August 10, 2001

Mr. Oliver D. Kingsley, President
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: DRESDEN NUCLEAR GENERATING STATION UNIT 3 SPECIAL INSPECTION NRC INSPECTION REPORT 50-249/01-16(DRP)

Dear Mr. Kingsley:

On July 13, 2001, the NRC completed a Special Inspection at your Dresden Nuclear Generating Station Unit 3. The enclosed report documents the inspection findings which were discussed on July 13, 2001, with Mr. R. Fisher and other members of your staff.

On July 5, 2001, Unit 3 was manually scrammed from 100 percent power and a general station emergency plan Alert (the second lowest of four emergency classification levels) condition was declared by your staff due to unexplained increasing pressure in the primary containment. Your initial investigation determined that the Unit 3B reactor building closed cooling water (RBCCW) temperature control valve failed when the valve stem separated from the plug, which resulted in a temporary loss of containment cooling. No other significant equipment performance problems were noted subsequent to the reactor scram. Control room operators responded to the scram effectively and per procedure.

This event could have been avoided. An RBCCW temperature control valve failure with an identical root cause occurred at Quad Cities Station in 1998. Had there been effective communication between Quad Cities and Dresden Station personnel of the cause of the valve failure at the time, corrective actions could have been made in a timely manner at Dresden to prevent the scram.

Based on the risk and deterministic criteria specified in Management Directive 8.3, “NRC Incident Investigation Program,” and Inspection Procedure 71153, “Event Followup,” and due to the equipment performance problems which occurred, a Special Inspection was initiated in accordance with Inspection Procedure 93812, “Special Inspection,” to evaluate the facts and circumstances surrounding the event as well as the actions taken by your staff in response to the unexpected system performance issues encountered. The inspection focused on: (1) the development of the sequence of events related to the loss of RBCCW temperature control, the subsequent scram, and the actions following the scram, including event notification and classification; (2) the determination of the root cause of the loss of RBCCW temperature control; (3) safety and risk significance of the event; (4) the extent of condition review for the root cause of the loss of RBCCW; (5) operator performance during the event, including the adequacy of existing procedures; (6) the material condition of the service water system to determine if any damage was sustained as a result of the sudden loss of service water flow; (7) the adequacy of corrective actions taken, if any, for previously identified issues with

RBCCW temperature control; and (8) the implementation of the regulatory requirements of 10 CFR 50.65 (maintenance rule) for the RBCCW system.

Based on the results of this inspection, the inspector identified one issue of very low safety significance (Green). The finding was associated with the adequacy of the normal operating procedure for venting the drywell which was referenced within an emergency operating procedure and was considered a violation of NRC requirements. However, because of its very low safety significance and because it has been entered into your corrective action program, the NRC is treating this issue as a Non-Cited Violation in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny this Non-Cited Violation, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington DC 20555-0001; and the NRC Resident Inspector at the Dresden facility.

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

Sincerely,

/RA by Steven A. Reynolds Acting for/

Geoffrey E. Grant, Director
Division of Reactor Projects

Docket No. 50-249
License No. DRP-25

Enclosure: Inspection Report 50-249/01-16(DRP)

See Attached Distribution

See Previous Concurrence

DOCUMENT NAME: G:\dres\dre2001016 drp.wpd

To receive a copy of this document, indicate in the box: "**C**" = Copy without attachment/enclosure "**E**" = Copy with attachment/enclosure "**N**" = No copy

OFFICE	RIII		RIII	N	RIII			
NAME	PPelke/trn:dtp		MRing		GGrant			
DATE	08/01/01		08/10/01		08/10/01			

OFFICIAL RECORD COPY

cc w/encl: W. Bohlke, Senior Vice President, Nuclear Services
C. Crane, Senior Vice President - Mid-West Regional
J. Cotton, Senior Vice President - Operations Support
J. Benjamin, Vice President - Licensing and Regulatory Affairs
H. Stanley, Operations Vice President
J. Skolds, Chief Operating Officer
R. Krich, Director - Licensing
R. Helfrich, Senior Counsel, Nuclear
DCD - Licensing
P. Swafford, Site Vice President
R. Fisher, Station Manager
D. Ambler, Regulatory Assurance Manager
M. Aguilar, Assistant Attorney General
Illinois Department of Nuclear Safety
State Liaison Officer
Chairman, Illinois Commerce Commission

