June 13, 2003

Mr. John L. Skolds, President
Exelon Nuclear
Exelon Generation Company, LLC
Quad Cities Nuclear Power Station
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: QUAD CITIES NUCLEAR POWER STATION, UNIT 2
NRC SPECIAL INSPECTION REPORT 50-265/03-06

Dear Mr. Skolds:

On April 29, 2003, the U. S. Nuclear Regulatory Commission (NRC) completed a special team inspection at your Quad Cities Nuclear Power Station, Unit 2. The enclosed report documents the inspection findings which were discussed with Mr. Tulon and other members of your staff on April 29, 2003.

At 1:22 p.m. on Wednesday, April 16, 2003, power operated relief valve (PORV) 3B on Unit 2 unexpectedly opened and failed to close. When temperature in the torus reached 95 degrees Fahrenheit, the reactor was manually scrammed. The plant shut down as expected. At 1:59 p.m. an ALERT (the second lowest of four emergency classification levels) was declared according to emergency procedures for a stuck open relief valve. Based on the risk and deterministic criteria specified in Management Directive 8.3, “NRC Incident Investigation Program,” and Inspection Procedure 71153, “Event Followup,” a Special Inspection was initiated in accordance with Inspection Procedure 93812, “Special Inspection.” The purpose of the Special Inspection was to evaluate the facts, circumstances, and your staff’s actions surrounding this event.

The failing and sticking open of the 3B PORV and subsequent ALERT probably could have been prevented if you had adequately monitored and trended the tailpipe temperature and taken the necessary action to repair the leaking pilot.

Based on the results of this inspection, one finding of very low safety significance (Green) was identified. This finding was that inadequate monitoring and trending was performed for tailpipe temperatures on the 3B power operated relief valve. This resulted, in part, from failure to fully implement vender recommendations on tailpipe temperature monitoring and inadequate resolution of long term high temperature readings. The response to the event, use of emergency procedures, event classification, and notifications by your staff were observed to be good.

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 (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 by Steven A. Reynolds Acting for/

Geoffrey E. Grant, Director
Division of Reactor Projects

Docket Nos. 50-254; 50-265
License Nos. DPR-29; DPR-30

Enclosure: Inspection Report 50-265/03-06

See Attached Distribution

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REGION III

Docket Nos:	50-265
License Nos:	DPR-30
Report No:	50-265/03-06
Licensee:	Exelon Nuclear
Facility:	Quad Cities Nuclear Power Station, Unit 2
Location:	22710 206th Avenue North Cordova, IL 61242
Dates:	April 16 through April 29, 2003
Inspectors:	R. Lerch, Project Engineer M. Kurth, Acting Senior Resident Inspector B. Dickson, Resident Inspector, Dresden A. Dunlop, Engineering Inspector
Approved by:	Mark Ring, Chief Branch 1 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000265/2003-006; Exelon Nuclear; on 4/16/03-4/29/03, Quad Cities Nuclear Power Station; Unit 2. Special Inspection for a stuck open power operated relief valve and reactor scram "ALERT" event.

This special inspection examined the facts and circumstances surrounding a Unit 2 spurious opening of a power operated relief valve, manual reactor scram and subsequent general station emergency plan ALERT declaration which occurred on April 16, 2003. One Green finding was identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be "Green" or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

Inspector-Identified and Self-Revealing Findings

Cornerstones: Barrier Integrity and Initiating Events

Green. The inspectors identified a Green finding for deficient monitoring and trending of tailpipe temperatures on the 3B power operated relief valve due, in part, to not fully implementing the recommendations of General Electric Service Information Letter 196 and the long-term acceptance of high temperatures that masked a potential degraded condition. (Section 1R3)

This issue was more than minor because the issue is associated with both the Initiating Events and the RCS (reactor coolant system) Barrier Cornerstones due to the relief valve spuriously lifting. This directly affects the associated cornerstone objectives of limiting the likelihood of those events that upset plant stability and maintaining the functionality of the reactor coolant system. This capability is important for mitigating events which can lead to core damage. A Phase 3 analysis concluded the safety significance of the inspection finding based on the change in CDF (core damage frequency) to be very low.

