UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-4005

May 2, 2003

EA-03-077

Paul D. Hinnenkamp
Vice President - Operations
River Bend Station
Entergy Operations, Inc.
P.O. Box 220
St. Francisville, Louisiana 70775

SUBJECT: RIVER BEND STATION - NRC SPECIAL INSPECTION REPORT 50-458/02-07 PRELIMINARY WHITE FINDING

Dear Mr. Hinnenkamp:

On February 7, 2003, the NRC issued the subject special inspection report, which discussed a finding concerning the failure to properly lock open Condensate Prefilter Vessel Bypass Flow Control Valve CNM-FCV200. This apparent violation resulted in a loss of feedwater flow to the reactor pressure vessel following a reactor scram on September 18, 2002.

This finding was assessed based on the best available information, including influential assumptions, using the Significance Determination Process described in NRC Inspection Manual Chapter 0609 and was preliminarily determined to be a finding with some increased importance to safety (White), which may require additional NRC inspection. The finding has a preliminary assessment of low to moderate safety significance because the combination of risk associated with a loss of feedwater and from external events, such as a fire in conjunction with a loss of the feedwater system, over a period of approximately 126 days. This caused the probability of core damage to increase by approximately 1.6 X 10^{-6}/yr. The most significant transient sequences contributing to the increase in risk were those involving the failure of the main feedwater system following a reactor trip combined with the random failures of the high pressure core spray, the reactor core isolation cooling, and the automatic depressurization systems. The NRC Significance Determination Process Phase 3 risk evaluation is enclosed. Our preliminary determination was discussed with members of your staff during a telephone call on April 11, 2003.

This finding was also an apparent violation of Technical Specification 5.4.1 in that Valve CNM-FCV200 was not properly locked open in accordance with required procedures and is being considered for escalated enforcement action in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600. The current Enforcement Policy is included on the NRC's Web site at http://www.nrc.gov/what-we-do/regulatory/enforcement.html.

Since the NRC has not made a final determination in this matter, no Notice of Violation is being issued for this inspection finding at this time. In addition, please be advised that the number and characterization of apparent violations described in the subject inspection report may change as a result of further NRC review.

Before we make a final decision on this matter, we are providing you an opportunity to: (1) present your perspectives on the facts and assumptions used by the NRC to arrive at the proposed finding and its preliminary significance at a Regulatory Conference, or (2) submit your position on the proposed finding and preliminary significance to the NRC in writing. If you request a Regulatory Conference, it should be held within 30 days of the receipt of this letter and we encourage you to submit supporting documentation at least one week prior to the conference in an effort to make the conference more efficient and effective. If a Regulatory Conference is held, it will be open for public observation. If you decide to submit only a written response, such submittal should be sent to the NRC within 30 days of the receipt of this letter.

Please contact David Graves at 817/860-8141 within 10 business days of the date of receipt of this letter to notify the NRC of your intentions. If we have not heard from you within 10 days, we will continue with our significance determination and enforcement decision and you will be advised by separate correspondence of the results of our deliberations on this matter.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Arthur T. Howell III, Director
Division of Reactor Projects

Docket: 50-458
License: NPF-47

Enclosure:
SDP Phase 3 Evaluation

cc w/enclosure:
Executive Vice President and
Chief Operating Officer
Entergy Operations, Inc.
P.O. Box 31995
Jackson, Mississippi 39286-1995

Vice President
Operations Support
Entergy Operations, Inc.
P.O. Box 31995
Jackson, Mississippi 39286-1995

General Manager
Plant Operations
River Bend Station
Entergy Operations, Inc.
P.O. Box 220
St. Francisville, Louisiana 70775

Director - Nuclear Safety
River Bend Station
Entergy Operations, Inc.
P.O. Box 220
St. Francisville, Louisiana 70775

Wise, Carter, Child & Caraway
P.O. Box 651
Jackson, Mississippi 39205

Mark J. Wetterhahn, Esq.
Winston & Strawn
1401 L Street, N.W.
Washington, DC 20005-3502

Manager - Licensing
River Bend Station
Entergy Operations, Inc.
P.O. Box 220
St. Francisville, Louisiana 70775

The Honorable Richard P. Ieyoub
Attorney General
Department of Justice
State of Louisiana
P.O. Box 94005
Baton Rouge, Louisiana 70804-9005