ADAMS Distribution:
AJM
DFT
LWR
RidsNrrDipmIipb
GEG
HBC
DRC1
C. Ariano (hard copy)
DRPIII
DRSIII
PLB1
JRK1

U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-249
License No:	DPR-25
Report Nos:	50-249/01-16(DRP)
Licensee:	Exelon Generation Company, LLC
Facility:	Dresden Nuclear Generating Station, Unit 3
Location:	6500 N. Dresden Road Morris, IL 60450
Dates:	July 9 through July 13, 2001
Inspectors:	C. Phillips, Senior Resident Inspector, Braidwood
Approved by:	Mark Ring, Chief Branch 1 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000249-01-16(DRP); on 07/09-07/13/01, Exelon Generation Company; Dresden Nuclear Generating Station; Unit 3; Special Inspection; inspection module 93812.

This special inspection examined the facts and circumstances surrounding a Unit 3 manual reactor scram and subsequent general station emergency plan Alert declaration which occurred on July 5, 2001. One Green finding was identified which also involved a non-cited violation.

The Braidwood Senior Resident Inspector conducted the inspection. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609, "Significance Determination Process" (SDP). The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

A. Inspector Identified Findings

Cornerstone: Barrier Integrity

Green. Dresden Emergency Operating Procedure 200-1, "Primary Containment," referenced the use of Dresden Operating Procedure 1600-1, "Normal Pressure Control of the Drywell or Torus," Revision 17, for venting the drywell with pressure below 2 pounds per square inch gauge. Dresden Operating Procedure 1600-1 was inappropriate for the circumstances in that the wording of the procedure prevented the performance of the Emergency Operating Procedure step. The inspector identified this as a non-cited violation due to an inadequate normal operating procedure for performing the Dresden Emergency Operating Procedure 200-1.

This event was more than minor because it had a credible impact on safety in that normal operating procedure steps precluded the performance of an emergency operating procedure step. The purpose of the emergency operating procedure step was to allow for a more controlled approach to minimizing drywell pressure before the introduction of drywell spray. Therefore, the venting of the drywell to maintain pressure and delay or prevent drywell spray has a credible impact on the amount of time operators have to combat an event which could eventually challenge the integrity of the reactor containment. The issue was determined to be of very low safety significance because the ability to spray down the drywell was not affected and, although complicated by the procedure inadequacy, the drywell could still be vented. (Section 1R5)

B. Licensee Identified Violations

No findings of significance were identified.

Report Details

Summary of Plant Event

On July 5, 2001, control room operators noticed that the Unit 3 primary containment (drywell) pressure was increasing steadily. The unit supervisor gave direction to lower reactor power but drywell pressure continued to rise and at 1.5 psig positive pressure in the drywell the reactor was manually scrammed. As drywell pressure continued to rise, Group 2 and 3 containment isolations were received and the emergency core cooling systems (ECCS) initiated. Operators took actions to prevent high pressure core injection (HPCI) from injecting after verifying that high pressure feedwater was available from the feedwater pumps after the scram. At a positive pressure of 2 psig in the drywell the licensee declared an Alert (the second lowest of four emergency classification levels) in accordance with Emergency Action Level FA1 of the General Station Emergency Plan. The drywell atmosphere was cooled by reactor building closed cooling water (RBCCW). After operators noticed that RBCCW temperatures were higher than expected a second train of RBCCW was valved in. Drywell pressure reached a maximum of 2.3 psig before it started to decrease. The licensee was operating with only one RBCCW train in service at a time. The rise in drywell pressure was caused by a failure of the single service water temperature control valve (TCV) for the RBCCW train in service. The TCV 3-3904-B valve stem separated from the plug causing it to fall into the flow stream. Operators performed post scram actions as expected. The RBCCW high temperature alarm had drifted out of calibration and sounded about 15 degrees higher than expected which may have slowed the operators' understanding of what was happening in the drywell. The alarm switch was recalibrated and there were no significant equipment malfunctions after the scram.

The Special Inspection

Based on the risk and deterministic criteria specified in Management Directive 8.3, "NRC Incident Investigation Program," and Inspection Procedure 71153, "Event Followup," and due to the equipment performance problems which occurred, a special inspection was initiated in accordance with Inspection Procedure 93812, "Special Inspection."