REPORT DETAILS

Summary of Plant Status

Both Units 1 and 2 were operating at their full achievable power level of 912 MWe prior to the event.

Unit ½ emergency diesel generator (½ means able to supply both Unit 1 and Unit 2) was out of service for preventive maintenance and inoperable. The associated diesel generator cooling water pump was also inoperable. Therefore, both Unit 1 and Unit 2 were in a 7-day limiting condition for operation in accordance with Technical Specification (TS) 3.8.1.B.

Summary of Plant Event

On April 16, 2003, an “ALERT” emergency condition was entered for Unit 2 following the spurious opening of the 3B power operated relief valve (PORV). Operators were unsuccessful in their attempts to close the valve. The discharge from the open valve is piped directly to the torus (suppression pool). Torus water temperature increased as the open valve transferred steam from the reactor. The residual heat removal (RHR) system in torus cooling mode could not overcome the amount of heat transferred to the torus. This was the expected torus temperature response for a stuck open relief valve. The operators initiated a manual scram of the reactor. With the reactor shut down, torus cooling began to reduce the suppression pool temperature. Simultaneously, the reactor depressurized, forcing less energy into the pool. Operators closed the main steam isolation valves to reduce the rate of depressurization and reactor cooldown. The cooldown rate did exceed the Technical Specifications limit of 100 degrees Fahrenheit/hour. After several hours, the reactor depressurized to a point at which shutdown cooling could be initiated. The relief valve closed at approximately 50 psi of system pressure.

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 relief valve failure, a special inspection was initiated in accordance with Inspection Procedure 93812, “Special Inspection.”

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R1 Sequence of Events Related to the 3B Power Operated Relief Valve, the Reactor Shutdown and Cooldown, and Event Classification and Notifications

.1 Sequence of Events for the ALERT Emergency Condition

a. Inspection Scope

The inspectors reviewed logs and sequence of event recorder printouts from the control room, Technical Support Center (TSC), and the Emergency Operations Facility (EOF). The inspectors interviewed several plant personnel to validate the sequence of events. Also, the inspectors observed control room activities as the event progressed. Based on the reviews, interviews, and direct observation, the inspectors developed a sequence of events for the April 16, 2003, ALERT emergency condition resulting from the spurious opening of the 3B power operated relief valve.

b. Findings

The inspectors determined the following sequence of events resulted in the ALERT emergency condition and its subsequent termination on April 16 through 17, 2003:

April 16, 2003

Time (all times in Central Daylight Time)	**Event Description**
1:22 p.m.	Control room alarm received indicating 3B Power Operated Relief Valve (PORV) Open. Operators enter procedure QCOA 0203-01, “Failure of a Relief Valve to Close or Reseat Properly”
1:30 p.m.	Initiated 2B and 2C RHR service water (SW) pumps and 2A, 2B, 2C, and 2D RHR pumps for torus cooling in accordance with Quad Cities Operating Procedure (QCOP) 1000-09, “Torus Cooling Startup and Operation”
	Begin Torus Temperature monitoring
1:37 p.m.	Manual Scram initiated
	Entered Quad Cities General Abnormal procedure (QGA) 200, “Primary Containment Control,” when torus temperature reached 95 degrees Fahrenheit
	Entered QGA 100, “RPV Control,” when reactor water level less than zero inches due to plant response to manual scram
1:40 p.m.	Pulled electrical fuses for 3B PORV and entered Technical Specification 3.4.3.A for Automatic Depressurization System (ADS) valve inoperable

