10CFR2.201

SCE&G

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Ollie S. Bradham
Vice President
Nuclear Operations

September 16, 1988

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Virgil C. Summer Nuclear Station
Docket No. 50/395
Operating License No. NPF-12
Response to Notice of Violation
NRC Inspection Report 88-13

Gentlemen:

Enclosed is the South Carolina Electric & Gas Company (SCE&G) response to the Notice Of Violation (NOV) dated August 17, 1988 (EA-88-151). Attachments 1 and 2 to this letter are the SCE&G "Response to the Notice of Violation" (see 10 CFR § 2.201) and "Answer to the Notice of Violation" (see 10 CFR § 2.205), respectively.

As indicated during the Enforcement Conference, SCE&G denies that a violation of plant technical specifications occurred. In sum, SCE&G believes that (1) post-trip recovery reviews were adequate, (2) degraded Service Water System flow was discovered by the Licensee within a reasonable time period, (3) SCE&G was conservative in declaring both Reactor Building Cooling Units inoperable after degraded flow conditions were discovered, (4) Reactor Building Cooling Units were at all times capable of performing their intended safety function, and (5) the post trip review program was effective in that it was via this process that the Licensee identified the degraded flow condition. If the Staff ultimately determines that a violation did occur, the SCE&G position is that the event that gave rise to the NOV has been incorrectly categorized by the NRC as a "cause for significant concern" and should not be categorized greater than a Severity Level IV enforcement action.

SCE&G would like to emphasize that, even though it believes no Technical Specification Limiting Conditions for Operations were violated, and there was no significant impact on the ability to protect the health and safety of the public as a result of this incident, it recognizes and appreciates the potential seriousness of incidents of this nature. SCE&G believes that its record in identifying and correcting potential problem areas in its operations is an excellent one, and will make every effort to assure that it continues in that manner. SCE&G believes that the corrective actions taken in regard to this incident demonstrate the strength of its concern with safe operation of the Virgil C. Summer Nuclear Station.

IE01
1/1

If you should have any questions, please advise.

Very truly yours,

O. S. Bradham

O. S. Bradham

HID/OSB:lcd
Attachments

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ATTACHMENT 1

RESPONSE TO NOTICE OF VIOLATION
VIOLATION NUMBER 50-395/88-13-01

I. INTRODUCTION

As discussed further below in the 10 CFR §§ 2.201 and 2.205 responses, SCE&G believes that it should not have necessarily known and was not required by procedure to verify that flow to at least one train of the Reactor Building Cooling Units (RBCU's) had degraded to a point that Action Statement 3.6.2.3 was invoked until the completion of its post-trip recovery analysis. In addition, post-trip review procedures allow plant startup pending an in-depth review of system and component trends following the trip.

In the alternative, SCE&G maintains that Service Water Booster Pumps (SWBP's) degraded flow to the RBCU's was not sufficiently safety significant to warrant escalated enforcement action.

II. DISCUSSION

The Notice of Violation states:

"During the NRC inspection conducted on May 1 - 31, 1988, a violation of NRC requirements was identified. In accordance with the 'General Statement of Policy and Procedure for NRC Enforcement Actions,' 10 CFR Part 2, Appendix C (1988), the violation is listed below:

"Technical Specification (TS) 3.6.2.3 requires two independent groups of RBCU's be operable in Modes 1, 2, 3, and 4. ACTION Statement 'b' of TS 3.6.2.3 requires that with both trains of RBCU's inoperable and both trains of reactor building spray system operable, to restore at least one train within 72 hours or be in at least hot standby within the next 6 hours and in cold shutdown within the next 30 hours.

"TS 3.0.4 specifies entry into an operational mode shall not be made unless the conditions of the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION requirements.

"TS 6.8.1 requires procedures to be established and implemented covering the activities referenced in Appendix 'A' of Regulatory Guide 1.33, Revision 2, February 1978. Appendix 'A' of Regulatory 1.33, Revision 2, specifies that administrative procedures be established and implemented. Station Administrative Procedure 132 requires that the shift engineer review the computer post trip review printout (that includes service water flow) prior to plant restart. It specifically states to ascertain the cause of each alarm and determine that any required automatic action functioned properly.

"Contrary to the above, entry into Mode 2 was made at 8:24 p.m. on May 12, 1988, and subsequently into Mode 1 at 12:21 a.m. on May 13, 1988, with both trains of RBCU's inoperable. The post trip review failed to detect that RBCU's were inoperable due to low service water flow, and therefore, the plant was in TS 3.6.2.3 Action Statement 'b.'

