFSAR Table 10.5.4.
SFPCS	SFPCS Heat Exchanger Tube Fouling	20 percent	Consistent with current licensing basis calculation.
SFPCS	SFPCS Heat Exchanger Tube Plugging	5 percent	2.5 times that used in current licensing basis calculation (VYNPS UFSAR Table 10.5.1 Note 2 and UFSAR 10.5.4).
SFPCS	SFPCS Pump Heat addition to SFP per pump in operation	0.08 million BTU/hr	SFPCS Pump heat using head/flow, assuming efficiency of 72.5 %.
NFPCS	Ultimate Heat Sink Temperature (RBCCW to Heat Exchangers). Note that RBCCW is cooling medium for NFPCS HX.	100°F	Service water temperature is at design temperature consistent with Section 10.6.5 of the VYNPS UFSAR.
NFPCS	NFPCS Heat Exchanger flow rate (SFP side)/ pump	450 gpm	Consistent with VYNPS UFSAR Table 10.5.4.
NFPCS	NFPCS Heat Exchanger flow rate (RBCCW side)/ Heat Exchanger	350 gpm	Consistent with VYNPS UFSAR Table 10.5.4.
NFPCS	NFPCS Heat Exchanger Tube Fouling	20 percent	Consistent with current licensing basis calculation.
NFPCS	NFPCS Heat Exchanger Tube Plugging	No tube plugging	Consistent with current licensing basis calculation. Note that RBCCW uses demineralized water.
NFPCS	NFPCS Pump Heat addition to SFP per pump in operation	0.09 million BTU/hr	NFPCS Pump heat using head/flow, assuming efficiency of 72.5 %.
RHR Augmented FPC	Ultimate Heat Sink Temperature (Service Water to Heat Exchanger). Note that service water is cooling medium for RHR HX.	85°F	Service Water Temperature is at Design Temperature consistent with Section 10.6.5 of the VYNPS UFSAR.

NON-PROPRIETARY INFORMATION

TABLE SPLB-A-7-1
(continued)

System	Parameter	Value	Basis
RHR Augmented FPC	RHR Heat Exchanger flow rate (SFP side)	3000 gpm	Consistent with current licensing basis calculation
RHR Augmented FPC	RHR Heat Exchanger flow rate (Service Water side)	2700 gpm	Consistent with current licensing basis calculation
RHR Augmented FPC	RHR Heat Exchanger Tube Fouling	20 percent	Consistent with current licensing basis calculation
RHR Augmented FPC	RHR Heat Exchanger Tube Plugging	5 percent	Consistent with current licensing basis calculation (VYNPS UFSAR Table 10.5.1 Note 3)
RHR Augmented FPC	RHR Pump Heat addition to SFP per pump in operation	2.63 million BTU/hr	RHR Pump heat using head/flow

TABLE SPLB-A-7-2

Parameter	Value	Basis
Ambient air relative humidity on refuel floor for the SFP temperature calculation	100 percent relative humidity	Bounding value to minimize evaporation from Spent Pool Surface. This minimizes evaporative cooling of the spent fuel pool and maximizes the calculated spent fuel pool temperature.
Ambient air relative humidity on refuel floor for the SFP makeup water calculation	Zero percent relative humidity	Bounding value to maximize evaporation from Spent Pool Surface. This maximizes the calculation of make up water requirements to the spent fuel pool.
HVAC ambient temperature on the refueling floor	100°F	Bounding value. (Ref. UFSAR sections 5.3.5.2 and 10.12.3)
Ventilation flow rate	N/A	No credit taken for ventilation flow rate.

NON-PROPRIETARY INFORMATION

RAI SPLB-A-8

Service Water Systems (SWS):
(Reference 1, Attachment 6, Section 6.4)

a) In Section 6.4.1.1 of Attachment 6 to Reference 1, it is stated that:

"The performance of the safety-related portion of the SW system during and following the most demanding design basis event, the LOCA, was demonstrated. Adequate SW system heat transfer capabilities exist at CPPU [constant pressure power uprate] to support the above components. In addition, the SW flow rates do not change."

- i) With regard to performance, heat-loads, heat transfer capabilities, flow rates, and flow velocities in the SWS for post CPPU conditions, please explain how the above conclusions were reached.
- ii) Also, describe the analyses that have been performed, assumptions, and input parameters that were used; and explain the impact of the proposed EPU on UFSAR Section 10.6.4, Safety Design Bases, Items 1, 2, and 3.

b) Regarding the Residual Heat Removal Service Water (RHRSW) system, in Section 6.4.1.3 of Attachment 6 to Reference 1, it is stated that:

"The post-LOCA containment and suppression pool responses have been calculated based on an energy balance between the post-LOCA heat loads and the existing heat removal capacity of the RHR and RHRSW systems. As discussed in 3.5.2 and 4.1.1, the existing suppression pool structure and associated equipment have been reviewed for acceptability based on this increased post-LOCA suppression pool temperature....Thus, the RHRSW system has sufficient capacity to serve as the coolant supply for long-term core and containment cooling as required for CPPU conditions. The RHRSW system flow rate is not changed."

- i) With regard to performance, heat-loads, heat transfer capabilities, flow rates, and flow velocities in the RHRSW system for post CPPU conditions, please explain how the above conclusions were reached.
- ii) Also, describe the analyses that have been performed, assumptions, and input parameters that were used; and explain the impact of the proposed EPU on the UFSAR Section 10.7.4, Safety Design Bases, Item 1.

c) Confirm that the analyses performed for the proposed EPU are consistent with the existing plant licensing basis and that the worst-case ultimate heat sink temperature was used in calculating flow requirements of the safety-related SWS and the RHRSW systems for the proposed CPPU conditions.

NON-PROPRIETARY INFORMATION

- d) Please describe any impacts that the proposed EPU will have on the issues discussed in Generic Letters 89-13, "Service Water System Problems Affecting Safety-Related Equipment," 96-06, "Assurance of Equipment Operability and Containment Integrity during Design Basis Accident Conditions," and 96-06, Supplement 1, including the basis for your determination. In particular, confirm that water hammer and two-phase flow will not occur in the SWS, RHRSW, and other safety-related cooling water systems due to the EPU. Also, confirm that the power uprate will not result in overpressurization of water-filled piping between containment isolation valves.

Response to RAI SPLB-A-8

- a) The safety objective of the Service Water (SW) system is to provide cooling water to systems and equipment required to operate under accident conditions. The SW system design bases are further described in UFSAR Sections 10.6.2, 10.6.3, and 10.6.4.

The following safety related components serviced by the safety related portion of the SW system were evaluated for potential impact as a result of CPPU:

- RHR Heat Exchangers,
- Standby Fuel Pool Cooling System Heat Exchangers,
- Emergency Diesel Generator Coolers,
- ECCS Room Coolers (RRU 7 and RRU 8),
- RHRSW Pump Motor Coolers

The following review of each of the above safety related heat exchangers supports the discussion in CPPU Section 6.4.1.1 of Attachment 6 of Reference 1:

▪ RHR Heat Exchangers

The RHR Heat Exchangers are supplied with service water by the RHRSW pumps that take suction from the station SW system and return the "heated" water back to the SW system. Responding to the specific questions above:

i) Performance, heat loads, heat transfer capabilities

The CPPU analysis of the RHR system (Section 3.10 of PUSAR) and the Containment Analysis (Section 4.1 of PUSAR) reviewed all of the current loads on the RHR Heat Exchangers and any potential changes to these same loads at CPPU. These analyses concluded that no changes are required to the CLTP SW system supply input parameters including the SW flow rate to the RHR Heat Exchangers, the supply SW temperature, and the SW supply pressure.

As discussed in PUSAR Section 3.10, the RHR system is designed to operate in the following modes: LPCI mode, Shutdown Cooling (SDC), Suppression Pool Cooling (SPC), Containment Spray Cooling (CSC), and Fuel Pool Cooling (FPC) assist.