H. Anne Plettinger
3456 Villa Rose Drive
Baton Rouge, Louisiana 70806

President
West Feliciana Parish Police Jury
P.O. Box 1921
St. Francisville, Louisiana 70775

Michael E. Henry, State Liaison Officer
Department of Environmental Quality
P.O. Box 82135
Baton Rouge, Louisiana 70884-2135

Brian Almon
Public Utility Commission
William B. Travis Building
P.O. Box 13326
1701 North Congress Avenue
Austin, Texas 78701-3326

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ENCLOSURE

Preliminary Significance Determination

A. Brief Description of Issue

In May 2002, a full-flow condensate filter bypass valve was improperly manipulated such that, if a large feedwater transient occurred, such as one that would occur following a reactor scram, the valve could fail closed, and the feedwater and condensate systems would be lost as a source of makeup water to the reactor vessel.

On September 18, 2002, following a reactor scram, the valve failed closed, resulting in the loss of feedwater and condensate systems. Additionally, the reactor water cleanup system, which returns water to the reactor via the feedwater system, continued to inject hot water into the feedwater system and backwards into the condensate system through open recirculation valves. The control rod drive hydraulic pumps, which take suction from the condensate system, began cavitating (which lasted for approximately 70 minutes) as a result of the hot reactor water being introduced into the pump suction. The cavitation of the operating pump was not detected by the control room operators.

Main feedwater is the normal source of water supply to the reactor, and CRD hydraulics provides an additional source of makeup that is utilized when necessary. Both systems have risk significance.

The event is described in significant detail in NRC Inspection Report 50-458/2002-07.

B. Statement of the Performance Deficiency

Performance Deficiency: The licensee installed a plant modification, in a temporary condition, without providing sufficiently detailed operating procedures and/or operator training.

Supporting Information: On May 15, 2002, operators failed to properly lock Condensate Prefilter Vessel Bypass Flow Control Valve CNM-FCV200 in the open position as required by SOP-0007, Revision 21, “Condensate System.” As a result, following a reactor scram, normal system flow oscillations caused the valve to close. The closure of Valve CNM-FCV200 resulted in a loss of all power conversion system flow.

Bypass Valve CNM-FCV200 was installed during the licensee’s Spring maintenance outage as part of a modification to provide full-flow condensate filtration. The segment of line in which the valve was installed carries 100 percent of the condensate flow without redundancy. The modification was left in an incomplete condition. Normal operations of the system were permitted, provided the bypass valve was locked in the open position, and the bifurcating lines to the filters were isolated. This configuration should have left the system operating as if in a premodified condition. The configuration of the valve was not reassessed until after the September 18 reactor scram.

The air operator for Valve CNM-FCV200 was not connected to a controller, nor to an air source. Therefore, the valve had to be mechanically restrained in the open position. Craftsmen left the valve in the open position with the handwheel disengaged; plant procedures had been revised to require that the valve be locked in the open position. However, this valve operator type was new to River Bend, and plant operators were not familiar with the operation of the valve. Operators, therefore, locked the handwheel, but did not engage the manual operator.

In summary, the following actions resulted in condensate and feedwater being unavailable to respond following an accident:

♦ Valve CNM-FCV200 was installed in a section of condensate pipe that handled full system flow without redundancy.

♦ The full-flow filtration system installation was not completed prior to normal plant operation.

♦ The motive force (instrument air) and controller for Valve CNM-FCV200 were not installed.

♦ The design of Valve CNM-FCV200 was new to plant operators and they had not received training on the operation of the valve.

♦ Craftsmen left Valve CNM-FCV200 in the open position with the manual mechanism not engaged. Operators later locked the handwheel in the open position but did not engage the manual mechanism. This left the valve in a condition that packing and actuator piston friction were the only things keeping the valve open.

♦ Normal feedwater flow oscillations, following a reactor trip, resulted in the valve closing and being held closed with system differential pressure. This resulted in a loss of all condensate flow.

♦ Upon loss of all condensate flow, the feedwater pumps tripped on loss of suction.