"This is a Severity Level III violation (Supplement 1)."

ADMISSION OR DENIAL OF THE ALLEGED VIOLATION

SCE&G does not agree that it should have necessarily known prior to entering Mode 2 on May 12, 1988, and subsequently into Mode 1 on May 13, 1988, that the flow to the RBCU's had been degraded to a level that required the implementation of Action Statement 3.6.2.3. In addition, after determining that a degraded flow condition existed, SCE&G conservatively declared both trains inoperable when, in fact, it was likely that only one of two trains actually had degraded flow below technical specification requirements.

SCE&G also believes that post-trip reviews were adequate and were performed according to procedure. In fact, post-trip reviews resulted in the discovery of the service water low-flow condition.

REASON FOR THE VIOLATION

Not Applicable

CORRECTIVE STEPS TAKEN AND RESULTS ACHIEVED

Notwithstanding the SCE&G denial of the above violations, certain actions have been taken to improve the ability of plant personnel to quickly perform comprehensive post-trip analyses:

- The format for displaying data for the post-trip review has been upgraded, making potential system problems more readily apparent (graphical comparison of specific plant parameters plotted against their expected or alarm values).

- Chemical treatment/flushing of the RBCU's, and inspecting/cleaning of clams out of the intake structure has resulted in restoration of flow (greater than the minimally required flow rate) from the SWBP's to the RBCU's.

- Modifications to the RBCU's to allow more frequent on-line flushing/backflushing capabilities are currently being considered.

- A modification that reduces minimum flow requirements by providing flow isolation of two of the four RBCU's is currently in progress. Only one RBCU is required to meet design basis cooling for the system.

- Additional emphasis will be placed on operator response to annunciators and the importance of following all applicable procedures.

ATTACHMENT 2

ANSWER TO NOTICE OF VIOLATION
ENFORCEMENT ACTION 88-151

A. SUMMARY OF POSITION

As noted in Attachment 1 above, SCE&G denies the subject violation. In the alternative, should the Staff maintain its position regarding the occurrence of a violation, as-found conditions were not sufficiently safety significant to warrant escalated enforcement action.

As discussed further below, the degraded flow conditions did not result in the RBCU's being unable to perform their safety function.

B. DISCUSSION

1. Post-Accident Recovery Process

As previously discussed at the June 24, 1988 enforcement conference, the post-trip recovery process is a two-tier system. The first tier involves analysis of the following issues: (1) the cause of the trip; (2) whether the cause of the trip still exists; (3) whether the event requires the implementation of the Emergency Plan; (4) whether any Limiting Safety System Setting has been exceeded; and (5) whether any Safety Limit has been exceeded.

As correctly referenced in the NOV, the Shift Engineer shall also review the Plant Process Computer Sequence of Events Printout to (1) ascertain the cause of each item on the printout, (2) verify that Reactor Trip Breakers opened as required, (3) verify that a Manual Reactor Trip was initiated, (4) verify a Turbine Trip as required, (5) verify Main Steam Line Isolation as required, (7) verify other safety equipment start as required, and (8) verify Emergency Feedwater start as required. Additional actions by the Shift Supervisor, Control Room Supervisor and Reactor Operator are required prior to returning the plant to power. (See Station Administrative Procedure 132 § 6.4.3.) Startup of the Service Water Booster Pumps (SWBP) and verification of initial flow through the RBCU's (4000 gpm) was verified by the Shift Engineer. The Shift Engineer should not have necessarily known and was not required by procedure to verify whether the RBCU flow had decreased subsequent to initial SWBP's startup and establishment of minimum flow. In addition, the Shift Engineer could have accepted the downward flow trend as a result of Operator action being taken to secure the SWBP's. Flow rates to the RBCU's could remain at ≈2000 gpm due to the operating Service Water Pumps.

Subsequent to a restart determination, the Reactor Trip Package (RTP) was forwarded to the Independent Safety Engineering Group (ISEG) for the second tier review/analysis. It was this review that discovered the degraded flow condition.

SCE&G personnel satisfactorily completed all steps of the procedure and properly authorized the restart of the unit. As discussed during the Enforcement Conference, the reason the initial post-trip review did not discover that a SWBP low-flow alarm had actuated was due to the fact that limited computer output capacity precluded its inclusion in the RTP.

