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February 21, 2005 PY-CEI/NRR-2863L

United States Nuclear Regulatory Commission Document Control Desk Washington, DC 20555

Perry Nuclear Power Plant Docket No. 50-440 LER 2004-002-00 and 2005-001-00

Ladies and Gentlemen:

Attached are Licensee Event Reports (LER) 2004-002-00, "Unplanned Automatic Oscillation Power Range Monitor SCRAM" and 2005-001-00, "Manual Reactor SCRAM Following Unexpected Reactor Recirculation Pump Trip." Any actions discussed within these LERs are described for information only and are not regulatory commitments.

If you have questions or require additional information, please contact Mr. Jeffrey J Lausberg, Manager - Regulatory Compliance, at (440) 280-5940.

Very truly yours,

Attachments

cc: NRC Project Manager NRC Resident Inspector NRC Region III



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U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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17. NARRATIVE (If more space is required, use additional copies of NRC Form 366A)

I. INTRODUCTION

On December 23, 2004, at 2345 hours, both reactor recirculation system (RRC) pumps [AD] at the Perry Nuclear Power Plant (PNPP) unexpectedly downshifted from fast to slow speed. Prior to the RRC pump speed downshift, the plant was stable in Operational Condition 1 at 100 percent rated thermal power. The reactor pressure vessel (RPV) was at 1020 psig and saturated conditions. Following the downshift, at 2354 hours a reactor scram occurred due to core oscillations being detected by the Oscillation Power Range Monitor (OPRM)[JC]. At the time of the scram, the plant was at approximately 55 percent rated thermal power. All control rods fully inserted as a result of the scram signal.

The primary purpose of the RRC system is to provide forced circulation through the reactor core to achieve full power operation and permit variations in power level without control rod movement. Control interlocks are provided for RRC pumps to automatically downshift the pump from fast to slow speed. These controls are provided to prevent cavitation in RRC system components and mitigate the effects of various operational transients on reactor water level and reactivity. The OPRMs are designed to detect reactor core power oscillations and suppress the oscillations by providing a trip signal to the reactor protection system, which results in a reactor scram.

On December 24, 2004, at 0307 hours, the required non-emergency four-hour notification was made to the NRC pursuant to the requirements of 10CFR50.72(b)(2)(iv)(B), reactor protection system actuation while critical (NRC Event Number 41290). This event is being reported under the requirements of 10CFR50.73(a)(2)(iv), any event or condition that resulted in manual or automatic actuation of any of the specified systems.

II. EVENT DESCRIPTION

At 2345 hours on December 23, 2004, both RRC pumps A and B unexpectedly transferred from fast to slow speed. At the time of RRC pump speed downshift, the plant was stable in Operational Condition 1 at 100 percent rated thermal power. After the downshift, reactor power stabilized at about 44 percent and then gradually increased to about 55 percent rated thermal power. The power and flow reduction placed the plant in the immediate exit region of the power-flow map. In this region, the reactor core is susceptible to power oscillations. At 2346 hours, off-normal instruction, "Unplanned Change in Reactor Power or Reactivity," was entered. At 2348 hours the first OPRM alarm was annunciated and subsequently cleared. The magnitude of the core power oscillations detected by the OPRM instrumentation was not observable to operators monitoring reactor power on the control room instrumentation. The OPRM alarm came in three additional times at 2351, 2352 and 2354 hours also with no observable power oscillations.

At 2354 hours a reactor scram occurred due to OPRM trip signals. All control rods fully inserted. The operators properly entered, off-normal instruction, "Reactor SCRAM". The operators also correctly entered plant emergency instruction, "Reactor Pressure Vessel Control" when the reactor vessel level momentarily decreased as expected below Level 3 (178 inches above the top of active fuel) due to void collapse. Level was restored by the