NON-PROPRIETARY INFORMATION

During the FPC assist mode, using the existing RHR heat removal capacity, adequate supplemental fuel pool cooling to the NFPCS system and the SFPCS is provided as needed. The RHR and the RHRSW parameters (i.e., RHR flow rate and RHRSW flow rate) remain unchanged.

During the SPC and CSC modes, the resulting higher CPPU suppression pool temperature (194.7°F) and containment pressure during a postulated LOCA, do not affect the hardware capabilities of the RHR equipment (i.e., the RHR flow rate or the RHRSW flowrate). As discussed in the response to RAI EMEB-B-6 (Entergy's January 31, 2004 submittal, BVY 04-008), the RHR pump seals were evaluated and requalified for the higher temperatures.

In the shutdown cooling (SDC) mode, the increased reactor decay heat will require a slightly longer time for cooling down the reactor. However, the calculated normal reactor shutdown time still satisfies the SDC design time criterion used as one of the bases for sizing the RHR heat exchangers. Therefore, there is no adverse impact on plant safety.

ii) Analyses, assumptions, input parameters, and design basis impacts

As described in the response to (i) above, the CPPU analysis was based on the CLTP parameters regarding SW conditions being unchanged at CPPU and concluded that these unchanged parameters were capable of supporting the CPPU operating conditions. Since there is no impact on the SW systems for the analyzed scenarios, the safety design basis discussed in UFSAR Section 10.6.4, Safety Design Basis, Item 1 remains valid.

• Standby Fuel Pool Cooling System

i) Performance, heat loads, heat transfer capabilities

The analysis performed and documented in Section 6.3.1 "Fuel Pool Cooling" evaluated the current loads on the SFPCS Heat Exchangers and any potential changes to these same loads at CPPU. This analysis concluded that no change is required to the CLTP SW system supply input parameters including the service water flow rate to the SFPCS Heat Exchangers, the supply SW temperature, and the SW supply pressure. See also response to RAI SPLB-A-7 for additional related information.

ii) Analyses, assumptions, input parameters, and design basis impacts

As described in the response to (i) above, the CPPU analysis was based on the CLTP parameters regarding SW conditions being unchanged at CPPU and concluded that these unchanged parameters were capable of supporting the CPPU operating conditions. Thus, since there is no impact on the SW systems for the analyzed scenarios, the safety design basis discussed in UFSAR Section 10.6.4, Safety Design Basis, Item 3 remains valid.

NON-PROPRIETARY INFORMATION

- Emergency Diesel Generator Coolers

- i) Performance, heat loads, heat transfer capabilities

The CPPU analysis performed related to station electrical loads under emergency operation / distribution conditions (Emergency Diesel Generators (EDG)) discussed in PUSAR Section 6.1.2, "AC Power (normal operation) consisted of evaluating the current loads on the emergency power system and evaluating any changes to these same loads at CPPU. The analysis concludes that there is no change to the EDG operating conditions because no increase in flow or pressure is required of any AC-powered ECCS equipment for CPPU. As such, the amount of power required to perform safety-related functions (pump and valve loads) is not increased with CPPU and the current emergency power cooling water requirements remain unchanged.

- ii) Analyses, assumptions, input parameters, and design basis impacts

Since there is no impact on the SW system to support the safety design basis of the emergency diesel generators, the UFSAR Section 10.6.4, Safety Design Basis, Item 2 is unaffected at CPPU.

- ECCS Pump Room Coolers

- i) Performance, heat loads, heat transfer capabilities

The CPPU analysis performed related to ECCS pump room heatup consisted of evaluating the changes in heat load to the room due to post accident CPPU conditions. Due to the slightly higher fluid temperature being pumped through the ECCS pumps during LOCA conditions, the heat load experienced by the ECCS pump room coolers will increase. The temperatures in the torus room and reactor building post LOCA are also increased at CPPU. As in the discussion related to the RHR heat exchangers, and the SFPCS, the design input into the analysis to determine the adequacy of the ECCS pump room coolers at CPPU, is that the supply service water temperature and flow rates do not change. The following assumptions were made in the ECCS pump room heatup analysis:

- The calculation assumes a loss of offsite power coincident with a LOCA,
- The ECCS corner room coolers in the lower level rooms (reactor recirculation unit (RRU) 7 and RRU 8) and their related fans operate,
- The heat sources, long term, in the lower corner room are one RHR pump and one core spray (CS) pump and the fan motor for the related room RRU (RRU 7 or 8). The heat sources for the upper corner room are one RHR SW pump and a fan motor for RRU 5 or 6. The heat from the RHR and CS piping, as well as the passive heat from the RHRHX is also modeled.

Note: although no credit is taken for the heat removal capability of RRU 5 or 6, the fan motor is modeled as a heat source.

NON-PROPRIETARY INFORMATION

- The RRU effectiveness corresponds to a one year post-LOCA fouling.
- The accident LOCA is assumed to occur at the beginning of the month with the highest river water temperature of 85°F (the month of August).
- The corner rooms are assumed to be initially at 100°F, a conservative assumption.

The resulting conclusion from the CPPU analysis is that the post-DBA LOCA ECCS corner room temperatures change from 155°F to 159°F for the lower ECCS corner room and from 153.4°F to 159.8°F for the upper ECCS corner room. The impact of this temperature increase on the equipment (EQ) in this environment has been determined to be acceptable.

ii) Analyses, assumptions, input parameters, and design basis impacts

Since the SW systems requirements to support the ECCS pump room coolers do not change from CLTP to CPPU, there is no impact on the UFSAR section 10.6.4, Safety Design Basis, Item 1.

• RHRSW Pump Motor Coolers

i) Performance, heat loads, heat transfer capabilities

The RHRSW Pumps supply the RHR heat exchangers. As discussed above, the SW supply input parameters to the RHR heat exchanger remain unchanged at CPPU conditions. Since the operating parameters for the RHRSW pumps remain unchanged, there is no change in heat removal capability requirements.

ii) Analyses, assumptions, input parameters, and design basis impacts.

Since the SW system's requirements to support the RHRSW pump motor cooler operation do not change from CLTP to CPPU, there is no impact on the UFSAR Section 10.6.4, Safety Design Basis, Item 1.

b) The RHRSW system safety design bases (see UFSAR Section 10.7.4) are:

1. Provide sufficient cooling capacity for the RHR System during design basis accident.
2. Minimize the probability of release of radioactive contaminants to the environs.

The RHRSW system consists of RHRSW pumps, RHR heat exchangers and the piping, valves, and instrumentation necessary to ensure system operation. The RHRSW pumps are supplied from the SW system and the RHRSW system returns water back to the SW system. The discussion above regarding the CPPU capabilities of the safety related SW system to supply the RHR heat exchangers applies to the RHRSW system since the RHRSW system is dedicated to providing SW to the RHR heat exchangers.

NON-PROPRIETARY INFORMATION

The conclusions reached in response to item (a) above for the SW supply temperatures to the RHRSW also apply here.

- c) The existing plant licensing basis and the worst-case ultimate heat sink temperature of 85°F was used in calculating flow requirements of the safety-related SWS and the RHRSW system for the proposed CPPU conditions.
- d) From the above, no change to the SW and the RHRSW flow rates, pressures and temperatures are required as a result of CPPU. In addition, key heat exchanger parameters (e.g., fouling factors, effectiveness, and tube plugging allowance) used in the CPPU analyses, remain consistent with the existing GL 89-13 program. Thus, current evaluations, testing, and monitoring performed to support GL 89-13, "Service Water System Problems Affecting Safety-Related Equipment," will support CPPU operation.

GL 96-06, "Assurance of Equipment Operability and Containment Integrity during Design Basis Accident Conditions," including Supplement 1, was reviewed for CPPU operation with the conclusion that the potentially impacted system is the RBCCW system, not the SW or the RHRSW system. The GL 96-06 evaluations for impact on the RBCCW system conclude that:

- Over-pressurization of the RBCCW system is not an issue due to installed over-pressurization protection on this system. Furthermore, VYNPS does not rely on the use of the RRUs inside containment that are serviced by RBCCW for containment heat removal.
- The CLTP analysis performed to evaluate RBCCW voiding and potential water hammer during a DBA LOCA and main steam line break accident used drywell pressure and temperature input values that bound the CPPU drywell conditions.