While SCE&G acknowledges a deficiency, the circumstances did not result in a violation of existing procedures or Technical Specifications.

2. Discovery of the SWBP Low Flow Condition

10 CFR Part 2, Appendix C, Section V.A states that, "Licensees are not ordinarily cited for violations resulting from matters not within their control, such as equipment failures that were not avoidable by reasonable licensee quality assurance measures or management controls." As previously stated, SCE&G believes that the cited degraded flow condition was not necessarily apparent and was not required by procedure to be verified during the initial post-trip review. The initial reviewer reviewed the plant process alarm printout which did not indicate a low flow alarm. Subsequent component/system performance was appropriately analyzed during the RTP review by ISEG after plant startup had been authorized.

3. Safety Significance of As-found Conditions

Analysis by SCE&G concluded that flow in RBCU Train A had reduced to approximately 3900 gpm and Train B had reduced to approximately 2000 gpm some time after the SWBP's had initially started after the trip. Conservatively, SCE&G declared both trains inoperable and entered the appropriate Technical Specification Action Statement 3.6.2.3. The Train A indicated flow was likely at or above the required flow when considering instrument error (± 400 gpm).

Even if the lowest flow rate is assumed, analysis concludes that flow was sufficient to meet all design basis conditions. Therefore, there is no safety significance regarding the degraded flow condition (analysis previously provided; copy attached for reference).

4. Severity Level of the Violation

Should the Staff ultimately determine that a violation occurred as stated, it should be categorized at no greater than a Severity Level IV enforcement action. 10 CFR Part 2, Appendix C, Supplement 1, § D.1 states that a Severity Level IV violation involves "A less significant violation of a Technical Specification Limiting Condition for Operation where the appropriate Action Statement was not satisfied within the time allotted by the Action Statement."

SCE&G believes that the facts discussed above clearly indicate that the degraded flow condition did not result in a safety significant issue in that all affected systems could have performed their design basis functions. Therefore this should not be categorized as higher than a Severity Level IV enforcement action.

SOUTH CAROLINA ELECTRIC & GAS COMPANY

Inter-Office Correspondence

ENGINEERING SERVICES
(Office)

CGSS: 21641
File: 4.534
14.3700

SUBJECT V. C. Summer Nuclear Station
Reactor Building Cooling Units
Consequence Analysis

Date July 19, 1988

Reference CGGS-37485 dated July 19, 1988

To A. R. Koon

Attention of

A "Consequence Analysis" to evaluate the impact on plant performance of reduced RBCU heat removal capability due to the combination of reduced RBCU fan flow of 54,200 ACFM and reduced service water flow of 2,200 gpm for the 'B train' RBCU's has been performed.

The "Consequence Analysis" evaluations performed have demonstrated that the degraded RBCU performance combined with postulated accidents does not result in exceeding any regulatory guidelines or loss of any equipment required for safe shutdown.

If additional discussion is necessary, please contact me at extension 4703.

B. T. Estes, Jr.

B. T. Estes, Jr.
Senior Mechanical Engineer
Design Basis Engineering

R B Clary

R. B. Clary, Manager
Design Engineering

cc: B. T. Estes
G. V. Meyer
NPCF
File/R. B. Clary

Gilbert/Commonwealth engineers and consultants

GILBERT/COMMONWEALTH, INC., P.O. Box 1498, Reading, PA 19603 / Tel 215-775-2600 / Cable Gilasoc / Telex 836-431

July 19, 1988

Mr. R. B. Clary, Manager
Design Engineering
South Carolina Electric & Gas Company
P.O. Box 88
Jenkinsville, SC 29065

Attention: Mr. B. T. Estes

CGGS-37485

Re: V. C. Summer Nuclear Station
G/C W. O. 04-5650-500
Reactor Building Cooling Units,
Consequence Analysis
File Code: 1.1.6-500/4.14
Response Code: NRR

Dear Mr. Clary:

Per your request, a "Consequence Analysis" to evaluate the impact on plant performance of reduced RBCU heat removal capability due to the combination of reduced RBCU fan flow of 54,200 ACFM and reduced service water flow of 2,200 gpm for the 'B train' RBCU's has been performed. The tasks identified for evaluation and their final status is as follows:

	TASK	FINAL STATUS
1.	Establish degraded RBCU performance	New values provided by American Air Filter
2.	Evaluate SW system pressure and the impact on RBCU performance	Verified calculation yielding minimum SW pressure at RBCU's
3.	MSLB Pressure/Temperature analysis inside Reactor Building	Verified calculation yielding pressure/temperature above licensing values but within regulatory guidelines
4.	LOCA Pressure/Temperature analysis	Verified calculation yielding pressure/temperature above licensing values but within regulatory guidelines
5.	Equipment Qualification (EQ) Evaluation	Verified calculation, all equipment qualified for Tasks 3 and 4 pressure/temperature conditions
6.	Offsite/Control Room Doses	No change from previously evaluated doses
7.	Instrument Loop Accuracies	Verified calculation, all RB 1E instrument loop accuracies analyzed for pressure/temperature conditions equal or greater than those evaluated in tasks 3 and 4

Each of these tasks is addressed in more detail within Attachment 1 of this letter.

Very truly yours,

R E Anderson

R. E. Anderson
Applied Engineering Analysis Task Engineer

K E Nodland

K. E. Nodland
Engineering Project Manager

REA/KEN:tln
Attachment

cc: NPCF	w/att	B. T. Estes	w/att
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G. Meyer	w/att	P. L. Bunker	w/att
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		D. J. Lengel	w/att
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		R. E. Anderson	w/att
		S. M. Cisek	w/att

V.C. Summer Nuclear Station

RBCU Consequence Analysis

TABLE OF CONTENTS

INTRODUCTION

A "Consequence Analysis" evaluating the impact of degraded RBCU performance on the post accident response of the V.C. Summer Nuclear Power Plant has been completed. This evaluation identifies and quantifies the post accident impact on safety concerns due to degraded RBCU performance resulting from the combination of reduced RBCU fan flow (fan capacity at lower Tech. Spec. limit) and reduced service water flow to the 'B' train RBCU's of 2,200 gpm, which corresponds to the minimum documented RBCU flow rate which would have been present during plant restart.

The major assumptions used to perform this evaluation are as follows:

1. Loss of Offsite Power
2. 'A' Train Diesel Generator failure to start
3. One 'B' Train RBCU fan is in service
4. 'B' Train RBCU fan flow is 54,200 ACFM
5. 'B' Train Service Water flow of 2,200 gpm, 1,100 gpm to each RBCU
6. Service Water temperature of 66.7°F (maximum SW temperature during degraded flow conditions)
7. No Reactor Building Cooling Unit identified leakage.

Section 1: Establish Degraded RBCU Performance

American Air Filter (AAF), the RBCU vendor, was asked to evaluate the heat removal capabilities of the V.C. Summer RBCU's for the conditions given in Table 1. Table 2 provides the calculated heat removal rates developed by AAF using the methodology described in their approved Topical Report No. TR7101A.

Section 2: Service Water System Pressure Analysis

The reduced SW flow through the Reactor Building Cooling Units will affect the pressure in the SW System. A new RBCU outlet SW pressure was calculated using the reduced 2,200 gpm flow (1,100 gpm/RBCU) and a normal low pond level of 420.5 ft. The pressure at the RBCU was calculated working back from the discharge and assuming that the increased system pressure drop contributing to the flow reduction is entirely upstream of the RBCU outlet. This assumption will result in a bounding worst case minimum SW pressure value being calculated at the RBCU outlet. The calculated pressure is 5.95 psia at the RBCU coil outlet.

The 5.95 psia SW pressure at the RBCU outlet results in a saturation temperature of 169°F. When the temperature of the SW in the RBCU coil reaches 169°F, some steam formation will occur due to heat transfer and will result in decreasing the heat transfer coefficient for the coil downstream of the point where the saturation temperature is reached. Performance data for the RBCU coils at 1,100 gpm indicates that a 200°F Reactor Building post accident temperature will result in a SW outlet temperature from the RBCU of 163°F. Since this is less than the 169°F saturation temperature, no steam formation in the RBCU's will occur with Reactor Building temperatures of 200°F or less.

Section 3: MSLB Pressure/Temperature Analysis Inside Reactor Building

The Licensing Basis Main Steam Line Breaks (MSLB) were reanalyzed using the RBCU heat removal capacity identified in Table 3. Table 3 provides a comparison of the degraded RBCU performance versus the RBCU performance used for the Licensing Basis Accidents. The degraded RBCU performance is based on the data provided by AAF for conditions 3 and 4 (see Tables 1 and 2). Energy removal by the RBCU was conservatively set to zero when the Reactor Building (RB) temperature exceeds 200°F. This approach very conservatively bounds RBCU performance when Service Water flashes within the RBCU due to low Service Water pressure as discussed in Section 2.