1:44 p.m.	Reactor water level maintained above 0 inches with a band of 0 to 48 inches established and to be controlled by feedwater injection
1:50 p.m.	Scram signal reset
1:52 p.m.	MSIVs closed in attempt to minimize reactor cool down
1:59 p.m.	General Station Emergency Plan (GSEP) ALERT declared when torus temperature reached 110 degrees Fahrenheit with stuck open 3B power operated relief valve
	TSC and EOF activated
2:06 p.m.	Site notified Illinois Emergency Management Agency, Illinois Department of Nuclear Safety, Iowa Emergency Management Division, Scott County Sheriff, and Clinton County Emergency Operations Center of the ALERT emergency condition
2:47 p.m.	Site notified NRC Operations Center of the ALERT Emergency Condition
3:45 p.m.	Torus water temperature decreased to less than 110 degrees Fahrenheit
5:00 p.m.	Exited QGA 100 after reactor water level was stabilized
5:24 p.m.	Torus temperature reduced to less than 95 degrees Fahrenheit
6:08 p.m.	QGA 200 exited
9:04 p.m.	Shutdown cooling initiated
9:06 p.m.	3B PORV indicating closed
10:37 p.m.	Mode 4 entered. Cold shutdown conditions reached
10:51 p.m.	ALERT condition exited
	Site notified NRC, Illinois Emergency Management Agency, Illinois Department of Nuclear Safety, Iowa Emergency Management Division, Scott County Sheriff, and Clinton County EOC of termination of the ALERT emergency condition

April 17, 2003

12:22 a.m.	Per EP-AA-115, "Recovery from a Classified Event," event response terminated
	Site notified Illinois Emergency Management Agency, Illinois Department of Nuclear Safety, Iowa Emergency Management Division, Scott County Sheriff, and Clinton County EOC of termination of emergency condition

.2 Personnel Response to Event Condition

a. Inspection Scope

The inspectors evaluated the licensee's response to the event in the areas of operations and emergency preparedness program implementation. The inspectors interviewed operations crew members who were on-shift during the time of the event. Emergency preparedness personnel, TSC staff, and site management were also interviewed. The inspectors had direct observation of operations personnel during the event conditions. Control room logs, emergency classification, sequence of event printouts, and notification sheets were reviewed.

b. Findings

The inspectors verified that the operations crew responded appropriately to the event conditions. The inspectors verified that the operators responded as required to the alarm response procedures upon receipt of the control annunciators for the 3B power operated relief valve opening. In accordance with the alarm response procedure and QCOA 0203-01, "Failure of a Relief Valve to Close or Reseat Properly," operations personnel attempted to close the valve using the valve position keylock switch and were unsuccessful in their attempts.

The shift supervisor provided the proper command and control during the event. Instructions were provided and completed by crew members. The inspectors observed that the proper procedures were implemented in attempting to close the valve. The operating crew appropriately anticipated the continued heatup of the torus water with the valve stuck open. Emergency operating procedure QGA 200, "Primary Containment Control," was in use by the shift supervisor to anticipate what actions were needed if certain parameters were met or exceeded. In particular, the torus temperature heatup limitations and mitigating actions that were to be taken. The reactor cooldown rate was reduced by closing the main steam isolation valves. Although the cooldown rate of 108 degrees Fahrenheit/hour exceeded the 100 degrees Fahrenheit/hour limit of Technical Specification 3.4.9, the action statement, to determine reactor coolant system operability, was met through a subsequent analysis performed by General Electric.

The inspectors verified that accurate and timely classifications and notifications were made during the course of the event. Licensee personnel notified the federal, state, and local authorities and provided the necessary information throughout the event.

1R2 Determination of the Initial Cause of the Stuck Open Power Operated Relief Valve

a. Inspection Scope

The inspectors reviewed the licensee's initial cause determination of the stuck open power operated relief valve (PORV). The licensee's evaluation included potential electrical and mechanical causes.

The two electrical issues reviewed by the licensee were a potential short/ground, which could have provided energy to the solenoid to cause the PORV to electrically open; and a dim position indication light for the 3B PORV on the main control board. An in-field troubleshooting plan found no electrical abnormalities. In addition, attempts to close the valve by removing the power fuses were unsuccessful. As for the cause of the dim “open” indicating light, the licensee determined it was most likely due to a poor contact within the position indication reed switch. The reed switch is actuated when the permanent magnet comes near it and actuates the switch. The licensee indicated there is a known problem with properly locating the switches in their optimum location since the valve is never fully open when the switches are calibrated. The valve’s permanent magnet likely went slightly past the reed switch and did not fully close the switch.