C. Significance Determination Basis

♦ Reactor Inspection for IE, MS, B cornerstones

1. **Phase 1 screening logic, results, and assumptions**

In accordance with Inspection Manual Chapter 0612, Appendix B, “Issue Disposition Screening,” the inspectors determined that the finding was more than minor because the issue was associated with the configuration control of operating equipment alignment. The failure to properly lock open Valve CNM-FCV200 resulted in the loss of condensate and feedwater flow to the reactor vessel. Therefore, this performance deficiency affected the mitigating systems cornerstone objective “to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences.”

In accordance with Inspection Manual Chapter 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” the inspectors determined that this issue affected high pressure primary injection systems, involved mitigating systems, did not affect fire protection defense-in-depth, and did not affect the safety of a shutdown reactor. Therefore, the mitigating systems column was followed. The following questions were answered:

Is the finding a design qualification deficiency confirmed not to result in loss of function per GL 91-18? No.

Does the finding represent an actual loss of safety function of a system? Yes, the main feedwater system.

Therefore, Phase 1 directed that a Phase 2 estimation be completed.

2. **Phase 2 Risk Estimation**

In accordance with Inspection Manual Chapter 0609, Appendix A, the inspectors conducted a Phase 2 estimation using the Risk-Informed Inspection Notebook for River Bend Station, Unit 1, Revision 1. The dominant affected accident sequences are provided in Table 1. The inspectors assumed that the plant conversion system and control rod drive hydraulics were unavailable to respond to accidents.

The Phase 2 estimation resulted in an estimated White finding. The performance deficiency existed for significantly less than the 365 days assumed in the notebook and licensee analysts stated that the Phase 2 worksheets were overly conservative for analyzing a loss of control rod drive hydraulics. Therefore, the analyst determined that the finding should be evaluated using the Phase 3 process.

3. **Phase 3 Analysis**

<u>Internal Initiating Events</u>:

The following techniques were used in this evaluation.

a. The analyst quantified the internal risk using the INEEL Standardized Plant Analysis Risk Model for River Bend Station, Revision 3i. Rather than attempt to model the loss of short-term availability of the control rod drive hydraulics system, the analyst quantified the model with a loss of condensate and feedwater. The model was then quantified with the control rod drive system also unavailable. The resulting quantification indicated an increase over the baseline core damage frequency between 4.0×10^{-6} and 6.4×10^{-6} over a 126-day exposure.

b. The licensee developed a model for analyzing the internal risk associated with the performance deficiency using a modified version of their Revision 3 PRA model. The licensee revised the model by setting the frequency of a loss of all feedwater initiating event to the frequency of a normal trip. Additionally, the model was changed to indicate a loss of short-term cooling from the control rod drive system, while providing for long-term recovery of the control rod drive system following the cavitation event, provided that high pressure core spray (HPCS) and reactor core isolation cooling (RCIC) had been available.

The analyst reviewed the licensee's modified model in detail during a site visit on January 9, 2003. Several errors in modeling and/or imprecise assumptions were detected and changed during this visit. Based on this review, the analyst concluded that the revised model was adequate for evaluating the performance deficiency and that the assumptions used by the licensee were reasonable. Therefore, the analyst determined that this result was the best estimate of the internal risk associated with the failure to ensure that Valve CNM-FCV200 remained in the open position during power operations.

The resulting increase over the baseline core damage frequency was 7.7×10^{-7} over a 125-day period.

NOTE: The licensee used 125 days in their analysis, while the analyst used 126 days. The actual time from entering Mode 2 following the maintenance outage and the September 18 reactor scram was 126 days, 5 hours. The analyst determined by hand calculation that the difference of 1 day exposure time does not impact the results of the evaluation.

c. The following influential assumptions were made:

- Valve CNM-FCV200, if not fixed in the open position, would have failed shut during any at-power initiating event resulting in, or that should have resulted in, a scram.

- This condition existed prior to plant restart on May 15, 2002, until the reactor trip on September 18, 2002.

- Operators were not familiar with the valve type and had not been trained on the design of the valve. Therefore, it is assumed that any operator could have made a similar error in locking the valve in position.

- The forces holding the valve open (the packing resistance and the actuator friction) would not have changed appreciably in 4 months. Therefore, the exposure time for this condition was 126 days.

- Valve CNM-FCV200 failing closed results in the loss of all condensate and feedwater flow to the reactor pressure vessel.