A comparison of Licensing Basis and degraded RBCU CONTEMPT LT-26 pressure/temperature results for the peak pressure MSLB (1.4 ft^2 DER at 102% power) are provided in Table 4. The maximum calculated degraded RBCU analysis pressure, including an initial RB pressure of 1.5 psia to account for maximum allowable normal operation Technical Specification, is 51.23 psig. This peak calculated pressure of 51.23 psig is well under the Reactor Building design pressure of 57.0 psig which Standard Review Plan (SRP) 6.2.1.1.A, Section II.a specifies for licensing of operating plants, and is also less than the peak calculated value of 51.8 psig which corresponds to the 10% safety margin specified in SRP 6.2.1.1.A, Section II.a for plants in the Construction Permit stage.

A reanalysis of MSLB peak temperature (0.645 Split Rupture at 102% power) was not performed since this calculated temperature is due to superheating of the RB post accident atmosphere which is quenched by RB sprays prior to RBCU initiation.

Conservative assumptions incorporated into this analysis include:

1. Multiple failures (Diesel Generator Failure, Main Steam Isolation Valve Failure, Emergency Feedwater Control Valve Failure) used for determining mass/energy release (Westinghouse BIT Removal Analysis Assumptions).

2. Use of maximum Technical Specification allowable normal operation pressure of 1.5 psig.

3. No heat removal by RBCU at RB temperatures above 200°F.

Section 4: LOCA Pressure/Temperature Analysis

The degraded RBCU performance was also analyzed for the LOCA event (the long term governing pressure/temperature event). The Double Ended Pump Suction LOCA Contempt LT-22 model was updated for use of Contempt LT-26. Model changes incorporated include:

			Licensing Analysis	RBCU Consequence Analysis
1.	Environmental conditions	Temperature	90°F	95°F
	conservatively changed to:	Humidity	50%	70%

2. Heat transfer coefficients used with passive heat sink models are the Tagami heat transfer coefficient used through blowdown (approximately 17.2 seconds post accident) and the Uchida heat transfer coefficient thereafter.

3. Spray initiation and service water flow to RBCU timing is conservatively set to the same times as for the MSLB.

4. Convection and radiation heat transfer is allowed to the environment from the outer face of the RB concrete shell and dome.

5. The RBCU performance given in Table 3 was used.

The resultant RB pressure/temperature profile is tabulated in Table #5. These results show calculated peak pressure/temperatures slightly higher (60.0 psia vs. 59.36 psia, 267.9°F vs. 266.7°F) than the Licensing Basis LOCA values. Calculated RB LOCA pressure remains more than 10% below design pressure. Calculated RB LOCA temperature remains below design of 283°F.

Conservative assumptions incorporated into the analysis include:

1. Use of maximum Technical Specification allowable normal operation RB pressure of 1.5 psig.

2. RB sprays are assumed to automatically go into the recirculation mode for 24 hours

3. No heat removal by RBCU at RB temperatures above 200°F (the RBCU's begin to remove energy from the RB atmosphere at approximately 18 hours post accident).

Additionally, a LOCA evaluation assuming RB Spray operation for only 2 hours per FSAR Section 6.2.2.2.1.2 was performed. Results of this analysis are also tabulated in Table 5.

A comparison of these two cases shows that sprays running for 24 hours yields the highest Reactor Building pressures and vapor temperatures; whereas, the 2 hour spray case yields the highest Reactor Building sump temperatures.

Section 5: Equipment Qualification Evaluation

An evaluation was performed to determine the effects of increased LOCA and MSLB temperatures and pressures resulting from RBCU reduced flow conditions on Class 1E equipment which would have been exposed to the postulated accident environment.

The evaluation compared the newly calculated LOCA and MSLB temperature and pressure versus time profiles with the LOCA test profiles to which the Class 1E equipment was subjected during environmental qualification testing. Acceptance criteria was based on whether the tested profile enveloped the postulated accident profiles.

Any potential deviations or postulated temperature excursions which exceeded those of the test profile were documented and evaluated. It was determined that the affected Class 1E equipment would have been qualified if they had been exposed to the postulated LOCA/MSLB environment occurring during RBCU low flow conditions.