The mechanical issues included leakage past the PORV pilot seat due to steam cutting or foreign material in the seat, the location of the thermocouples on the downstream tailpipes used to indicate valve seat leakage, and valve testing conducted prior to installation and at low power. Prior to removing the 3B valve from the system, the licensee removed the downstream elbow such that they were able to view the valve main seat and verified there was no significant foreign material that caused the valve to stick open. When the valve was disassembled, the licensee determined there was significant steam cutting on the internal main and secondary pilot valves, which caused the amount of steam flow past the pilot valves to exceed the orifice flow into the PORV control chamber. When the control chamber pressure on top of the main PORV disk became less than the line pressure acting on the bottom of the main PORV disk, the 3B PORV failed open. The valve did not reseat until the pressure in the line was approximately 50 psig.

The PORV pilot valve leakage phenomenon has been know for many years within the industry. General Electric (GE) Service Information Letter (SIL) 196 Supplement 4 dated October 31, 1977, “Thermocouple Location for Safety/Relief Valve Discharge Lines,” provided guidance on methods to monitor for this condition and to take action to repair/replace the leaking pilot valve assembly before the PORV could inadvertently open. The SIL provided recommendations for the proper type of thermocouple and its location in the tailpipe, along with acceptance criteria in order to determine when the PORV leakage is approaching the 50 to 100 lbm/hr steam flow value that the vendor determined may actuate the PORV.

b. Findings

Introduction. The inspectors identified a Green finding for deficient monitoring and trending of tailpipe temperatures on the 3B PORV due in part to not fully implementing the recommendations of GE SIL 196 and the long-term acceptance of high temperatures that masked a potential degraded condition.

Discussion. The GE SIL contained several issues and recommendations that were not fully implemented or addressed by the licensee. These issues included the positioning of the thermocouple on the discharge piping, the type of thermocouple installed, and the normal expected temperature reading from the thermocouple. The licensee accepted the long-term high temperatures on the tailpipe due to the belief the temperature in the tailpipe would significantly increase prior to valve actuation. The licensee also did not have adequate acceptance criteria for seat leak testing the PORVs.

In addition, during previous refurbishment testing, there was significant leakage which was caused by a bent main disk rod preventing the disk from properly seating. The licensee had not evaluated whether the bent main disk rod that was straightened during the previous refurbishment played a role in causing the pilot valve leakage. The licensee planned to review this issue during the licensee's formal root cause process.

Normal Expected Temperature Reading from the Thermocouple

The SIL recommended that the normal expected thermocouple reading in the tailpipe be between 160-190F. However, the normal temperature for the 3B PORV was greater than 200 degrees Fahrenheit. Under these conditions, the expected temperature increase of 30 degrees Fahrenheit that would indicate pilot valve leakage as discussed in the SIL would be masked due to the higher normal temperature. Leakage through the pilot valves seats would not cause the same temperature increase as it would if the normal temperature was within the range recommended by the SIL.

Location of the Thermocouple on the 3B PORV Tailpipe

One of the recommended actions of the SIL involved the location of the thermocouple in the PORV discharge piping. The preferable location of the thermocouple is stated as 4.5 to 6 feet downstream of the valve's discharge flange in a straight run of pipe. For the 3B PORV, a 2 foot radius elbow is attached to the PORV discharge flange, which then is connected to piping that slopes downward to conform with the inside drywell wall. The licensee located the thermocouple for the 3B PORV approximately 1 foot downstream of the elbow, which was approximately 3 feet from the valve's discharge flange. This did not conform to the SIL recommendations of 4.5 to 6 feet downstream of the valve discharge flange in a straight run of pipe. Having the thermocouple so close to the valve resulted in a higher (>200 degrees Fahrenheit) than expected "normal" tailpipe temperature. As a result, the temperature increase of 30 degrees Fahrenheit expected by the SIL for a leaking valve was masked by this higher "normal" tailpipe temperature and did not trigger the appropriate response by the licensee to address valve leakage prior to a valve spurious operation.

The licensee confirmed for the other three PORVs in Unit 2, the locations of the thermocouples were within the preferable location as recommended by the SIL.