The impact relative to the issues raised in Generic Letter 96-06 for CPPU conditions was found to be acceptable. See additional discussion in PUSAR Section 4.1.6.

NON-PROPRIETARY INFORMATION

RAI SPLB-A-9

Ultimate Heat Sink (UHS) / Alternate Cooling System (ACS):
(Reference 1, Attachment 6, Section 6.4.5)

a) In Section 6.4.5 of Attachment 6 to Reference 1, it is stated that:

“The ACS was evaluated for CPPU in a manner that is similar to the UHS evaluation for newer plants (e.g., inventory requirements and heat removal capability with increased decay heat)...The heat removal requirements of the following affected components during the ACS operating mode have been evaluated and found to be acceptable at CPPU...”

- i) With regard to performance, heat-loads, heat transfer capabilities, flow rates, and flow velocities in the ACS for post CPPU conditions, please explain how the above conclusions were reached.
- ii) Also, describe the analyses that have been performed, assumptions, and input parameters that were used; and explain the impact of the proposed EPU on UFSAR Section 10.8.2, Safety Design Bases, Items 1, 2, and 3.

b) In Reference 5, Attachment 6, MATRIX 5, Page 8, SE 2.5.3.4, it is stated that no SW flow or SW supply temperature changes are required to support the CPPU normal, LOCA or shutdown operations. Please explain.

c) Confirm that the analyses performed for the proposed EPU are consistent with the existing plant licensing basis and that the worst-case ultimate heat sink temperature was used in calculating flow requirements of the ACS for the proposed CPPU conditions.

d) In Reference 1, Attachment 6, Section 6.4.5, as well as in Reference 5, Attachment 6, MATRIX 5, Page 8, SE 2.5.3.4, it is stated that a modification to re-circulate ACS (RHRSW) pump motor cooler water back to the cooling tower, instead of discharging it to the river, is planned to ensure adequate inventory to meet the 7-day requirement associated with the ACS design-basis functional scenario. Please provide a description of the modification, including a flow diagram. In addition, discuss the regulatory requirements applicable to the modification.

Response to RAI SPLB-A-9

a)(i) For the alternate cooling system (ACS) mode of operation of the service water system the following are analytical constraints on thermal/hydraulic conditions in the system.

- Thermal Constraints:
 - 1. Return temperature to cooling tower $\leq 130^{\circ}\text{F}$ to protect fill material.
 - 2. Spent fuel pool $\leq 150^{\circ}\text{F}$ (higher limits are allowed under upset conditions, but for conservatism, this normal design limit is used for ACS).
 - 3. RHRSW $\leq 150^{\circ}\text{F}$ (becomes a constraint on heat removal rate from RHR heat exchangers).

NON-PROPRIETARY INFORMATION

4. Torus temperature $\leq 176^{\circ}\text{F}$ (conservatively set for margin in meeting NPSH requirements for ECCS pumps).
 5. Reactor cooldown rate $\leq 90^{\circ}\text{F/hr}$ (this is an administratively set limit to remain below the Tech Spec limit of 100°F/hr).
 6. Be in safe shutdown condition, that is, subcritical with the ability to transfer decay heat (core and spent fuel pool) and primary system sensible heat to the ultimate heat sink. Although VYNPS' license does not require it to achieve cold shutdown condition following this event, for conservatism, the analysis was performed on the basis that following event initiation, it is desirable to be in cold shutdown condition within 24 hours.
- Hydraulic constraints:
 1. Total flow to the cooling tower must be within the limits for which test data are available to support performance projections. Total flowrate must be ≥ 3500 gpm and ≤ 8800 gpm.
 2. Maintain positive margin on basin water inventory for 7 days of continuous ACS operation.
 3. Maintain positive NPSH margin on RHRSW pumps.
 - Analysis:

The thermal and hydraulic analyses that are conducted to evaluate ACS operation against the criteria listed above are separate but related tasks that must be carried out in parallel because of the interplay between them. The following conclusions were reached as a result of this analysis:

1. Initially, a large amount of cooling water is required to remove the amount of decay heat being generated and under worst case conditions, two pumps are required per train to provide this amount of cooling water without violating NPSH requirements.
2. At CLTP, it was determined that after 48 hours of operation, the number of operating pumps had to be reduced to one per train to maintain positive inventory. Running only one pump per train reduced the water loss because there is no motor bearing cooling water loss for the isolated pump and the EDG load is reduced and thus the evaporative losses for the heat load due to EDG operation are reduced.
3. Determination of the point when the system must be switched to two pump operation and the flowrate required per pump after this time step is an iterative process that balances the need for decay heat removal against the need for maintaining pump NPSH and basin inventory.
4. At CPPU there is an increase in the decay heat rate for both the core and spent fuel pool, but analysis determined that for the worst case design conditions, no changes would be required in the values currently specified in the ACS operating procedures for total system flow rate, number of operating RHRSW pumps and time step for reduction in number/flow rate of operating RHRSW pumps.
5. At CPPU, because the higher decay heat rates result in an increase in evaporative losses from the cooling tower basin, several design changes

NON-PROPRIETARY INFORMATION

were determined to be necessary to preserve both water inventory and pump NPSH margin.

a)(ii) Thermal performance of the SW system during the ACS mode of operation is evaluated using a computer model that was created on an Excel spreadsheet. For pre-specified incremental progressions in time over the postulated 7 day ACS event, the program performs a mass/energy balance to conservatively predict the effectiveness of the cooling tower as the ultimate heat sink for all heat rejected to the closed loop RHRSW system during this abnormal event. The program also calculates the reduction in basin inventory due to evaporative, drift and pump cooler losses. There are 40 user defined inputs to this spreadsheet. The inputs that change as a result of CPPU are as follows:

1. Q cooldown – 1.39×10^8 BTU - differential in sensible heat content of the reactor coolant system (RCS) and internals from hot shutdown condition to SDC point (50 psig). This assumes that entire RCS and internals are at saturation conditions for 550 °F. For CPPU conditions, the heat content is less due to differences in core design, but for conservatism, the pre-CPPU value is retained.
2. RRU heat load – 9.42×10^5 BTU/hr/train. Although the actual load for post-CPPU operation is greater than that for pre-CPPU operation, this value has not been changed since it already conservatively assumes post-LOCA operation conditions, which are approximately 3 times greater than ACS conditions.
3. Maximum spent fuel pool heat load – 1.48×10^7 BTU/hr. This is the heat load at the start of the ACS event and is based on the conservative assumption of the event occurring immediately after a short duration refueling outage of only 6 days. The pre-CPPU value for this input was 1.1×10^7 BTU/hr. For post-CPPU operation, the curve used for the 7-day, spent fuel pool heat load is based on the methodology in ANS 5.1, as opposed to the methodology of BTP ASB 9-2 which was used for the pre-CPPU analysis. As a result, total integrated heat load over 7 days is now slightly less than that in the pre-CPPU analysis (1.55×10^9 BTU vs. 1.59×10^9 BTU). However, the analysis is still very conservative, since it is based on the assumption of a 6 day refueling outage and does not take any credit for heat losses to the concrete or air above the pool. (Note: the decay heat rate is incorporated into the spreadsheet in the form of a curve fit formula for the design basis fuel pool decay heat rate developed by GE and as documented in GE-NE-0000-0015-1737-01. As such, it is not an input that can be varied by the program user.)
4. RHRSW pump motor bearing cooler loss – A value of 4 gpm/pump was used for the pre-CPPU analysis. Modifications reduce this to zero gpm/pump for post-CPPU operation.
5. Core decay heat is based on a maximum thermal power level of 1950 MWt for post-CPPU operation vs 1593 MWt for pre-CPPU operation. Total integrated decay heat load over the postulated seven day ACS event increases from 4.41×10^9 BTU to 5.44×10^9 BTU. (Note: this input is incorporated into the spreadsheet in the form of a curve fit formula for the design basis decay heat rate developed by GE and as documented in GE-VYNPS-AEP-146. As such, it is not an input that can be varied by the program user.)