The purpose of the inspection was to evaluate the facts and circumstances surrounding the event as well as the actions taken by licensee personnel in response to the unexpected system performance issues encountered. In particular, the inspection focused on the following: (1) the development of the sequence of events related to the loss of RBCCW temperature control, the subsequent scram, and the actions following the scram, including event notification and classification; (2) the determination of the root cause of the loss of RBCCW temperature control; (3) safety and risk significance of the event; (4) the extent of condition review for the root cause of the loss of RBCCW; (5) operator performance during the event, including the adequacy of existing procedures; (6) the material condition of the service water system to determine if any damage was sustained as a result of the sudden loss of service water flow; (7) the adequacy of corrective actions taken, if any, for previously identified issues with RBCCW temperature control; and (8) the implementation of the regulatory requirements of 10 CFR 50.65 (maintenance rule) for the RBCCW system.

C. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R1 Sequence of Events Related to the Loss of RBCCW Temperature Control, the Subsequent Scram, and the Actions Following the Scram, Including Event Notification and Classification

a. Inspection Scope

The inspector reviewed logs, alarm printouts, and other documentation, including the licensee's preliminary investigation, interviewed cognizant individuals who responded to the scram, and developed the following sequence of events for the July 5, 2001, Unit 3 reactor scram:

Time	Event Description
09:58	Service water header pressure step change from 96 psig to 102 psig (indicative of sudden valve closure).
10:00	Unit 3 Nuclear Station Operator (NSO) noticed drywell to torus differential pressure was slightly elevated during routine panel monitoring. Two separate indications were reading about 1.3 psig. It was also noted that the torus nitrogen differential pressure valve was closed, the recirculation pump seal pressure was stable, the drywell floor drain sumps were not running or in alarm, and all drywell coolers were in service. The Unit 3 unit supervisor directed the NSO to manually scram the plant when drywell pressure reached 1.5 psig. The only abnormal alarm indication at this time was the drywell to torus differential pressure alarm.
10:06	A manual scram was inserted and emergency operating procedures were entered.
10:09	The ECCS initiated at 1.71 psig and containment group 2 and 3 containment isolations were received. The NSO took steps to prevent HPCI from injecting into the vessel after verifying that reactor feedpumps were supplying feedwater.
10:19	Alert declared.
10:36	2 psig in drywell.
10:38	Station minimum emergency response organization staffing requirements were met.
10:53	Station full emergency response organization staffing requirements were met.

10:55	Emergency Response Data system was activated.
10:59	RBCCW high temperature alarm annunciated.
11:04	Technical support center assumed command and control of the event.
11:13	Started 3A RBCCW pump (and heat exchanger).
11:28	Drywell pressure below 2 psig.
11:31	Emergency Operating Facility assumed command and control of the event.
15:00	Operations reports drywell sumps indicate normal leakage indicating no reactor coolant system leakage.
15:30	Maintenance crews determine that TCV 3-3904-B stem had separated from the plug.
16:02	Alert terminated.

The inspector reviewed the licensee's 10 CFR 50.72 notification against the sequence of events, alarm printouts, and control room logs. In addition, the inspector reviewed the Emergency Action Levels in the General Station Emergency Plan, Dresden Annex, against computer printouts of drywell pressure.

b. Findings

No findings of significance were identified.

1R2 Determination of the Root Cause of the Loss of RBCCW Temperature Control

a. Inspection Scope

The inspector interviewed cognizant maintenance and system engineering personnel. The licensee disassembled TCV 3-3904-B and found that the stem had separated from the plug. The stem threaded into the plug and was pinned in place with a type 420 stainless steel roll pin. The licensee found that the roll pin had corroded away; and in addition, the stem had been installed only hand tight into the plug and had not been torqued by the manufacturer to the specifications described in the vendor manual. The licensee sent the pin off for analysis and found that the pin was type 420 stainless steel and had failed from intergranular stress corrosion cracking. The inspector reviewed all previous preventive and corrective maintenance on TCV 3-3904-B to determine if the failure was caused by a Dresden station maintenance deficiency. The preventive maintenance for the valve was performed every three years since installation in 1994 and last performed in April 2000. The preventive maintenance called for an inspection of the stem and plug but did not require the pin to be removed and inspected. Therefore, there was no documentation as to the material condition of the pin at the time of the last inspection. Visual inspection of the pin without removal would make it difficult to identify early signs of degradation but should have indicated a failure of the pin. There were no corrective maintenance activities that involved the removal and reinstallation of the valve stem. The inspector reviewed condition reports (CRs) for the RBCCW system for the last two years and found no previous indication that would have predicted the failure of TCV 3-3904-B. However, the inspector reviewed a Quad Cities CR (Q1998-03589) which identified a similar failure in 1998 on RBCCW TCV 1-3904-B. Because Quad Cities operated with two RBCCW trains in service the valve failure did not have the same operational impact as the Dresden event. There were no formal communications between the Quad Cities and Dresden personnel to identify the failed pin to the Dresden engineering or maintenance staffs in 1998.