Type of Thermocouple Installed

The SIL recommended that the thermocouple in the tailpipe be located in a thermowell such that the device would read discharge line internal temperature, as opposed to a strap-on thermocouple that reads pipe wall temperature. This recommendation was initially proposed in GE SIL 178, which discussed the improved thermocouple and thermowell design. The licensee did not appear to address this SIL recommendation in their initial cause determination until after the concern was raised by the inspectors. The licensee obtained a letter from GE stating that it was acceptable under certain cases to use strap-on thermocouples. The letter stated that strap-on thermocouples typically provide a slower response time than penetration-type thermocouples. However, the time dependent leakage phenomena caused by corrosion/erosion allows adequate time for strap-on devices to function. The GE letter stated that the location of the thermocouple was the most important factor in detecting temperature changes. The letter also recommended that Exelon continue its practice of periodic inspections each refueling outage of the thermocouples to ensure there is not a film buildup between the pipe and the device, which would reduce its sensitivity.

Long Term Acceptance of High Tailpipe Temperatures

The Target Rock PORVs were installed in Unit 2 during a refueling outage in 1997. Since that time, the licensee had noted higher temperatures on the 3B PORV tailpipe thermocouple than the other PORV line thermocouples. Problem Identification Form (PIF) Q1997-03537 evaluated the elevated tailpipe temperature for the 3B PORV. The PIF evaluation concentrated on the PORV performing its safety function of opening when needed and that these functions would not be affected by the elevated temperature. The PIF stated that the steam leakage through the valve had not caused a spontaneous opening and it would not in the future without increased leakage across the pilot, which would be shown by increased tailpipe temperatures and increased torus temperature heatup rates. However, as shown by the April 16, 2003, spontaneous 3B PORV opening event, there was not a significant increase in tailpipe temperature, nor an increase in torus temperature heatup rates prior to the valve failing fully open. The 3B PORV thermocouple reading had started increasing in early February 2003 from 208 degrees Fahrenheit to 214 degrees Fahrenheit, a change of 6 degrees, which, based on a review of data for other safety/relief valves, was not a significant change. The other three PORVs also showed a slight increase at that time, although their temperatures fluctuated within ±2 to 3 degrees Fahrenheit of their "normal" temperatures, while the 3B PORV temperature continued to rise.

Coming out of the last refueling outage in 2002, Action Request (AR) 00109682 documented an increase in dose rates on the bottom of the torus, which was attributed to 3B PORV leakage. Corrective action was to replace the 3B PORV during the next outage of sufficient duration. No monitoring or trending of tailpipe temperatures for valve degradation was specified.

Approximately 1 month prior to the spontaneous opening event, numerous "3B PORV open" nuisance alarms were received on the sequence of events recorder and documented in AR 00098715. The operators verified the PORV was not open via tailpipe temperatures and acoustic monitoring prior to disabling the alarm and initiating a work request to repair the cause. Resolution of the problem was to increase the sequence of events recorder alarm activation debounce settings from 1 msec to 15 msec such that the signal had to be present for a longer period before activating the alarm. Although these actions appeared appropriate to reset the alarm activation sequence, it did not address or evaluate the continuing degradation of the PORV. At that time, the temperature had increased approximately 5 degrees Fahrenheit from the previous "normal" temperature. Again no additional monitoring or trending of tailpipe temperatures for valve degradation was performed.