For each time step in the thermal analysis, the program user first estimates the tower outlet (cold) temperature, from which the tower performance characteristic (BTUs

NON-PROPRIETARY INFORMATION

removed per °F tower range per lbm air flow) is determined using the Merkel Theory and compared with the tower performance characteristic calculated using vendor supplied performance data. This process is iterated using different estimates of cold outlet temperature until the Merkel Number agrees with the tower vendor data. Once the cold outlet temperature is found, the heat removal and evaporative losses for each time step are determined using the standard equations for steady state heat transfer.

The hydraulic analysis was performed using Pipe-Flo, which is a commercially available computer program for calculating flow rates in fluid networks. The program also computes pressure in the system at any desired point, so by designating the pump suction connection as one such point, the available NPSH for the pump under any given set of flow conditions can be determined. The required inputs to this program are listed below:

1. Fluid – warm water having an average temperature of 85°F.
2. Piping materials, sizes, fittings and lengths – from as-built piping physical drawings.
3. Piping roughness – the roughness varies in different sections of the system due to differences in materials and flowrates, which in turn affect the amount of microbiological induced corrosion (MIC) that can be expected. To confirm that conservative roughness factors are being used in the model, benchmark testing was performed against several sets of field measurements of pressure drops for various flowrate conditions.
4. Pump curve – For conservatism, the vendor pump curve was uniformly degraded 20% for use in the calculations. Note that the ASME Code requires corrective action to be taken if the actual pump performance at any point drops 6% below the certified test curve. So use of a 20% degraded pump curve is very conservative.

Note that these inputs are unchanged for power uprate since the required flowrates for each SW user are the same as those required prior to EPU and since no modifications were required in the piping (other than pump suction barrel material and ¾ inch lines for cooling water to the RHRSW pump motor bearing coolers, which are not included in the model because of their small flowrates). The proposed EPU has no adverse impact on the UFSAR Section 10.8.2 Safety Design Bases, Items 1, 2 and 3.

b) The equipment supplied by the service water system include:

- RBCCW Heat Exchangers
- TBCCW Heat Exchangers
- Generator H₂ Coolers
- Generator Stator Water Coolers
- Generator ALTERREX Coolers
- Standby FPC heat Exchangers
- Reactor Feedwater Pump Area Coolers (TRU-1, 2, 3, 4)
- Condensate Pump Area Cooler (TRU-5)
- Turbine Lube Oil Coolers
- Reactor Recirc System MG Set Lube Oil Coolers

NON-PROPRIETARY INFORMATION

- Circulating Water Pump A, B, C, motor coolers and gland seals
- MS and Feedwater Pipe Tunnel Coolers (RRU-17A, RRU-17B)
- Reactor Building Air Handling Units RRU 10 through RRU 16
- Reactor Building Air Conditioning 1A through 1D
- Administrative Building Water Cooled Chiller SCH 2
- Standby Gas Treatment Loop Seals
- Sample Coolers for Heating Boiler
- RHRSW Pump Motor Coolers
- Emergency Diesel generator Coolers
- ECCS Pump Room Coolers RRU 5, 6, 7, and 8
- RHRSW supply to the RHR Heat Exchangers
- Fire Protection Pressurization Line

Normal Operation:

During Normal Operation, the following service water system loads can potentially be affected by CPPU:

- RBCCW Heat Exchangers – There is a slight increase in heat load of approximately 0.6%. See also discussion in PUSAR section 6.4.3. There is no change to the service water flow rate or supply temperature.
- TBCCW Heat Exchangers – There is no increase in heat load. See discussion in PUSAR section 6.4.4. There is no change to the service water flow rate or supply temperature.
- Generator H2 Coolers – There is a slight increase in heat load to the service water system of about 2%. The SW design supply temperature is 85°F, which bounds the required 92°F supply temperature to the Generator H2 Coolers. Also, the required SW flow decreases about 14% because of a change in design requirements.
- Generator Stator Water Coolers – Due to a design change, there is a decrease of about 13% in the heat loads from the Generator Stator Water Coolers. The SW flow rate decreases and the maximum allowed SW supply temperature of 95°F is higher than the SW design temperature of 85°F. Also, the SW required flow decreases about 14%.
- Generator ALTERREX Coolers – There is no change in the Generator ALTERREX Coolers heat load, SW flows or supply temperatures.
- Standby FPC heat Exchangers – No change to the service water flow rate or supply temperature. See discussion in PUSAR section 6.3 and related discussion regarding heat rates, flows and supply temperatures in response to RAI SPLB-A-7 above.
- Reactor Feedwater Pump Area Coolers (TRU-1, 2, 3, 4) – There is an increase of approximately 36% in the heat load at CPPU compared to CLTP. However, this increase in heat load does not result in adverse temperatures. The SW supply temperature and flow rates remain unchanged.
- Condensate Pump Area Cooler (TRU-5) – There is a small increase (i.e., approximately 9%) in the heat load at CPPU compared to CLTP. This increase is acceptable because area equipment is not adversely affected. The SW supply temperature and flow rates remain unchanged.
- Turbine Lube Oil Coolers – No change in heat load, flow rates, or temperatures.

NON-PROPRIETARY INFORMATION

- Reactor Recirc System MG Set Lube Oil Coolers – No change in heat loads, flow rates, or temperatures.
- Circulating Water Pump A, B, C, motor coolers and gland seals -- No change to the service water heat load, flow rate, or supply temperature since there is no change to circulating water system flows at CPPU.
- MS and Feedwater Pipe Tunnel Coolers (RRU-17A, RRU-17B) – There is an insignificant increase in the area temperature of approximately a 0.6°F. No change in SW flow rates or supply temperature.
- Other Equipment Supplied by Service Water – The RHRSW pump motor coolers, the EDG coolers, and RRUs 5, 6, 7 and 8 were addressed in the response to RAI SPLB-A-8. The remaining equipment heat loads are not significantly affected by EPU and there is no change in the SW flow rate or supply temperature.

Conclusions for Normal Operation:

- The SW design temperature limit of 85°F bounds all of the equipment above for CPPU operation.
- There is a slight decrease in the total required SW flow.
- There is a slight decrease in the total heat removed by the SW system.

LOCA or Shutdown Events:

The limiting scenario is the LOCA scenario. As such, the conclusions reached in the response to RAI SPLB-A-8 above are applicable.

- c) The service water supply temperature is in accordance with the plant design basis of 85°F as discussed in UFSAR section 10.6.5. For ACS operation, see the above response to part (a).
- d) For purposes of the following discussion, refer to the SW system flow diagram included below as Figure SPLB-A-9-1. For each pump in each train of the RHRSW system, this minor modification involves the addition of a new 3-way ball valve in the motor bearing oil cooling water (MBOCW) return piping that will allow routing of the cooling water back to the pump suction line during ACS mode operation only to save cooling tower basin water inventory. This new line will connect to the existing 14-inch diameter pump suction piping. The second outlet port of the three-way ball valve will be connected to the existing piping that connects to the reactor building storm drains.

This change will have negligible impact on system hydraulics or the PRA model. The new ball valve is a full-ported type that offers minimal resistance to fluid flow. For normal plant operation and post-LOCA recovery, the system will be configured and operated in a manner that is identical to that called for by the current design, except that the cooling water will be routed through the three-way ball valve. Since the system will normally be set in this alignment, no additional valve manipulation is required and the increase in human error probability (HEP) for normal or post-LOCA operation is consequently very small. For ACS operation, the only new operator action required is to reposition the 3-way ball valve. Therefore, for ACS operation the increase in HEP is very small.