- No recovery credit should be given for the feedwater or condensate systems. Because the full-flow filtration system modification was incomplete, no indication of the Valve CNM-FCV200 position was available in the main control room. Additionally, plant operators visually inspecting the condensate system for indications of system abnormalities would have observed a valve in the “locked open” position. It should be noted as additional supporting evidence that the operators did not determine the cause of system failure during their response to the event on September 18, 2002.

- The control rod drive hydraulic system would not be functional for the first 6 hours of a transient because, during the event, the pump cavitated for a significant period of time (approximately 70 minutes) while provided high temperature suction water. The 6 hours corresponds to the long-term cooling function of the system, given that another system has provided cooling water to the reactor until that time.

  Reactor water cleanup flowing backwards through the feedwater and condensate systems, combined with extraction steam heating of the isolated systems until the turbine tripped, resulted in excessive control rod drive suction temperatures.

  Prior to the installation of Valve CNM-FCV200, a loss of condensate to the feedwater pump suction would have resulted in the long-cycle recirculation pathway remaining closed. However, because Valve CNM-FCV200 went closed, this bypassed normal system logic and the recirculation path opened when condensate system flow was lost. Therefore, reactor water cleanup system flow would have continued to pump forward to the reactor through the feedwater injection lines.

  There are the following two possible event progressions:

  (1) All flow, with the exception of control rod drive system flow, is lost early in the event.

In this case, reactor vessel level would decrease rapidly and the reactor water cleanup system would automatically isolate on a low level signal.

In this scenario, reactor water cleanup would not cause high temperatures in the control rod drive system suction. However, in this case, control rod drive hydraulics would not provide sufficient flow to the core. These sequences are already accounted for in the licensee's baseline model, and would not affect the change in core damage frequency.

(2) Feedwater provides flow to the vessel for a short period of time, as occurred on September 18, and either the HPCS or RCIC systems are available for a short period of time.

In this case, the reactor water cleanup system would remain in service and respond as observed on September 18. In accordance with the emergency operating procedures, control room operators would "maximize CRD flow." Maximizing flow requires starting a second pump and increasing both pumps' flow rates.

These sequences would result in cavitation of the pumps and a loss of control rod drive system flow in the short term (less than 6 hours). This loss of short-term control rod drive flow results in a change from the baseline risk and was modeled in the licensee's revised PRA.

The following calculations were performed and the results of all model runs are documented in Table 3:

- The analyst used the SPAR model to quantify the internal change in risk for a loss of feedwater; a loss of condensate and feedwater; and a loss of condensate, feedwater, and control rod drive hydraulics. The latter is the closest to actual circumstances, while the other runs provide indication of the sensitivity for lesser system failures.

  The SPAR model provides a basic event designating the probability that all the pumps in a system will fail to run within the same mission time. This basic event represents the common cause failure rate for the pumps. For each of the three systems, the analyst set this common cause failure rate to 1.0 using a logical TRUE house event. The model was then quantified.

- The analyst requested that the licensee use their model of record to analyze the internal risk associated with a loss of all main feedwater and the control rod drive system. Designated as “Licensee’s R3.0" in Table 3, this result provided a validation of the SPAR model and indicated the upper bound for internal risk as quantified by the licensee’s model. Finally, this result shows the sensitivity of the results to the modeling of long-term recovery of the control rod drive system. The dominant accident sequences for this analysis are provided in Table 2.

External Initiating Events:

The analyst used the following methods for determining the change in risk from external events. The change in risk from a loss of feedwater during a fire was determined to be between 8.2×10^{-7} and 2.1×10^{-6} for the 126-day period. The methods used are documented below:

Fire:

a. The analyst used the River Bend IPEEE to estimate the change in risk resulting from external events. Although the feedwater system is not considered a safe shutdown system in the licensee’s IPEEE, feedwater was used to screen out 18 fire areas that had been previously modeled as greater than 1×10^{-6} /yr CDF.

   The licensee reanalyzed each of these fire areas by using the developed fire scenario and quantifying their internal PRA model with fire damaged equipment taken out of service. Without crediting feedwater, each of these scenarios resulted in a core damage frequency of less than 1×10^{-6}/yr CDF. These would have screened out in the IPEEE; however, the numbers indicated that feedwater was important to these scenarios.