PORV Testing

The licensee replaced the PORVs with refurbished valves every other refueling outage (approximately 4 years). The valves were tested, after refurbishment in the shop, with nitrogen at 600 psi to verify actuation and to conduct seat leakage tests. When the 3B PORV was refurbished and tested, it was determined to have a pilot leak rate of 26 cc/minute. The PORV vendor indicated during the initial cause determination that a leak rate of less than 5 cc/minute was considered "like-new." The licensee test procedure did not have any maximum allowable leakage criteria for the pilot valve. Testing of the PORVs is required by American Society of Mechanical Engineers (ASME) Section XI Code (the Code), which references OM-1, “Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices,” for the testing of relief devices. The PORVs are ASME Code Class 1 relief devices, whose testing requirements are stated in OM-1, Section 3.3.1, “ASME Class 1 Pressure Relief Devices.” Required testing includes a seat tightness determination. This is further discussed in OM-1, Section 4.2, “Seat Tightness Testing,” which requires acceptance criteria from either the original valve equipment design specification or as established by the licensee. The work requests that performed the required seat leakage test did not contain any acceptance criteria. Although in some cases the licensee determined there were valve failures based on excessive leakage, no criteria had been established as required by the Code. Although the vendor specified a 5 cc/minute leak rate to be considered “like-new,” the licensee subsequently established a 29 cc/minute leak rate acceptance criteria as allowed by the Code. As such, all the PORV leak rate tests for the installed valves met the new licensee established criteria. Failure to establish an acceptance criteria in accordance with the Code requirement is considered a violation of 10 CFR 50.55a (f)(4). Since an acceptance criteria was subsequently established by the licensee that met the requirements of the Code, this finding constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. The licensee documented the problem in Condition Report (CR) 156277.

Analysis. In accordance with Inspection Manual Chapter 0612, the inspectors determined the finding, a spurious stuck open PORV due to deficient monitoring and trending, was more than minor because the issue was associated with both the Initiating Events and the reactor coolant system (RCS) Barrier Cornerstones due to the relief valve spuriously lifting. This directly affects the associated cornerstone objectives of limiting the likelihood of those events that upset plant stability and maintaining the functionality of the reactor coolant system. This capability is important for mitigating events which can lead to core damage. Since the finding was associated with both initiating events and the RCS barrier integrity, the Reactor Safety Significance Determination Process (SDP) applied.

In accordance with Inspection Manual Chapter 0609, Attachment A, the inspectors conducted an SDP Phase 1 screening and determined that the finding impacted the reactor coolant system barrier integrity. Thus an SDP Phase 2 evaluation was required.

Based on the Phase 2 evaluation, the dominant sequence [SORV + CHR + CV] was found to be potentially risk significant. A review of these results indicated that they were conservative; Quad Cities SDP Worksheets did not provide credit for the power conversion system (PCS) following a stuck-open relief valve (SORV) and no credit was provided for late injection (LI) following containment failure. PCS was not credited on the worksheets because under certain scenarios, such as loss of off-site power, PCS would not be available. Late injection was not credited on the worksheets, as a general rule, because core damage is generally assumed following containment failure.

Crediting late injection in the SORV event tree results in a modified dominant sequence (SORV+CHR+CV+LI).

The Region III Senior Reactor Analyst (SRA) performed a Phase 3 analysis and reevaluated the dominant sequence, considering the appropriateness of crediting late injection (LI) following containment failure. In the SDP worksheet model, it is assumed that core damage will occur if containment heat removal fails due to failure of both suppression pool cooling and containment venting; therefore, late injection was not credited. In the Quad Cities probabilistic risk analysis (PRA) model, it is assumed that in 94 percent of such cases the drywell failure will be favorable following late injection. In discussions with the licensee it was determined that the containment is expected to fail in such a manner that it would result in a slow depressurization of the containment verses a catastrophic containment failure. The failure is expected to be above the suppression pool water line and would have minimal effect on the suppression capability of the steam being discharged into the suppression pool from the PORV vent piping. The SRA therefore determined that it would be appropriate for this finding to credit LI following containment failure. The systems that were found to be available to support LI following the containment failure are condensate/feedwater, control rod drive (CRD), and safe shutdown makeup (SSMP). These systems do not take a suction from the suppression pool and are; therefore, not affected by containment failure. The analyst determined that it would be appropriate to credit SSMP and condensate as a LI source. This credit was assigned a failure probability of 2 (E-2) consistent with other applications of LI in the SDP worksheets. This value is considered reasonable as the actions necessary to accomplish LI are straight forward operator actions that can be performed from within the control room.

The SRA concluded the safety significance of the inspection finding based on the change in CDF to be Green **(50-265/2003-06-01)**. A Green finding represents a finding of very low safety significance.

Enforcement

- The finding did not constitute a violation of regulatory requirements.