NON-PROPRIETARY INFORMATION

When switching to the ACS mode of operation, an additional procedural step will be required to re-position the three-way ball valve to align it with the branch line to the pump suction. The valve will be mounted in a convenient spot near the pump. Therefore, the two hour limit on set-up time for ACS operation will not be jeopardized. Set-up time was significantly reduced in 2001 when a permanent cross-tie to the standby fuel pool cooling system was installed, eliminating the labor-intensive and time-consuming work associated with installing the temporary spool piece originally required for making this cross-tie. Operators are expected to already be in the area for related tasks that can be accomplished in a few simple steps. Thus, there is currently considerable margin between the time that it actually takes to set-up the system and the two hour design limit. In addition, realignment of this ball valve can be done simultaneously with other required valve manipulations, rather than in series with them.

From a hydraulic performance point of view, aligning the MBOCW discharge to the pump suction rather than to the storm sewers actually represents a considerable improvement since the hydraulic resistance is significantly reduced (due to the shorter length of piping and lower discharge point back pressure).

The pipe routing for each new MBOCW return line will have no impact on other safety related SSCs since the RHRSW pump which it services is the only SSC in the vicinity of the new line.

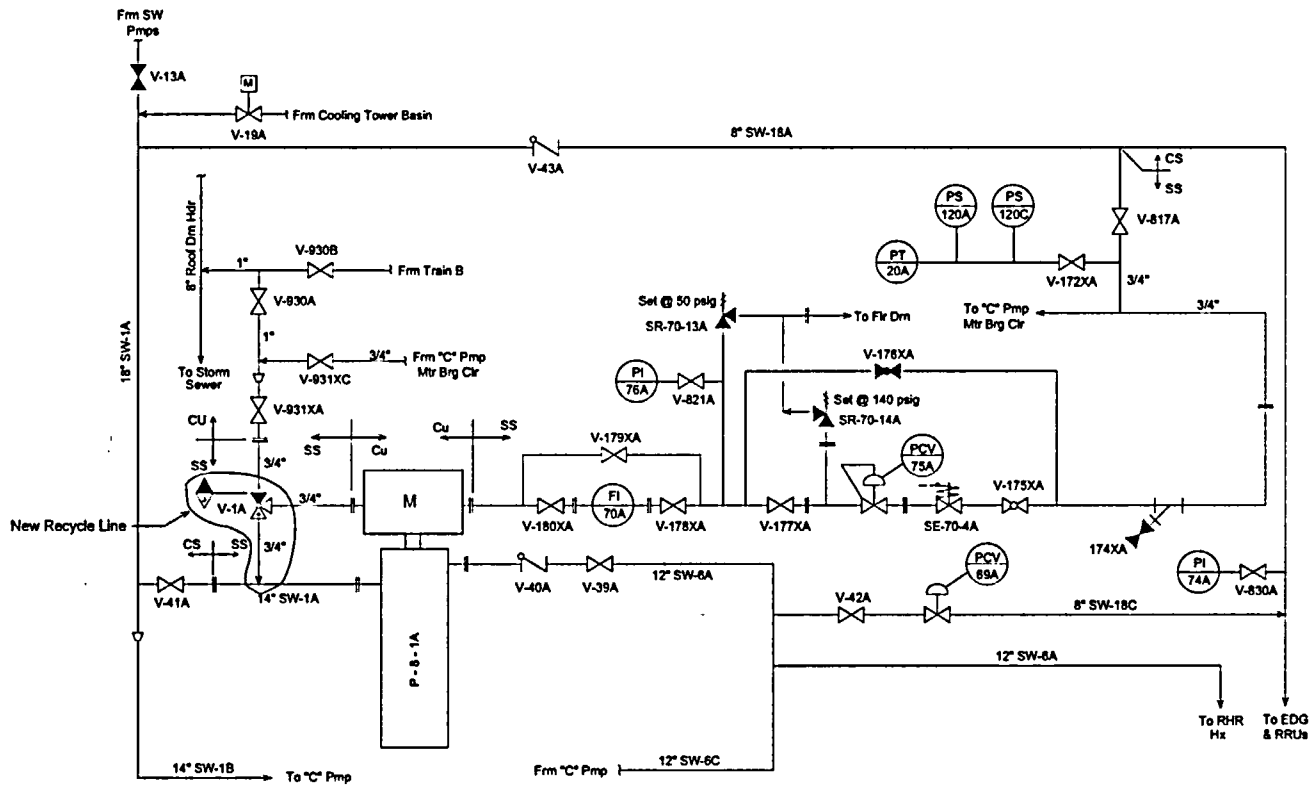


FIGURE 1 - Modification for Recycling Cooling Wtr to RHRSW Pump Motor Bearing Oil Coolers

Figure SPLB-A-9-1

NON-PROPRIETARY INFORMATION

NON-PROPRIETARY INFORMATION

Probabilistic Safety Assessment Branch (SPSB)

RAI SPSB-C-28

Provide additional information regarding the potential impact of the CPPU on those HVAC systems discussed in the Standard Review Plan sections 6.4, 6.5.1, 9.4.1, 9.4.2, 9.4.3, 9.4.4 and 9.4.5. This should include a discussion of the impact, if any, during both normal and post-accident operations resulting from increases in heat loads due to CPPU and the bases for your determination of system acceptability post-CPPU.

Response to RAI SPSB-C-28

The following Table SPSB-C-28-1 provides a list of the heating, ventilation, and air conditioning (HVAC) systems discussed in the Standard Review Plan (SRP) sections noted in this RAI. It should be noted that VYNPS is not a SRP plant.

Table SPSB-C-28-1

SRP Section	Title
6.4	Control Room Habitability Systems
6.5.1	ESF Atmospheric Cleanup Systems
9.4.1	Control Room Area Ventilation System
9.4.2	Spent Fuel Pool Area Ventilation System
9.4.3	Auxiliary and Radwaste Area Ventilation System
9.4.4	Turbine Area Ventilation System
9.4.5	Engineered Safety Feature Ventilation System

Documents referenced in this response are available for review at VYNPS upon request.

Control Room Habitability Systems

The control room habitability aspect of the constant pressure power uprate (CPPU) is addressed in Entergy's Alternative Source Term (AST) license amendment request for VYNPS that was submitted to the NRC by letter dated July 31, 2003 (BVY 03-70).

ESF Atmospheric Cleanup Systems

The ESF atmospheric cleanup system at VYNPS is the standby gas treatment system (SGTS). As discussed in Section 4.5 of the Power Uprate Safety Analysis Report (PUSAR) (i.e., Attachment 4 to Entergy letter dated September 10, 2003), the acceptability of the SGTS at VYNPS was confirmed by reviewing plant specific data at CPPU conditions against the criteria stated in Section 4.5 of the PUSAR. With respect to heat loads due to CPPU and the basis for determination of system acceptability post-CPPU, the SGTS is acceptable for CPPU conditions if the SGTS inlet temperature (normal and post accident operation) is below [[]] degrees F. The VYNPS secondary containment in both normal and accident conditions was confirmed to be below [[]] degrees. Therefore, the SGTS is acceptable for CPPU.

NON-PROPRIETARY INFORMATION

Control Room Area Ventilation System

The heat loads in the control room are not a function of reactor power level. Heat sources in the control room are from electrical equipment, ambient outside air temperature, and personnel. None of these sources are expected to increase at CPPU conditions. Therefore, the control room HVAC system's ability to provide appropriate temperature and humidity conditions for personnel and equipment during all modes of operation and emergency conditions is not impacted by CPPU. In addition, CPPU has no impact on the control room HVAC system's ability to provide for heating during cold weather conditions.

The control room habitability aspect of this system is discussed in the July 31, 2003 AST submittal, as noted in the control room habitability systems section above.

Spent Fuel Pool Area Ventilation System

VYNPS does not have a separate spent fuel pool (SFP) area ventilation system. The SFP area is serviced by the reactor building HVAC system.

The fuel pool cooling and demineralizer system (FPCDS) was evaluated for CPPU for both batch and full core off-loads. For normal operation, it was determined that although the decay heat load would increase for CPPU, the SFP temperature would remain within current limits. Therefore, there is no impact on the heat load to the reactor building HVAC during normal operation.