b. Findings

No findings of significance were identified. The inspector concluded that the failure of the RBCCW TCV was not due to a Dresden station maintenance deficiency. However, the failure (and resulting scram) could have been avoided with effective communication between Quad Cities and Dresden station personnel.

1R3 Safety and Risk Significance of the Event

a. Inspection Scope

The inspector interviewed the station probabilistic risk assessment engineer and reviewed the station risk assessment of conditional core damage probability (CCDP) for the event. The inspector also discussed the licensee's risk assessment with the NRC regional senior reactor analyst. The significance of operational power reactor events are evaluated for risk by determining the CCDP which reflects the loss of defense-in-depth due to the event, regardless of the cause. The CCDP accounts for actual plant configuration, including equipment rendered unavailable due to maintenance, testing, or other reasons. Although CCDP represents a fundamentally different concept for events than for degraded conditions that do not initiate an event, the same guidelines may be applied to each in assisting management in its risk-informed decision-making. Inspection Manual Chapter 0609, "Significance Determination Process," was designed to evaluate the incremental increase in core damage frequency due to a performance deficiency; however, it does not address CCDP determination for event assessments.

b. Findings

No findings of significance were identified. The inspector concluded that there was no significant difference between the risk assessment performed by the licensee and the NRC regional Senior Reactor Analyst.

1R4 Extent of Condition Review for the Root Cause of the Loss of RBCCW

a. Inspection Scope

The inspector interviewed system engineering personnel in regard to the extent of condition review to be conducted by the licensee. The licensee planned to replace the anti-rotation roll pin with a solid pin of different material in all the remaining RBCCW TCVs. The inspector reviewed a list generated by the licensee of all Copes-Vulcan valves installed on both units at Dresden Station. The licensee planned to review each Copes-Vulcan valve for similarities in construction and usage and develop a plan to change the pin design where necessary. In addition, the licensee notified the Exelon corporate engineering staff so that the issue could be reviewed at each of the Exelon stations.

b. Findings

No findings of significance were identified. The inspector concluded that the licensee's planned extent of condition review was sufficiently broad in scope.

1R5 Operator Performance During the Event, Including the Adequacy of Existing Procedures

a. Inspection Scope

The inspector interviewed members of the operating crew who were on shift during the event, simulator training instructors, and the emergency operating procedure coordinator. The inspector reviewed emergency operating procedures, abnormal operating procedures, normal operating procedures, alarm response procedures, and annunciator alarm printouts.

b. Findings

A finding of very low safety significance (Green) was identified for an inadequate drywell venting procedure which would prevent taking action directed in the emergency operating procedure to keep drywell pressure below 2 psig. The inadequate procedure was a violation of 10 CFR Part 50, Appendix B, Criterion V.

The inspector concluded that the operating crew followed procedures. However, the operating crew entered Dresden Emergency Operating Procedure (DEOP) 200-1, "Primary Containment Control," Revision 9, at 1.5 psig in the drywell. The DEOP stated, "Hold drywell and torus pressures below 2.0 psig using standby gas treatment and drywell purge (DOP [Dresden Operating Procedure] 1600-1)." The normal operating procedure for venting the drywell or torus, DOP 1600-1, "Normal Pressure Control of the Drywell or Torus," Revision 17, Step F.2 stated:

> Vent Containment to the Standby Gas Treatment System ONLY IF sample results indicate that release rates are LESS THAN the following Limits:
>
> a. Iodine 131: 3.5 X 10^{-7}uCi/cc.
>
> b. Beta/Gamma (total particulate): 3.9 X 10^{-6}uCi/c
>
> c. IF sample results are above the values stated in F.2.a and F.2.b, THEN venting through the Standby Gas Treatment System may occur

provided Radiation Protection performs an offsite dose calculation and verifies the release will be below 10 CFR 50 Appendix I limits.