- The inspectors agreed with the licensee's apparent root cause which was concluded to be steam cutting damage to PORV pilot valve seats and discs.

- Inadequate monitoring of the valve impeded the licensee from responding to indications and possibly preventing this event.

1R3 Extent of Condition Review for the Root Cause

a. Inspection Scope

The inspectors reviewed the licensee's assessment of the extent of condition for the event. The 93V-001 Model Target Rock PORVs are only installed in Quad Cities Unit 2 (3B, 3C, 3D, 3E). The Unit 1 PORVs are Dresser Electromatic valves, which the licensee also considered to be potentially subject to the same problem of valve leak-by and tailpipe temperature monitoring. Both Unit 1 and Unit 2 3A PORVs are 3-stage Target Rock relief/safety valves and are also potentially subject to the same inadvertent opening event.

Dresden PORVs were also reviewed for applicability. Each Dresden unit has four Dresser Electromatic relief valves and one 3-stage Target Rock relief/safety valve.

Dresden had recently (November 2002) moved the tailpipe thermocouple for the 3A relief/safety valve further downstream to reduce the normally “high” temperature reading of 200 to 225F. The thermocouple was moved to comply with the General Electric SIL and tailpipe temperature readings are now consistent with the other PORVs at Dresden.

The licensee conducted walkdowns of the other Quad Cities Unit 2 PORVs and determined that the tailpipe thermocouples were located in an optimum location to identify leakage through the valves. A review of tailpipe temperature data indicated that the temperatures acted in concert with changes in reactor power and steam flow, different from the 3B PORV temperature. This data provided the licensee with evidence that the tailpipe temperature readings were providing valid trend data of valve leakage.

On Unit 1, the licensee had “high” tailpipe temperatures (>200F) on three valves; two Electromatic PORVs and one relief/safety valve. In May 2003, the licensee replaced these valves. In addition, the licensee verified proper location of the strap-on tailpipe thermocouple to monitor valve seat leakage for the one 3-stage Target Rock relief/safety valve. The thermocouples for the Electromatic PORVs are located in thermowells per the General Electric SIL.

b. Findings

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

4OA6 Meetings

Exit Meeting

The inspectors presented the inspection results to Mr. T. Tulon and other members of licensee management at the conclusion of the inspection on April 29, 2003. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

KEY POINTS OF CONTACT

Licensee

T. Tulon, Site Vice President
B. Swenson, Plant Manager
W. Beck, Regulatory Assurance Manager
M. Perito, Operations Manager

Nuclear Regulatory Commission

S. Reynolds, Deputy Director, Division of Reactor Projects
M. Ring, Chief, Reactor Projects Branch 1

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

50-265/2003-06-01	FIN	Deficient Monitoring and Trending of Tailpipe Temperatures on the 3B PORV

LIST OF ACRONYMS USED

ALARA	As-Low-As-Is-Reasonably-Achievable
ADAMS	Agencywide Documents Access and Management System
ASME	American Society of Mechanical Engineers
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CHR	Containment Heat Removal
CR	Condition Report
CRD	Control Rod Drive
CV	Containment Venting
EOF	Emergency Operations Facility
FIN	Finding
GE	General Electric
GSEP	General Station Emergency Plan
IMC	Inspection Manual Chapter
IR	Inspection Report
LI	Late Injection
MSIV	Main Steam Isolation Valve
NRC	U. S. Nuclear Regulatory Commission
PARS	Publically Available Records System
PCS	Power Conversion System
PIF	Problem Identification Form
PORV	Power operated relief valve
PRA	Probabilistic Risk Analysis
RCS	Reactor Coolant System
RHR	Residual Heat Removal
SDP	Significance Determination Process
SIL	Service Information Letter
SORV	Stuck-Open Relief Valve
SRA	Senior Reactor Analyst
SSMP	Safe Shutdown Makeup Pump
SW	Service Water
TS	Technical Specifications
TSC	Technical Support Center

LIST OF DOCUMENTS REVIEWED

1R1 Sequence of Events Related to the 3B Power Operated Relief Valve, the Reactor Shutdown and Cooldown, and Event Classification and Notifications