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1. FACILITY NAME	2. DOCKET	6. LER NUMBER	3. PAGE
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AT NADDATNE (15 mere analis manined use addition		2004 002 00	
 17. NARRATIVE (If more space is required, use addition feedwater control system operating in increase to about 230 inches at which Subsequent level control was with the All control rods inserted, no safety reoccurred, and no ECCS system was (RWCU) system [CE] was not available RWCU pumps have non-safety elected during the scram and caused the RW Pressure control was maintained by the valves and their control system for the signal from OPRM Channels A/E, B/F setpoint for Channels D/H was not exit that the OPRMs response during the scram signal was appropriate for the Although the operating crew was determing appropriate mitigating strategies (inset the region of potential instability. The by organizational and programmatic i resulting from a lack of rigor in the material operable. 	al copies of NRC F n automatic m h time the fee e motor driver lief valves wer used for level ole due to the rical power su /CU pumps to the turbine by unctioned as a spond and we nined to be a F and C/G folk ceeded. The event was pe plant condition ermined to har etting control n edelay in inse ssues, includi anagement of	2004 002 00 orm 366A) ode. Reactor water level contin dwater pump turbines tripped as n feedwater pump. re opened, no automatic ECCS control. The reactor water clea RWCU trip during the scram tra pplies that underwent a voltage trip. bass valves [SB-V] throughout th designed. The safety relief valver re not opened manually. Reactor Protection System (RPS bwing the RRC pump downshift. data review from the OPRMs der r design with no deficiencies norms. we performed no inappropriate a rods) were not adequate to prom- rting control rods in this event w ng inadequate training and prod- changes when the OPRM was normalised.	ued to a designed. response nup nsient. The transient he transient. es [SB-RV] S) scram The termined ted. The ctions, nptly exit as caused edures made
The engineering change process (EC implemented, but the change in the o years for technical reasons. When th Operable" status, the ECP process w process mechanism that re-initiated t	CP) was follow perable status le change was as already co he assessme	ed when the OPRM modification s of the system was delayed for s finally made to transition to the mpleted and therefore there was not of training or procedures.	n was over 3 "OPRMs s no
Because of this, the appropriate train were not provided with adequate train more timely actions that could have p experienced on December 23, 2004.	ing needs ana ning or proced prevented an a	lysis was not performed and the ures that would have provided t automatic scram under the cond	e operators he basis for tions
Troubleshooting of the RRC system p These investigations focused on the I Three of the five RRC pump downshi scram would have initiated simultane	oump speed d RRC pump fas ft signals were ously with the	ownshift was initiated to identify st to slow speed downshift logic e eliminated as a potential cause downshift from those signals.	the cause. circuitry. e since a
Since the other sources of pump dow Feedwater Flow and Low Reactor Lev	nshift signals vel were repla	were eliminated, the alarm card ced prior to plant restart. Thes	s for Low e cards
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 17. NARRATIVE (If more space is required, use addition were considered the highest probabil historical failure trends. Since the Rf definitively be confirmed, temporary r input voltage to the two replaced alar Subsequent to this event, on January downshifted to slow speed. The causisolator intermittent failure as a result of-cycle reactor recirculation pump tri (Reference PNPP LER 2005-001 for determined to be the reason for the F IV. EVENT ANALYSIS The primary purpose of the RRC syst core to achieve full power operation a movement. Control interlocks are propump from fast to slow speed. These system components and mitigate the Analysis of plant conditions at the tim confirmed that no process parameter were present. With the reactor at ful into the region of potential instability of stabilized at approximately 55 percentrip setpoints of the OPRMS. Evaluating safety limit was not exceeded. The plant response was consistent wi (USAR) event analyses for "Decreases shows that a downshift of both RRC gevent. The plant response following the turbine trip. Thus, this event was det A risk assessment was also performed core damage for the December 23, 27 early release was computed to be 4.6 that 1.0E-06 and a large early releaser risk significant events. 	al copies of NRC F lity to have can RC pump dow recorders were m cards and c / 6, 2005, the se of the dowr of an inadequ ip (EOC/RPT) further informat RC pump dow tem is to provi- and permit var ovided for the e controls are effects of vari- the of the RRC s or transients Il power, the R of the power to the of the RRC s or transients Il power, the R of the power to the following the a scram and t ith and bound e in Reactor R oumps event is he scram was ermined to be ed. The compo 004 scram was E-09. Transie e probability le	orm 366A) used the nshift sige installed other poi RRC put nshift wa iate surg control of ation). To within to de force iations in RRC put provided ous ope pump fa require RC pump fa require RC pump fa require RC pump fa scramm RM ever e RRC p he minin ed by the sounde within d uted rest s 3.7E-O ents with ss than	RR gnal d to ints i mps s de su circu fhis on D d circu mps f to p mps d to o mps f to p num e Up tion b ults i a circu int bu ump num e sig e sig ults i a circu int bu ump f a circu int bu unts i f a circu int bu int int bu int int bu int int bu int int int int int int int int int int int int int int int int	C pump was inti- monito n the R again L termine uppress it for th cause v ecembe culatior wer leve to auto prevent nal trar slow s initiate power f as a res offer dat downs critical bdated S Flow" in y the tri USAR n evalue were that of the p ore dan	o dov ermit r the RC o inex ed to ie RF ed to ie RF er 23 through the as is a con- hift. pow Safet a con- hift. pow Safet a con- hift. pow suit o r con- hift. r con- hi	vnshift ba ttent and power s circuitry. pectedly be an op- network i C pump- also subs , 2004. bugh the thout cor- cally dow tation in ts. I downsh- above int ught the ving the of exceed ncluded The OP- er ratio (y Analys- apter 15. both RR ion 15.2 I limits e probabi- bility of a probabi- consider w safety	ased on could not supply and otical n the end- s sequently reactor ntrol rod wnshift the RRC hift erlocks reactor downshift ding the that RMs did MCPR) sis Report 3 which C pumps .3 for a hillity of a large lity less red to be
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NRC FORM 366A (1-2001)	U.S. NUCLEAR REGULATORY COMMISS								
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17. NARRATIVE (If more space is required, use	additional copies of NRC F	orm 366A)							