Similar limits were placed on drywell purge by DOP 1600-1. The inspector asked the unit supervisor if the drywell had been lined up to be vented. The unit supervisor stated that since radiological conditions in the drywell were not known at the beginning of the event the drywell was not lined up to be vented. Control room operators do not know the drywell activity levels to the above specific levels at the beginning of an event. Since it takes between one and two hours to draw and count drywell samples, control room operators will never know the actual activity levels in the drywell because conditions will have changed by the time the sample results are received. Therefore, these procedural requirements preclude the performance of the DEOP step which directs the operators to vent the drywell to maintain pressure below 2 psig.

This finding had a credible impact on safety in that normal operating procedure steps precluded the performance of an emergency operating procedure step. The purpose of the emergency operating procedure step was to allow for a more controlled approach to minimizing drywell pressure before the introduction of drywell spray. Therefore, the venting of the drywell to maintain pressure and delay or prevent drywell spray has a credible impact on the amount of time operators have to combat an event which could eventually challenge the integrity of the reactor containment. Because this finding only affected the containment barrier cornerstone, the inspectors performed a Phase I analysis of the event using the Significance Determination Process. The inspectors answered "No" to all three questions regarding the containment barrier cornerstone which resulted in a Green finding. The inspector determined that the issue was of very low safety significance because the ability to spray down the drywell was not impacted and, although complicated by the procedure inadequacy, the drywell could still be vented. The inspector determined that, at the time of the event, procedure DOP 1600-1 was not appropriate to the circumstances, constituting a violation of 10 CFR Part 50, Appendix B, Criterion V, “Procedures.” However, because of the very low safety significance of the item and because the licensee has included this item in their corrective action program (CR D2001-03792) this procedure violation is being treated as a Non-Cited Violation (50-249/01-16-01(DRP)).

1R6 Material Condition of the Service Water System as a Result of the Sudden Loss of Service Water Flow

a. Inspection Scope

The inspector walked down the service water system and interviewed the system engineer to determine if there had been any adverse affects on material condition due to the sudden loss of service water flow when the TCV plug fell into the flow stream.

b. Findings

No findings of significance were identified. The service water system appeared to be in acceptable material condition from an external vantage point.

1R7 Adequacy of Corrective Actions Taken, If Any, for Previously Identified Issues With RBCCW Temperature Control

a. Inspection Scope

The inspector interviewed members of the system engineering and maintenance staffs, and reviewed CRs, including any associated apparent cause reports written between January 2000 and June 2001 associated with the RBCCW TCVs.

b. Findings

No findings of significance were identified. The RBCCW components were non-safety related, which meant that corrective action criteria from 10 CFR Part 50, Appendix B did not apply. However, several material condition problems with the RBCCW TCVs were not handled aggressively. For example, TCV 3-3904-A failed stuck at 10 percent open in November 1999 and was not scheduled for repair until April 2000. When TCV 3-3904-B failed at 60 percent open in February 2000, the priority on the work request on TCV 3-3904-A was upgraded and the valve was repaired quickly. Again TCV 3-3904-A was binding on November 11, 2000, December 12, 2000, and June 8, 2001. Discussions with maintenance staff personnel indicated that the reason for the binding was suspected to be an irregular surface area on the internal diameter of the spacer ring. The inspectors identified that the spacer ring had a lead time from the manufacturer of 67 days and had not been ordered until the week of July 2, 2001.

1R8 Implementation of the Regulatory Requirements of 10 CFR 50.65 (Maintenance Rule) for the RBCCW System

a. Inspection Scope

The inspector reviewed the licensee’s performance criteria for the RBCCW system and the maintenance rule data base where functional failure reviews of CRs on the system were documented. In addition, the inspector reviewed the licensee’s calculation that ties maintenance rule performance criteria to the licensee’s probabilistic risk assessment. Finally, the inspector interviewed the station’s maintenance rule coordinator.

b. Findings

No findings of significance were identified. The inspector identified that the licensee’s unavailability performance criteria for RBCCW heat exchangers did not agree with a station calculation that tied maintenance rule performance criteria to the licensee’s probabilistic risk assessment. The unavailability criteria for RBCCW heat exchangers was 7.5 percent versus 5 percent as stated in the calculation. The maintenance rule coordinator was unable to explain the difference. Actual unavailability for each heat exchanger was below 5 percent for the previous 24 months. In addition, the licensee identified that the performance criteria for RBCCW was established when the system was normally run with two trains and not updated when the normal system operation changed to a single train. These items were documented in CR D2001-03658, "NRC Question on RBCCW Maintenance Rule Criteria." The inspector concluded that maintenance rule regulatory requirements were met. However, because of the strict application of the rule requirements, repetitive significant component degradation without complete component failure resulted in a non-aggressive approach to system maintenance as discussed in the previous paragraph.