   At the analyst’s request, the licensee requantified each of the fire scenarios giving credit for feedwater at its nominal failure rate. The analyst determined that this represented the baseline fire risk for the 18 areas. As documented in Table 4, the inspector subtracted this baseline from the licensee’s analyzed risk without crediting feedwater. The change in risk upon a loss of feedwater for these 18 fire areas was calculated to be 8.16×10^{-7} for the 126-day period.

   2.36×10^{-6}/yr * (126/365)yrs exposure = 8.2×10^{-7}

b. In the second method, the analyst assumed, qualitatively, that feedwater was as important to external events as it is to internal events. The ΔCDF was divided by the licensee’s baseline CDF to quantify the importance. This importance was then multiplied by the licensee’s external events baseline to provide a quantitative result.

1.035×10^{-5} (case) - 8.110×10^{-6} (baseline) = 2.24×10^{-6}

$2.24 \times 10^{-6} \div 8.110 \times 10^{-6}$ = 27.6%

2.25×10^{-5} /yr (external baseline) * .276 * [125/365]yrs

= 2.1×10^{-6} ΔCDF

c. Finally, the analyst noted that this analysis only included 18 fire areas. Many other fire areas in the plant would normally be mitigated using the condensate and feedwater systems. However, the licensee does not have these areas modeled or assessed. Therefore, qualitatively, it can be assumed that the risk calculated in paragraph a. represents the lower bound.

Seismic, Winds, Floods, and Other External Events:

The licensee does not model these events in their PRA or IPEEE. However, the analyst noted that River Bend Station was susceptible to these external events, particularly high winds and flooding. Additionally, being a versatile system, feedwater should be available to provide core cooling in all accident scenarios except secondary line breaks and a loss of offsite power. The most probable external event scenarios would also result in a loss of offsite power. In these events, feedwater would not have been available regardless of the performance deficiency. Therefore, these other external events were determined, qualitatively, to impact the change in risk for this finding, but were not assumed to increase the overall risk by greater than one order of magnitude.

Quantitative Bounds:

To determine the range of possible outcomes, the analyst added the highest result from the internal review and the highest result from the external review. The lower bound of the range was the licensee's revised internal risk result, assuming that a negligible result in external events approaches zero quantitatively. These bounding valves are documented in Table 3.

Sensitivity of Results:

While the results of this evaluation are definitely sensitive to the first four assumptions documented in Section C.3.c, they are based on observed physical conditions during the September 18, 2002 event. Also, they are conservative in that they result in the higher risk significance, and the licensee does not dispute them. Therefore, a sensitivity analysis was not performed on these assumptions.

Case 1:

The licensee's assumption that control rod drive hydraulics would be available for long-term cooling is a best estimate assumption. However, should the operators have "maximized CRD flow" in accordance with emergency operating procedures while the suction temperature was above 300°F, there would have been a high likelihood of permanent pump damage. Therefore, the SPAR and licensee's models were quantified with control rod drive inoperable for short- and long-term cooling. The licensee's model is sensitive to this assumption in that the 1×10^{-6} threshold is crossed if CRD flow is not available.

Case 2:

Several methods were used to qualitatively assess the risk from external events caused by the performance deficiency. The two methods used by the analyst indicated values near that of the internal events model results. However, assuming that the fire risk associated with a loss of feedwater is zero clearly has an impact on the analysis. The analyst believes that the licensee's approach is invalid.

<u>Results and Preliminary Color</u>:

The calculated ΔCDF analysis results, including both internal and external assessments, fall in a range of 7.7×10^{-7} to 8.4×10^{-6}. The analyst's best estimate judgment was 1.6×10^{-6}. This used the licensee's revised model results combined with the qualitative estimate of the external risk using the licensee's IPEEE fire areas and an analysis quantified with the licensee's internal events PRA model. Therefore, the preliminary analysis indicates that the finding is of low to moderate risk significance (White).