Interim action to address inadequate training was completed prior to restart including training on the significance and pattern of the OPRM alarms and the urgency for exiting the immediate exit region of the power to flow map.

Following the January 6, 2005 RRC pump downshift, the failed optical isolator was replaced and a diode was installed across the downstream logic trip relay for surge suppression. Similar optical isolators, that could have been damaged by relay inductive surges, were also replaced.

Additional corrective actions that have been completed or are in progress:

- 1. Off-normal operating instruction, ONI-C51 "Unplanned Change in Reactor Power or Reactivity" was revised so that the operators can go immediately to the required actions.
- 2. The alarm response instruction was revised to indicate that a reactor scram was possible and to evaluate the need to enter off-normal operating instruction, ONI-C51 "Unplanned Change in Reactor Power or Reactivity.
- 3. Simulator training time will be reviewed to increase off-normal event training including events that have a higher probability to occur.
- 4. Procedures will be revised to consider training needs when incorporating changes to operational methods and to ensure that training precedes significant changes in operational methods.
- 5. Analysis of industry BWR power oscillation events will be performed and incorporated into simulator training sessions.

The above actions have been entered in the corrective action program.

VI. SIMILAR EVENTS

Manual Scram due to unexpected Reactor Recirculation Pump downshift, LER 1993-015. This manual scram was initiated in accordance with procedures in effect at that time when entering the power-to-flow region of core instability. The cause of the downshift was a failure of both RRP suction resistance thermal detectors, which is not similar to the December 23, 2004 event. Optical isolators were specifically evaluated as not being the cause.

Manual scram due to unexpected Reactor Recirculation Pump downshift, LER 1994-002. This manual scram was initiated in accordance with procedures in effect at that time when entering the power-to-flow region of core instability. The cause of the downshift was a failure of a K1 relay on an alarm card, which is not similar to the December 23, 2004 event.

Automatic scram following unexpected Reactor Recirculation Pump downshift, LER 2001-005.

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U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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Perry Nuclear Power Plant	05000440	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	0.40
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17. NARRATIVE (If more space is required, use additional copies of NRC Form 366A)

I. INTRODUCTION

On January 6, 2005, at 0106 hours, while operating at approximately 100 percent rated thermal power with stable conditions, the Perry Nuclear Power Plant (PNPP) reactor recirculation system (RRC) [AD] pumps A and B unexpectedly downshifted from fast to slow speed. Reactor power decreased to approximately 44 percent power. This power and flow reduction placed the plant in the immediate exit region of the power-flow map. A similar unexpected downshift from fast to slow speed of both RRC pumps occurred on December 23, 2004, which was reported in PNPP Licensee Event Report (LER) 2004-002.

At 0110 hours, while operators were inserting control rods in accordance with off normal operating instruction ONI-C51, "Unplanned Change in Reactor Power or Reactivity", RRC pump A unexpectedly tripped from slow speed to off. At 0112 hours, a manual reactor scram was initiated by the control room crew. All control rods [AA] fully inserted.

The RRC system consists of two independent piping loops, each with a two speed recirculation pump, a flow control valve (FCV), and associated piping and instrumentation external to the vessel, and jet pumps internal to the vessel. The primary purpose of the RRC system is to provide forced circulation through the reactor core to achieve full power operation and permit variations in power level without control rod movement. Control interlocks are provided for the RRC pumps to automatically downshift the pump from fast to slow speed.

II. EVENT DESCRIPTION

On January 6, 2005, at 0106 hours, while operating at approximately 100 percent power, the Perry Nuclear Power Plant (PNPP) RRC pumps A and B unexpectedly downshifted from fast to slow speed. Reactor power decreased to approximately 44 percent power. This power and flow reduction placed the plant in the immediate exit region of the power-flow map. In this region, the reactor core is susceptible to power oscillations due to higher power relative to core flow. Control room operators entered off normal operating instruction ONI-C51, "Unplanned Change in Reactor Power or Reactivity" and began to insert the specified control rods.