Post-LOCA, the reactor building HVAC isolates and SGTS initiates. Refer to the response to RAI SPSB-C-14 (in Entergy letter dated July 2, 2004) for a discussion of the post-LOCA reactor building heatup analysis results.

Auxiliary and Radwaste Area Ventilation System

The heat loads in the radwaste building are not a function of power level. Therefore, the radwaste building HVAC system is acceptable for CPPU operation, and its ability to vent potentially contaminated air is not affected by CPPU.

The offgas building ventilation system maintains a suitable environment for operating personnel and equipment as required to ensure proper operation of the equipment. The CPPU evaluation noted that while H₂ production is linear with respect to core thermal power, the operating temperatures of the recombiner, following CPPU, will remain at or below the design basis temperature of 655°F. An evaluation of the operating temperature of the recombiner room indicates an increase of 3°F or less at CPPU, which is within the capabilities of the offgas ventilation system.

Radwaste building HVAC and offgas ventilation are not credited during post-accident conditions.

Turbine Area Ventilation System

Increases in area heat gain and ambient air temperatures, as a result of CPPU, are predominantly caused by increases in operating temperature of piping systems, equipment, and

NON-PROPRIETARY INFORMATION

air-cooled motors operating under increased loads. For CPPU, it was determined that the following areas (as identified in Table SPSB-C-28-2) serviced by the turbine building HVAC would experience temperature increases.

Table SPSB-C-28-2

Area	CPPU Ambient Temperature Increase (°F)
LP Heater Area	4.1
HP Heater Area	1.7
Feedwater Pump Room	7.6
Condensate Pump Room	3.5

It was noted in Section 6.6 of the PUSAR (i.e., Attachment 4 to Entergy's September 10, 2003 license amendment request) that the 105°F design ambient room temperature may be exceeded for the condensate pump and feedwater pump rooms during the summer under CPPU conditions. This aspect of CPPU was subsequently evaluated and found to be acceptable. The evaluation was submitted to the NRC as EPU submittal Supplement No. 7, dated May 19, 2004 (BVY 04-50).

The turbine building HVAC is not credited during post-accident conditions.

Engineered Safety Feature Ventilation System

The emergency core cooling system (ECCS) corner rooms are cooled by reactor recirculation units RRU-5, RRU-6, RRU-7, and RRU-8, in addition to outside air provided by reactor building HVAC. At CPPU, normal heat loads and ambient temperatures do not increase. Therefore, the ability of RRU-5, RRU-6, RRU-7 and RRU-8 to maintain acceptable area temperatures during normal operation is unchanged.

Post-LOCA evaluation of ECCS corner rooms was performed using the GOTHIC code, as discussed in the response to RAI SPSB-C-14 (in Entergy's July 2, 2004 submittal).

RAI SPSB-C-29

Please provide the design basis and realistic values of inputs used in the determination of emergency core cooling system (ECCS) pump available net positive suction head (NPSH) (i.e., the values used in the MAAP probabilistic risk assessment (PRA) calculations and the SHEX calculations). Please include:

- a) service water temperature
- b) initial containment temperature
- c) initial containment pressure
- d) initial drywell and wetwell humidity
- e) initial suppression pool temperature
- f) drywell and wetwell airspace volume
- g) suppression pool water volume

NON-PROPRIETARY INFORMATION

Response to RAI SPSB-C-29

The design basis and realistic values are provided in the following Table SPSB-C-29-1.

The determination of emergency core cooling system (ECCS) pump available NPSH for the power uprate license application was based on the results of the SHEX calculations. There were no NPSH determinations based on MAAP calculations.

SHEX input parameters are selected to maximize the calculated suppression pool temperature and minimize the calculated wetwell pressure. MAAP input parameters are based on a more realistic depiction of the containment and heat removal systems.

The drywell airspace volume is characterized differently in SHEX than in MAAP. SHEX includes the pedestal and the vent system in the drywell volume, while MAAP splits the drywell into three regions called (1) drywell, (2) pedestal, and (3) DW vents, ring header, and downcomer pipes.

NON-PROPRIETARY INFORMATION

Table SPSB-C-29-1

Parameter	SHEX	MAAP or Realistic
a) service water temperature	85°F	32°F to 85 °F ⁽¹⁾
b) initial containment temperature:		
drywell	170°F	150°F
wetwell	90°F	90°F
pedestal	N/A	150°F
DW vents, ring header and downcomer pipes	N/A	120°F
c) initial containment pressure:		
drywell	16.4 psia	16.55 psia
wetwell	14.7 psia	14.55 psia
pedestal	N/A	16.55 psia
DW vents, ring header and downcomer pipes	N/A	16.55 psia
d) initial humidity:		
drywell	100%	50%
wetwell	100%	100%
pedestal	N/A	50%
DW vents, ring header and downcomer pipes	N/A	70%
e) initial suppression pool temperature	90°F	50°F to 90°F
f) airspace volumes:		
drywell	128,370 ft ³	127,122 ft ³
wetwell	105,900 ft ³	104,900 ft ³ ⁽²⁾
pedestal	N/A	7,078 ft ³
DW vents, ring header and downcomer pipes	N/A	16,606 ft ³
g) suppression pool water volume	68,000 ft ³	69,000 ft ³ ⁽³⁾
h) Total containment concrete heat sink area ⁽⁴⁾	2,068 ft ²	20,419 ft ²
i) Total containment steel heat sink area	34,001 ft ²	37,761 ft ²
j) RHR Heat Exchanger Heat Rate ⁽⁵⁾	52 MBtu/hr	> 59 MBtu/hr
k) Initial Core Thermal Power	1950 MWt	1912 MWt
l) Decay Heat Uncertainty (ANS 5.1-1979 Standard)	2 sigma	none
m) Pump Heat (Long-Term) 3 RHR + 2 CS	4129.2 HP (2918 Btu/sec)	Not Modeled
n) Feedwater Mass and Energy injected into containment post-LOCA	565,077 lb 1.54E+08 Btu	4,924 lb 1.81E+06 Btu ⁽⁶⁾

- (1) Maximum average hourly measured river temperatures from 1970 through 1989 had a mean of 78.9 °F with a standard deviation of 1.6 °F.
- (2) The wetwell airspace volume plus the suppression pool volume equals 173,900 ft³.
- (3) The Technical Specifications minimum and maximum values are 68,000 ft³ and 70,000 ft³, respectively.
- (4) MAAP includes the concrete surface area adjacent to the drywell outer wall. The drywell steel is modeled as a liner for heat sink purposes.
- (5) The RHR heat exchanger heat rates are provided for comparison purposes. The design basis heat rate is based on maximum allowable tube plugging (5%) and design fouling factor at a shell-side inlet temperature of 165 °F and a tube-side inlet temperature of 85°F. The realistic value is a projected heat rate based on test performance data adjusted to the design temperatures and 5% tube plugging.
- (6) SHEX conservatively assumes feedwater will continue to inject as long as its temperature exceeds that of the suppression pool. The MAAP/Realistic value assumes feedwater pumps trip at start of event.

NON-PROPRIETARY INFORMATION

RAI SPSB-C-30

Please describe how containment leakage is modeled in the design basis NPSH calculations. Is MSIV leakage included? If not, why not?

Response to RAI SPSB-C-30

A primary containment leakage rate of 1.5 wt.-% per day was used in the NPSH calculations. This value includes the MSIV leakage rate.

The GE SHEX code is used to calculate the containment conditions used in the NPSH evaluation. For containment calculations where leakage is modeled the SHEX code [[

]]

The primary containment leakage rate (L) used in the NPSH calculations is based on the leakage rates used in the AST proposed license amendment¹. The AST proposed license amendment included an Appendix J exemption to remove the MSIV leakage term from the maximum allowable leakage rate (La) term. The AST analysis supported maintaining the La of 0.8 wt.-% per day and increasing the MSIV leakage to an aggregate of 124 standard cubic feet per hour (scfh). In addition, the AST analysis assumed a secondary containment bypass leakage of 5 scfh as part of the La term. The overall primary containment leakage rate used in the EPU containment response and net positive suction head (NPSH) calculations was conservatively selected to bound the total AST assumed primary containment leakage value, including MSIV leakage.