4OA6 Meetings

Exit Meeting

The inspectors presented the inspection results to Mr. P. Swafford and other members of licensee management at the conclusion of the inspection on July 13, 2001. The inspector subsequently conducted a further exit meeting with Mr. R. Fisher and other members of licensee management on August 6, 2001 because of a change in the characterization of the one Green finding. The licensee acknowledged the findings presented. No proprietary information was identified.

KEY POINTS OF CONTACT

Licensee

D. Ambler, Regulatory Assurance Manager
K. Bowman, Operations Manager
C. Cerovac, Training Manager
T. Fisk, Chemistry Manager
R. Fisher, Station Manager
B. Grant, Shift Operations Superintendent
B. Hanson, Work Management Manager
T. Luke, Engineering Manager
J. Moser, Radiation Protection Manager
J. Nalewajka, Acting Nuclear Oversight Manager
R. Ruffin, NRC Coordinator
W. Stoffels, Maintenance Manager

Nuclear Regulatory Commission

G. Grant, Director, Division of Reactor Projects, Region III
M. Ring, Chief, Reactor Projects, Branch 1

LIST OF ITEMS OPENED

50-249/01-16-01	NCV	Procedure Inappropriate for the Circumstances

LIST OF ITEMS CLOSED

50-249/01-16-01	NCV	Procedure Inappropriate for the Circumstances

LIST OF ACRONYMS USED

ADAMS	Agencywide Documents Access and Management System
CCDP	Conditional Core Damage Probability
CCSW	Containment Cooling Service Water
CFR	Code of Federal Regulations
CR	Condition Report
DEOP	Dresden Emergency Operating Procedure
DOP	Dresden Operating Procedure
DW	Drywell
ECCS	Emergency Core Cooling System
HPCI	High Pressure Core Injection
HX	Heat Exchanger
IN	Information Notice
LPCI	Low Pressure Core Injection
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulations
NSO	Nuclear Station Operator
OSC	Operations Support Center
PIF	Problem Identification Form
PMT	Post Maintenance Test
PSA	Probabilistic Safety Assessment
psig	Pounds Per Square Inch Gauge
RBCCW	Reactor Building Closed Cooling Water
SDP	Significant Determination Process
TCV	Temperature Control Valve
TSC	Technical Support Center
URI	Unresolved Item
WR	Work Request

LIST OF DOCUMENTS REVIEWED

1R1 Sequence of Events Related to the Loss of RBCCW Temperature Control, the Subsequent Scram, and the Actions Following the Scram, Including Event Notification and Classification

Dresden Event Notification 38116	Unit 3 Manually Scrammed from 100 Percent Power Due to High Drywell Pressure	July 5, 2001
DOS 1400-05	Core Spray System Pump Test With Torus Available	Revision 25
	Desk Operator Center Log	July 5, 2001
	Control Room Log	July 5, 2001
	Drywell Pressure Graph	July 5, 2001
	Drywell Temperature Graph	July 5, 2001
	U2 Fire Header Pressure Graph	July 5, 2001
	U2 and U3 Sequence of Events Recorder Printout	July 5, 2001
PORC #01	Review of July 5, 2001 Dresden Unit 3 Drywell Pressure/Temperature Transient Event	July 6, 2001
	Event Summary Report of Alert Emergency Declared at Dresden Station	July 5, 2001
	OSC [Operations Support Center] Communicator Log	July 5, 2001
EP-AA-112	OSC Team Briefing Form - Team No. 5	Revision 3

1R2 Determination of the Root Cause of the Loss of RBCCW Temperature Control

WR 990081150 01	Disassemble TCV 3-3904-B and Remove Balance Seal From Plug	March 15, 2000
WR D20789	Overhaul RBCCW 3-3904B TCV	June 1994
CR 2001-03550	U3 ECCS System Operation During Reactor Scram and Subsequent High DW Press	July 5, 2001