Section 6: Offsite/Control Room Doses

The reduced air flow (54,200 ACFM) through the RBCU coils combined with reduced service water flow (1,100 gpm @ 66.7°F) to the RBCU coils does not impact the offsite or control room doses (dose assessment based on 54,200 ACFM RBCU fan flow previously reported in letters CGGS-37423 and CGGS-37450, dated June 24, 1988 and June 30, 1988 respectively). These calculated doses are conservatively based on design containment pressure for the first 24 hours post accident and 1/2 that value thereafter. Thus, the relatively small changes in RB pressure/temperature response resulting from reduced RBCU performance will not result in calculated offsite and control room doses above the current Licensing Basis.

Section 7: Instrument Loop Accuracies

An analysis was made of the impact on 1E instrument loop accuracies from increased MSLB and LOCA pressure/temperature resulting from the RBCU Consequence Analysis. Calculations for these loops included insulation resistance (IR) degradation effects of cabling from accident conditions as well as component errors.

There are no degraded protective function actuations as a result of the new RBCU temperature/pressure profile since all protective actuations occur within the first 5 minutes of the initiating event. During this period, the accident profiles are essentially identical to those previously evaluated.

Display accuracies are generally unaffected by the new temperature/pressure profile since the existing calculations utilized higher bounding temperatures than results from the new RBCU analysis. Reactor Building Level and Steam Generator wide range level indications are the only Post Accident Display channels which were not completely bounded, however, no significant affects (less than 0.1%) were created and no margins were reduced beyond allowable values.

Summary:

The "Consequence Analysis" evaluations performed have demonstrated that the degraded RBCU performance combined with postulated accidents does not result in exceeding any regulatory guidelines or loss of any equipment required for safe shutdown.

TABLE 1

RBCU Performance Capability - Degraded Conditions

REACTOR BUILDING CONDITION *	TEMPERATURE ($^{\circ}$F)	TOTAL PRESSURE (PSIA)
1a	283	59.4
1b	241	59.4
1c	200	59.4
1d	160	59.4
2	241	44.
3	200	30.
4	160	22.

* All evaluations based on:

54,200 ACFM, Fan Flow at Inlet

1,100 gpm, Service Water flow to 'B' train in-service RBCU

66.7°F, Service water temperature

100%, Reactor Building Humidity

0.0005, Cooling Coil Fouling Fact[illegible]

NOTE: Reactor Building design pressure is 57.0 psig, 71.7 psia

TABLE 2

RBCU Performance Capability

1.	ACFM at fan inlet	54203	54201	54200	54202	54200	54200	54200
2.	Heat Removal Capacity in BTU/hr (x106)	97.79	71.75	48.29	28.40	74.21	51.69	30.40
3.	Cooling Water GPM	1100	1100	1100	1100	1100	1100	1100
	Entering Temp °F	66.7	66.7	66.7	66.7	66.7	66.7	66.7
	Leaving Temp °F	254.6	201.8	156.2	118.9	206.7	162.8	122.6
4.	Coil Entering Air							
	ACFM	67412	63455	60557	58370	66820	66456	64028
	°F	283	241	200	160	241	200	160
5.	Coil Leaving Air							
	ACFM	54203	54201	54200	54202	54200	54200	54200
	°F	280.6	228.8	180.1	137.3	231.5	182.5	133.9
	Density	.1483	.2031	.2390	.2643	.1405	.1136	.0959
6.	Moisture Condensation rate in lb. water/hr	104,985	70,829	42,396	20,424	75,119	49,435	26,670

TABLE 3

RBCU Performance Capability - Degraded Conditions
Heat Removal Capability

REACTOR BUILDING TEMPERATURE (°F)	LICENSING BASIS MSLB (BTU/HR)	LICENSING BASIS LOCA (BTU/HR)	RBCU CONSEQUENCE ANALYSIS (BTU/HR)
283	125×10^6	100×10^6	0.
241	90×10^6	75.7×10^6	0.
200	57×10^6	51.8×10^6	51.7×10^6
160	29×10^6	28.5×10^6	30.4×10^6