¹ See Entergy letter BVI 03-70, dated July 31, 2003, as supplemented.

NON-PROPRIETARY INFORMATION

The total AST primary containment leakage is obtained by adding the MSIV and La components. The unit conversion of the MSIV mass flow rate to primary containment wt-%/day is calculated as follows:

$$124 \text{ scfh} \times \left[\frac{14.7 \text{ psia}}{(44 + 14.7) \text{ psia}} \right] \times \left[\frac{(460 + 338)R}{(460 + 68)R} \right] = 46.9 \text{ cfh}$$
$$46.9 \frac{\text{ft}^3}{\text{h}} \times \left[\frac{24 \text{ h}}{\text{d}} \right] \times \left[\frac{1}{232,302 \text{ ft}^3} \right] \times 100\% = 0.5 \text{ wt-\% per day}$$

Note: The 124 scfh is already at the post-LOCA accident pressure of 44 psig (P_A).

The drywell temperature corresponds to the maximum from a small steam line break and bounds all accidents. The primary containment volume of 232,302 ft³ is obtained by adding the minimum drywell and wetwell free volumes. Standard temperature and pressure are used in the AST application and are 68°F and 14.7 psia, respectively.

The total leakage is then 0.8 + 0.5 or 1.3 wt-% per day, and a bounding value of 1.5 wt-% per day was used for NPSH calculation.

NON-PROPRIETARY INFORMATION

RAI SPSB-C-31

The VYNPS Individual Plant Examination (IPE) report dated December 21, 1993 (Reference letter BVY 93-139), Section 3.1.2.1, "Large LOCA Event Tree," Event AI (Alternate Injection), models the failure of long-term core cooling due, in part, to "loss of LP/CS NPSH at high suppression pool temperature if the containment vent opens and the operator fails to control pressure by reclosing the vent." Concerning the accident sequence modeling for large LOCAs, describe all differences between the IPE and the PRA performed to support the EPU application.

Response to RAI SPSB-C-31

The response to this question is provided by dividing the modeling of large loss-of-coolant accident (LLOCA) accident sequences into the following major elements:

- Initiating events
- Thermal hydraulic analysis
- Accident sequence modeling
- Systemic / functional success criteria
- System fault tree modeling
- Component data
- Human reliability analysis
- Quantification process

Each of these elements is discussed below. The following responses focus on the core damage risk measure (i.e., Level 1 probabilistic safety analysis (PSA)).

Initiating Events

Typical of probabilistic risk assessment (PRA) industry techniques, the VYNPS individual plant examination (IPE) risk assessment adopted generic estimates for loss-of-coolant accident (LOCA) initiating event frequencies. The VYNPS IPE adopted the LLOCA initiating event frequency used in the WASH-1400 and NUREG-1150 industry risk assessment studies.

Consistent with recommendations in the November 2000 Industry Peer Review of the VYNPS PSA, the IPE LOCA frequencies have been updated to a more current industry reference. The LLOCA initiating event frequency (2.4E-5/yr) used in the EPU risk assessment is based on NUREG/CR-5750, "Rates of Initiating Events at U.S. Nuclear Power Plants: 1987 – 1995."

Thermal Hydraulic Analyses

Thermal hydraulic analyses are used in both the VYNPS IPE and EPU risk assessments to support determinations of functional/systemic success criteria, accident progression phenomena, accident sequence timings, and radionuclide release characteristics. The Electric Power Research Institute (EPRI) Modular Accident Analysis Program (MAAP) code was used to perform the thermal hydraulic analyses for both the IPE and EPU risk assessments. The MAAP code is used in the majority of the U.S. nuclear power industry PSAs.

NON-PROPRIETARY INFORMATION

Multiple MAAP runs for LLOCA scenarios were performed in support of both the VYNPS IPE and EPU risk assessments.

The MAAP code versions used for the IPE and the EPU risk assessments are different. The most current versions available at the time of each study were used. MAAP Version 3.0B was used for the VYNPS IPE, and MAAP Version 4.0.4 was used in support of the EPU risk assessment. In preparation for use of MAAP Version 4.0.4 and performance of EPU thermal hydraulic calculations, a number of modifications were made to the VYNPS MAAP parameter file and input decks to address the current plant condition and EPU configuration, including:

- core thermal power
- fuel information
- core peaking factor information
- core thermal hydraulic information (e.g., enthalpies, flow rates, etc.)
- setpoints (e.g., SRVs, RPV level, etc.)
- EOP thresholds and curves

In both the VYNPS IPE and EPU MAAP runs, the normally-open 3" line off the torus is modeled as closed. This is reasonable because the valve in the line will receive an isolation signal very quickly for all the accident scenarios modeled. In the case of a LLOCA, the 2.5 psi Hi DW isolation signal would be received at t = 0.

The LLOCA accident progression timings, although close, are different for the IPE and EPU due primarily to core thermal power changes and new fuel peaking factor information. Accident progression timings for the EPU and pre-EPU conditions for a representative LLOCA scenario (without coolant makeup) are summarized in the following Table SPSB-C-31-1:

Table SPSB-C-31-1

Event	LLOCA Accident Progression Times	
	Pre-EPU	EPU
LLOCA initiator	t = 0	t = 0
RPV Depressurized Below ECCS LP Permissive Pressure	t = 0	t = 0
Hi DW Pressure (2.5 psig) Signal	t = 1 sec.	t = 1 sec.
RPV Level @ 1/3 Core Height	t = 3.6 min.	t = 3.5 min.
1800°F Core Temperature (Onset of Core Damage)	t = 11.7 min.	t = 10.0 min.

NON-PROPRIETARY INFORMATION

Accident Sequence Modeling

A LLOCA causes rapid depressurization of the reactor and requires rapid injection of water from high volume systems. Thus, the LLOCA event tree does not credit use of HPCI and RCIC for high pressure injection to the reactor, and use of the ADS valves for reactor depressurization is not required.

The LLOCA break is assumed to be located on the discharge side of a recirculation line, since this will disable one train (two pumps) of LPCI injection. This assumption envelopes breaks of a core spray line, since core spray line breaks would also disable a single train (but only one pump) of injection. Also, the assumed location is conservative for recirculation suction side breaks since the recirculation loop isolation valves are designed to close for these breaks such that no train of low pressure injection would be disabled by the break location.

The LLOCA event tree includes the following Top Events:

- Control Rod Insertion (CR)
- Vapor Suppression (VS)
- LPCI (LP)
- Core Spray (CS)
- Torus Cooling (TC)
- Containment Venting (VT)
- Alternate Injection (AI)

The Alternate Injection (AI) top event models the success of continued core cooling following containment heat removal challenges. Due to the large inventory makeup requirements of a LLOCA and the comparatively rapid progression of such accidents, alternate injection sources (e.g., CRD, DDFP, Condensate Transfer, etc.) are not credited in the LLOCA event tree. The only available injection sources in the LLOCA event tree are the LP ECCS systems taking suction from the suppression pool (i.e., LPCI and CS). As such, the only issue modeled in the AI top event for LLOCA scenarios is loss of adequate NPSH for LPCI and CS following containment failure or venting. For LLOCA scenarios with initial coolant makeup but no containment heat removal (i.e., failure of top events TC and VT), the failure probability of continued LPCI or CS injection post containment failure is modeled as 1.0. For LLOCA scenarios with initial coolant makeup and successful emergency containment vent initiation, the failure probability of continued LPCI or CS injection is defined by the human error probability (AINPSH) for the operators failing to control the containment vent such that the containment pressure is reduced very rapidly and pumps taking suction off the suppression pool are assumed to fail due to low NPSH.

The LLOCA event tree structures (i.e., the accident sequence progression assumptions, the top events, the number of accident sequences, and the accident sequence end states) remain the same in the VYNPS IPE and EPU risk assessments.