1R3 Safety and Risk Significance of the Event

ER2001-9955	Risk Evaluations Pertinent to 7/5/01 Manual Scram of Dresden Unit 3	July 12, 2001
	Component Maintenance History for TCV 3-3904-B	July 5, 2001

1R4 Extent of Condition Review for the Root Cause of the Loss of RBCCW

MS Access Worklist	Copes-Vulcan Valve List - Dresden Station	July 10, 2001

1R5 Operator Performance During the Event, Including the Adequacy of Existing Procedures

CR D2001-03792	NRC Exit For Special Inspection Noted Items Which Require Follow Up	July 13, 2001
CR D2001-03646	Unexpected Response of MO 2-1501-21B	July 5, 2001
DEOP 200-1	Primary Containment Control	Revision 9
DOP 1600-01	Normal Pressure Control of Drywell or Torus	Revision 17
DOP 1600-15	Unit 2(3) Post-Accident Operations Venting and Sampling of the Primary Containment Atmosphere and Reactor Water Sampling	Revision 11
DOP 3700-01	RBCCW System Operation	Revision 18
DOP 5750-11	Unit 2(3) Drywell Cooling system Operation	Revision 9
DGP 02-03	Reactor Scram	Revision 54
DAN 902(3) A-13	Unit 2(3) Drywell Press HI	Revision 12
DAN 902(3)-4 G-17	Unit 2(3) Drywell Atmosphere Temp HI	Revision 8
DAN 902(3)-5 D-11	Unit 2(3) Drywell Press Hi-HI	Revision 11
DAN 902(3)-5 G-5	Unit 2(3) Drywell Press HI	Revision 9
DAN 923-1 E-1	U2 or U3 RBCCW Temp HI	Revision 2
Panel 902(3)-3 Operator Aid Designation #55	HPCI Control/Shutdown With Initiation Signal Present	November 28, 1999 Revision 2
Panel 902(3)-3 Operator Aid Designation #59	LPCI [Low Pressure Core Injection]/CCSW [Containment Cooling Service Water] Operations	Revision 4

1R7 Adequacy of Corrective Actions Taken, If Any, for Previously Identified Issues With RBCCW Temperature Control

IN 94-55	Problems With Copes-Vulcan Pressurizer Power-Operated Relief Valves	August 4, 1994
CR Q1998-03589	1-3904B TCV Disc Separated From Stem While Inservice	August 22, 1998
CR D2000-00142	3A RBCCW Heat Exchanger Service Water Outlet TCV	January 11, 2000
CR D2000-00719	3B RBCCW HX TCV Not Functioning Properly	February 8, 2000
CR D2000-01184	Unable to Complete Scheduled Activity - Placing 2A RBCCW HX In Service Due to Elevated RBCCW te	February 28, 2000
CR 2000-02094	3B RBCCW Failed to Control RBCCW Temperature and PMT Was Failed	April. 8, 2000
CR 2000-02144	3B RBCCW TCV Not Responding Properly	April 12, 2000
CR 2000-02802	Apparent Wrong Trim in 2A RBCCW HX TCV	May 15, 2000
CR 2000-06280	3A RBCCW HX TCV Erratic Operation	November 17, 2000
CR 2000-06284	3A RBCCW TCV Causes Operations to Swap Heat Exchangers	November 17, 2000
CR 2000-06981	3A RBCCW HX TCV Binding	December 31, 2000
CR 2001-03080	U3 RBCCW TCV Binding and Affecting Drywell Equipment and Parameters	June 8, 2001
Drawing VA.D.LSD.–CFLC-.D-342406	Copes-Vulcan Series D-600 Valve Assembly With Model D600-5D Actuator 12" Class 125	October 23, 1992
System Materials Analysis Department Report	Failure of a Roll Pin from the #1-3904B TCV at Quad Cities Station	September 22, 1998

1R8 Implementation of the Regulatory Requirements of 10 CFR 50.65 (Maintenance Rule) for the RBCCW System

CR D2001-03658	NRC Question on RBCCW Maintenance Rule Criteria - NRC Identified	July 11, 2001

DRE Z37-1	Maintenance Rule Performance Criteria To Provide Cooling for Equipment and Systems in Reactor Building and Radwaste Building	July 10, 2001
Calculation DRE 98-0021	PSA [Probabilistic Safety Assessment] Basis for Dresden Maintenance Rule Availability Performance Criteria	Revision 2