TABLE 4

Case Comparison - MSLB

TIME (SEC.)	PRESSURE* (PSIA)		TEMPERATURE (°F)	
	Licensing	RBCU**	Licensing	RBCU**
1	18.84	18.84	186.0	186.0
2	22.13	22.13	222.8	222.8
5	28.00	28.00	231.3	231.3
10	35.31	35.31	224.1	224.1
15	40.65	40.65	236.9	236.9
20	41.87	41.87	239.7	239.7
40	44.91	44.91	246.4	246.4
60	46.55	46.55	249.5	249.5
100	49.30	49.45	254.5	254.8
140	52.24	52.60	259.4	260.0
200	55.32	57.56	263.9	267.7
280	59.16	60.26	269.4	271.0
400	58.09	59.84	267.8	270.3
500	57.86	60.11	267.4	270.7
600	57.70	60.55	267.1	271.2
700	57.80	61.11	267.3	271.9
800	57.96	61.74	267.5	272.4
900	58.18	62.40	267.8	273.7
1000	58.43	63.10	268.2	274.6
1100	58.67	63.78	268.5	275.5
1200	58.87	64.43	268.8	276.4
1500	-	57.04	-	266.1
1800	-	50.85	-	256.4

* 1.5 psia should be added to this value to cover the max. allowable Tech. Spec. Normal Operation Pressure

** RBCU Consequence Analysis

NOTE: Reactor Building design pressure is 57.0 psig, 71.7 psia

TABLE 5
Case Comparison - LOCA

Time (Sec.)	Heat Transfer Coeff. to Passive Sinks ($BTU/Hr - ft^2 - {}^\circ F$) Licensing	Heat Transfer Coeff. to Passive Sinks ($BTU/Hr - ft^2 - {}^\circ F$) RBCU*	Pressure (PSIA) Licensing	Pressure (PSIA) RBCU*	Vapor/Sump Temperature (°F) (RB Sprays end at 24 hours) Licensing	Vapor/Sump Temperature (°F) (RB Sprays end at 24 hours) RBCU*	Vapor/Sump Temperature (°F) (RB Sprays end at 2 hours) RBCU*
5	139.1	135.4	40.85	40.66	233.2/192	232.9/191	232.9/191
10	195.7	194.2	51.73	51.68	254.7/211	254.7/211	254.7/211
15	239.3	237.9	56.20	56.29	262.0/218	262.2/219	262.2/219
20	228.2	94.5	54.59	55.45	259.4/220	261.0/220	261.0/220
40	118.5	93.9	53.21	55.14	257.2/223	260.4/222	260.4/222
60	80.6	94.1	53.83	55.23	258.2/224	260.5/224	260.5/224
100	63.2	94.5	55.17	55.48	260.3/228	260.9/228	260.9/228
200	63.4	97.1	57.41	57.27	263.8/233	263.8/234	263.8/234
300	65.9	101.7	59.11	59.42	266.3/241	266.9/241	266.9/241
350	65.8	103.1	59.36	60.00	266.7/244	267.9/245	267.9/245
550	63.4	101.6	57.60	59.38	264.0/254	266.8/255	266.8/255
950	58.9	98.1	54.65	58.00	259.3/263	264.8/265	264.8/265
1880/2000	41.9	78.5	43.38	47.04	238.3/261	246.3/263	246.3/263
3880/3500	22.5	45.3	30.40	34.19	202.0/2[illegible]9	218.9/246	218.9/246
5020/5000	21.4	52.3	29.72	36.66	199.4/236	224.4/244	224.4/244
10000	17.3	51.5	26.76	36.28	187.0/223	221.9/237	213.1/242
20000	13.0	44.3	23.74	33.55	171.4/198	214.2/221	156.4/231
40000	10.5	36.8	21.89	30.23	159.4/183	203.1/209	122.3/217
54000	-	-	21.51	29.39	156.6/178	200.0/204	118.5/211
60000	9.7	-	21.28	23.44	154.9/175	172.4/193	116.8/206
86000	-	-	20.20	20.01	146.0/160	148.3/163	114.5/1[illegible]4
90000	6.4	-	18.85	18.33	133.8/164	132.9/166	108.8/177
1.1+5	-	-	18.82	18.89	132.8/170	150.7/172	118.5/174
1.4+5	6.4	-	18.81	18.80	132.6/168	147.8/169	118.5/169
1.9+5	6.4	-	18.81	18.58	132.4/161	140.8/162	115.6/162
2.0+5	-	-	18.81	18.54	132.4/160	139.5/161	115.1/160
5.0+5	-	-	18.60	-	129.9/145	-/-	-/-
1.0+6	-	-	18.28	-	123.4/134	-/-	-/-

* - RBCU Consequence Analysis

NOTE: Reactor Building design pressure is 57.0 psig, 71.7 psia