Systemic / Functional Success Criteria

The system and functional top events modeled in the VYNPS LLOCA event tree are discussed above. The LLOCA event tree systemic and functional success criteria were re-considered in the VYNPS EPU risk assessment to consider the EPU thermal hydraulic calculations and the

NON-PROPRIETARY INFORMATION

EPU plant changes. No changes to the LLOCA systemic and functional success criteria were necessary for the EPU. The LLOCA systemic and functional success criteria are exactly the same for the VYNPS IPE and EPU risk assessments, as summarized in the following Table SPSB-C-31-2:

Table SPSB-C-31-2

Safety Function	LLOCA Success Criteria	
	IPE	EPU
Reactivity Control	All control rods inserted (RPS mechanical and electrical success)	Same
Primary System Overpressure Control	Not required	Same
Vapor Suppression	10 of 10 WW-DW vacuum breakers do not fail open	Same
High Pressure Injection	Not required	Same
RPV Emergency Depressurization	Not required	Same
Low Pressure Injection	1 LPCI pump or 1 CS pump	Same
Alternate Injection	None	Same
Containment Heat Removal	1 RHR Hx loop or Emergency Containment Vent	Same

System Fault Tree Modeling

The original IPE LPCI fault tree did not model the recirculation loop discharge valves as a "failure mode" of the associated LPCI subsystem. Depending on the postulated break size and location (particularly a large, suction side break) LPCI flow to the intact loop could short-circuit the core by flowing through the RPV lower plenum and out the break if the intact loop discharge valve is not closed. The LPCI fault tree was revised in 1998 IPE Update to include failure of the recirculation loop discharge valves as a failure mode of the associated LPCI subsystem.

At the time of the IPE, the plant operated with the hard piped torus vent MOV TVS-86 in the open position. Subsequently plant procedures were changed to operate with TVS-86 in the normally closed position. The fault tree model for top event VT, opening of the hard piped torus vent, was modified in the 1998 IPE Update to require operator opening of this MOV for successful torus venting.

NON-PROPRIETARY INFORMATION

Component Data

Both the VYNPS IPE and EPU risk assessments employ generic and plant-specific component data.

The generic component failure data used in the VYNPS IPE was based primarily on the generic database provided with the RISKMAN software. Plant-specific component failure analysis was performed for a large number of key components (e.g., EDGs, ECCS pumps and valves, RCIC, SW pumps and valves, CRD pumps, SLC pumps, etc.). The plant-specific failure data distributions were generated via a one-step Bayesian update statistical analysis (a standard PSA industry technique), using the RISKMAN database as the source of generic prior data distributions and updating with VYNPS specific component failure information from January 1973 through 1989 to create the posterior data distributions used in the PSA.

In addition to the component failure data, the VYNPS IPE employed generic and plant-specific equipment maintenance unavailability probabilities. The generic maintenance unavailabilities were based on the generic database provided with the RISKMAN software.

A number of component data updates have been performed since the VYNPS IPE. The plant-specific component data Bayesian updates included in the EPU risk assessment incorporate VYNPS component failure data up through March 2002. The list of components analyzed with plant-specific data is the same for the IPE and the EPU risk assessments. The list of component unavailabilities estimated using VYNPS specific data is larger in the EPU risk assessment than in the IPE (due primarily to information available from the Maintenance Rule database, which did not exist at the time of the VYNPS IPE). The generic database provided with the RISKMAN software is maintained as the generic component data source for the VYNPS EPU risk assessment. A review of the generic database was made against other industry generic sources during the VYNPS 2002 PSA Update, and it was determined that switching generic data reference sources would not provide any significant benefit to the PSA.

The following Table SPSB-C-31-3 is a comparison of the VYNPS IPE and EPU risk assessment values for major components relied upon in LLOCA accident scenarios:

Table SPSB-C-31-3

Component Failure Mode	Failure / Unavailability Probability	
	IPE	EPU
LPCI/RHR Pump FTS	1.64E-3 / demand	1.85E-3 / demand
LPCI/RHR Pump FTR	3.39E-5 / hr	2.80E-5 / hr
LPCI/RHR MOV FTO	1.25E-3 / demand	1.29E-3 / demand
RHR Loop T&M Unavailability	4.91E-3	4.54E-3
CS Pump FTS	1.15E-3 / demand	1.86E-3 / demand
CS Pump FTR	3.38E-5 / hr	3.37E-5 / hr
CS MOV FTO	2.69E-3 / demand	2.90E-3 / demand
CS Loop T&M Unavailability	4.91E-3	2.28E-3

NON-PROPRIETARY INFORMATION

Human Reliability Analysis

The two primary types of human error probability (HEP) events included in the VYNPS PSA are pre-initiator, or latent, errors (e.g., failure to properly restore equipment following test or maintenance) and post-initiator errors (e.g., failure to initiate RPV emergency depressurization).

The pre-initiator HEPs in the VYNPS IPE were calculated using the THERP methodology from NUREG/CR-1278, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications". The post-initiator HEP calculations for the VYNPS IPE were performed using either of two methods: the "EPRI method" (EPRI NP 6560L, "A Human Reliability Analysis Approach Using Measurements for Individual Plant Examination"), or the Time Reliability Correlation (TRC) method based on NUREG/CR-1278. The primary method used was EPRI NP 6560L.

Since the IPE, the VYNPS HRA has been updated to address plant hardware, training and procedural changes. The VYNPS HRA was updated in May 2000 primarily due to the conversion from EPG Rev. 4 based EOPs to EPG/SAG based EOPs. Because the EPG Rev. 4 based EOPs in use at VYNPS during the IPE are quite similar to the EPG/SAG based EOPs, a large number of the VYNPS HEPs from the IPE were unchanged by the May 2000 HRA update. No changes to pre-initiator HEPs were necessary in the 2002 HRA update.

In addition to the May 2000 HRA update, the post-initiator HEPs for the VYNPS EPU risk assessment were re-assessed due to the shorter available time frames (in some cases) caused by the increase in reactor power level (and thus decay heat). The post-initiator HEPs were re-calculated using the same methodologies used in the IPE and 2002 HRA update. No changes to pre-initiator HEPs were necessary for the EPU risk assessment.

The following Table SPSB-C-31-4 is a comparison of the IPE and EPU risk assessment HEPs for key actions in LLOCA accident scenarios (note that no credit is taken in the VYNPS IPE or EPU risk assessments for alternate injection sources, recovery of the main condenser, or for manual initiation of ECCS during a LLOCA):

Table SPSB-C-31-4

Event ID	Description	Human Error Probability (HEP)		
		IPE	2002 HRA Update	EPU
KOPACTFL	Operator Fails to Initiate Torus Cooling	1.0E-6	1.0E-6	1.0E-6
AINPSH	Operator Fails to Control Vent After Initiation	1.0E-2	1.1E-3	1.1E-3

The reduction in the HEP for the AINPSH action from the IPE to the 2002 HRA Update is due to removal of IPE conservative assumptions. The AINPSH HEP calculation in the IPE used conservative assumptions in certain elements, such as available time frames, in order to calculate a venting HEP applicable to many scenarios (including combustible gas control

NON-PROPRIETARY INFORMATION

venting). The 2002 HRA Update separated pressure control venting and combustible gas control venting in the assessment of this action.

Due to the very long time frames involved, changes in allowable action timings due to the EPU do not change the calculated HEPs for these actions.

Quantification Process

The VYNPS IPE and EPU risk assessments are developed and quantified in the same manner and software environment. The VYNPS PSA uses the "large event tree - small fault tree" PSA approach. The VYNPS PSA is developed and quantified using the RISKMAN PSA software. The event trees are systemic models. The fault trees model component level failures that support quantification of the event tree nodes. Typical of the industry, a 24-hour base core damage mission time is used.

The VYNPS IPE event tree quantifications were performed at a truncation level of 1E-13. The event tree quantification truncation levels used in the EPU risk assessment are not uniformly 1E-13, but they are sufficiently low and appropriate (ranging from 1E-12 to 1E-15).