EP-MW-114-100; MWROG Offsite Notifications; Revision 1

QCAN 901(2) - 3 D-13; Relief Valve 1(2) -203-3A and/or 3B is Open; Revision 4

QCGP 2-3; Reactor Scram; Revision 43

QCGP 2-5; Scram Report Data Sheet and Startup Authorization; Revision 16

QCOA 0203-01; Failure of a Relief Valve to Close or Reseat Properly; Revision 8

Condition Report 154275; 2B PORV Inadvertently Opened at Power

QCOS 0203-03; Main Steam Relief Valves Operability Test

QGA 100; RPV Control

QGA 200; Primary Containment Control

QCOP 1000-09; Torus Cooling Startup and Operation; Revision 16

Technical Specifications

Updated Final Safety Analysis Report

EP-AA-115; Recovery from a Classified Event; Revision 1

EP-AA-1006; Radiological Emergency Plan Annex for Quad Cities Station; Revision 16

GE letter GE-NE-0000-0015-6094-BJB1, dated April 16, 2003, "Quad Cities Unit 2 Reactor Vessel Cool-down Rate/Design Basis Evaluations"

1R2 Determination of the Initial Cause of the Stuck Open PORV

CR 154275, Received Annunciator 902-3 D-13 & E-13 ERV Open Annunciator; dated April 16, 2003

CR 154789; B and E PORV Tailpipe Temperature Exceeded 175 Degrees F; dated April 19, 2003

Troubleshooting Plan for the Unit 2 3B PORV 2-203-3B; dated 4/17/2003

GE SIL No. 196, Thermocouple location for Safety/Relief Valve Discharge Lines, Supplement 4; October 31, 1997

Work Order 00410535, Task 01; Rebuild Spare PORV Removed From 2-02; dated April 3, 2002

Work Order 00410535, Task 03; Perform Nitrogen Leak Test; dated April 3, 2002

Work Request 990149020, Task 01; Rebuild PORV S/N 5 Removed From 2-0203-3C During 2R15; dated January 23, 2001

Work Order 990149020, Task 02; Bench Test Spare PORV S/N 5; dated January 2, 2002

Work Order 990149020, Task 03; Perform Nitrogen Leak Test; dated January 8, 2002

AR 00109682; Increasing Contact Dose Rates with Bottom of Torus; dated 28, 2002

AR 00098715; 3B Power Operated Relief Valve Open Spurious Alarm; dated March 11, 2003

LIC-0203.doc; Automatic Depressurization System; April 12, 2001

Vendor Manual Report # 5597; Target Rock Power Operated Relief Valve 93V-001, Addenda 1; dated February 9, 1995

Work History for PORV 0203-3B; dated April 17, 2003

QCOS 0203-03; Main Steam Relief Valves Operability Test; Revision 19

0250-01; Main Steam System; Revision 1

2DW3; U2 Drywell 2nd Level 614 elevation; Revision 11

PIF Q1995-02194; Apparent Leakage Through the New PORVs; dated August 9, 1995

PIF Q1997-03537; Elevated Tailpipe Temperatures; dated June 10, 1997

PIF Q1997-00642; At Time of Reactor Scram 2-203-3C was Reading Approximately 211F and 2-203-3E was Reading 235F; dated February 28, 1997

GE Letter DRF 0000-0015-6094; Strap-on Thermocouples For Relief Valve Leak Detection; dated April 22, 2003

PORC Presentation-Quad Cities Unit 2 Scram Due to Inadvertent 3B PORV Opening; dated April 18, 2003

Event Response Team Charter for 2-0203-3B PORV Spuriously Opened and Stayed Open; dated April 18, 2003

Operability Evaluation 154789-08; PORVs 2-203-3B and 2-203-3E; Revision 0

QCOS 0203-02; Safety and Relief Valve Temperature Surveillances; Revision 12 TIC

Work Order 00323760; Main Steam Safety Valves Operability; dated March 4, 2002

Work Order 00417608; OAC Change SER Time Setting on PORVs; dated March 28, 2002