Table 1 Phase 2 Dominant Accident Sequences		
Initiating Event	**Sequence**	**Contribution**
Transients (Reactor Trip)	TRANS-PCS-RCIC-CHR-LICRD	9
	TRANS-PCS-RCIC-HPCS-DEP	7
Loss of Reactor Plant Component Cooling Water	TCCP-PCS-RCIC-CHR	9
	TCCP-PCS-RCIC-HPCS-DEP	9
Loss of 120 Vdc Emergency Division I	TDCI-PCS-CHR	6
	TDCI-PCS-HPCS-LPI	8
	TDCI-PCS-HPCS-DEP	8
Loss of 120 Vdc Emergency Division II	TDCII-PCS-CHR-LDEP	8
	TDCII-PCS-RCIC-CHR	7

Table 2
Phase 3 Dominant Accident Sequences

Model	Initiating Event	Sequence	Contribution
SPAR 3i	Transients (Reactor Trip)	HPCS, SRV, OPR-LPI	2.3×10^{-6}
		HPCS, ADS, RCIC	8.8×10^{-7}
	Loss of Vital Medium Voltage Bus	Div II Vital DC	5.8×10^{-7}
	Loss of Offsite Power	RCIC, SSW-CF, NOREC-1H	2.1×10^{-7}
Licensee's Revised	Transients (Loss of Feedwater)	ADS, HPCS, RCIC	3.1×10^{-7}
	Loss of Normal Service Water	SSW, NORECOVERY	2.3×10^{-7}

Table 3
Analysis Results

Model Used	Assumptions	Exposure	ΔCDF
SPAR	Feedwater Inoperable	126	3.9×10^{-6}
	Feedwater/Condensate	126	4.0×10^{-6}
	Feedwater/Condensate/Control Rod Drive	126	6.4×10^{-6}
Licensee's R3.0	Feedwater/Control Rod Drive	125	3.9×10^{-6}
Licensee's Revised*	Feedwater Initiating Event Loss of Short-Term CRD	125	7.7×10^{-7}
External IPEEE*	Feedwater Importance to 18 Named Fire Areas	126	8.2×10^{-7}
External Qualitative	Feedwater Importance the same in external events as internal	126	2.1×10^{-6}
Licensee's External	Feedwater not important	125	0
Total ΔCDF (External + Internal)	Highest	126	8.4×10^{-6}
Total ΔCDF (External + Internal)	Lowest	125	7.7×10^{-7}
Total ΔCDF (External + Internal)	Analyst's Best Estimate	126	1.6×10^{-6}

* The designated quantities were used in determining the analyst's best estimate.

Table 4
External Events (Internal Fire)

Fire Area	Fire Affected Components	Case (Feedwater Failed)	Baseline (Feedwater at Nominal)	ΔCDF/year
AB-2/Z-1	HPCS pump and MOVs	5.64E-08	4.64E-11	5.65E-08
C13A	A Standby, Normal Service Water	3.55E-10	1.55E-11	3.70E-10
C13B	B Standby, Normal Service Water	3.55E-10	1.55E-11	3.70E-10
C18	Div I DC, Normal Service Water	4.92E-07	3.80E-09	4.96E-07
C19	Div II DC, Normal Service Water	4.74E-07	3.56E-09	4.78E-07
C20	Div III AC, HPCS Battery	9.77E-09	1.34E-10	9.91E-09
C21	Div III AC, Div III DC	7.53E-09	1.12E-10	7.64E-09
C22	Div III AC, HPCS	2.33E-08	1.38E-10	2.34E-08
C23	B Battery Charger, Inverter	1.04E-08	1.02E-09	1.14E-08
C26	A Battery Charger, Inverter	1.40E-08	1.04E-09	1.50E-08
DG1	Div II AC, D/G, 4160 Switchgear, Normal Service Water	1.61E-07	1.61E-09	1.63E-07
DG2	Div III AC, HPCS, Normal Service Water	2.39E-09	4.64E-11	2.44E-09
DG3	Div I AC, 4160 Switchgear, HPCS, LPCS, RHR A, RCIC, Normal Service Water	3.90E-07	4.51E-09	3.94E-07
DG-5/Z-1	Div III AC, DC, D/G	3.51E-07	2.80E-09	3.54E-07
DG-5/Z-2	Div III AC, DC, D/G	2.83E-07	2.18E-09	2.86E-07
PH-01/Z-1	A Standby Service Water	2.24E-08	2.47E-08	4.71E-08
PH-01/Z-2	A Standby Service Water	1.92E-09	6.80E-10	2.60E-09
PH-02/Z-1	B Standby Service Water	7.40E-09	4.28E-09	1.17E-08
PH-02/Z-2	B Standby Service Water	3.17E-09	1.93E-09	5.09E-09
TOTAL				2.36E-06