At 0110 hours, while operators were inserting control rods, RRC pump A unexpectedly tripped from slow speed to off, resulting in single loop RRC operation. At 0112 hours, the reactor operator, with the concurrence of the control room Senior Reactor Operator (SRO), inserted a manual reactor scram, based upon conservative plant operations by the control room crew. All control rods fully inserted.

As anticipated, reactor water level decreased to Level 3 (178 inches above top of active fuel) immediately following the scram. Level was restored by the feedwater system [SJ] operating in automatic on the master level controller. Reactor level increased to Level 8 (219 inches above top of active fuel). At Level 8, the non-safety related reactor feedwater pump turbines tripped as designed to terminate feedwater addition to the reactor. Level continued to increase due to control rod drive hydraulic water addition and swell until level reached 226 inches.

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At 0114 hours, the turbine generator was manually tripped. At 0119 hours, following reactor level reduction below Level 8, operators attempted to manually start the non-safety related motor feedwater pump (MFP) [SJ-P] to control level. The pump failed to start when taken to start following two attempts. At 0122 hours, a reactor core isolation cooling (RCIC) pump [BN] was started for injection to the reactor vessel and continued to provide reactor level control.

Pressure control was maintained by the turbine bypass valves [SB-V] until 0210 hours, when the main steam isolation valves (MSIVs) were closed to limit reactor vessel cooldown. Once the MSIVs were closed, the RCIC System, which was being used for level control, was also utilized for pressure control. The safety relief valves [SB-RV], were not required to automatically respond and were not opened manually. No automatic Emergency Core Cooling System (ECCS) response was required.

There were no other structures, systems or components that were inoperable at the start of the event (other than the MFP as described above) that contributed to the event.

On January 6, 2005, at 0401 hours, the required non-emergency four-hour notification was made to the NRC pursuant to the requirements of 10 CFR 50.72(b)(2)(iv)(B), reactor protection system actuation while critical (NRC Event Number 41310). This event is being reported under requirements of 10 CFR 50.73(a)(2)(iv), any event or condition that resulted in manual or automatic actuation of any of the specified systems.

III. CAUSE OF EVENT

The cause of the reactor scram was a manual initiation by the reactor operator. The operator actions to initiate this manual scram were appropriate considering the plant conditions and within management expectations.

Troubleshooting of the RRC pump speed downshift instrument loops was initiated to identify potential component failures. A similar unexpected RRC downshift event occurred approximately two weeks earlier on December 23, 2004 [See PNPP LER 2004-002].

It was determined that the downshift originated from an optical isolator in the end-of-cycle RRC pump trip (EOC/RPT) circuitry for the B logic. An optical isolator is used for physical separation of safety and non-safety circuits.

The EOC/RPT trip is a safety supplement to the reactor protection system [JC], which will automatically initiate a RRC pump downshift to slow speed after a main turbine trip. A relay was found to have 40 VDC across its coil supplied from an optical isolator output, when voltage should have been 0 VDC. The sequence of events recorded during both the December 23, 2004 and the January 6, 2005 RRC high to slow speed downshift events could be duplicated during testing with 40 VDC being applied from the output of the subject optical isolator. The failure mechanism for the optical isolator was overvoltage stress, which caused the output transistor on the optical isolator to fail. The installed resistor/capacitor surge suppression network was insufficient to protect the output transistors from the inductive kick experienced during normal operation, thus providing the source for electrically overstressing

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17. NARRATIVE (If more space is required, use addition of the optical isolator card. This inad sensitive to normal high level electric malfunctioning output.	<i>al copies of NRC F</i> equate suppre al fast transie	orm 366A) ession design made the nt noise, which allowed	e optical is an interm	olator hittent
The cause of the RRC A tripping from the voltage regulator card [EC] in the electrical power to the RRC pump A a voltage from the LFMG A generator f seconds later. The ensuing attempt of resulted in a generator over-current re breaker.	n slow speed low frequency at slow speed ollowed by a r of the LFMG A elay trip and lo	to off was a failure of an y motor generator (LFN . The failure caused a apid recovery to near r A to restore voltage to the bock out the LFMG A ge	n amplifier IG) which momentar ated volta ne RRC p nerator ou	r circuit on supplies rily loss of ge seven ump motor utput
The cause of the motor feedwater pu 15 KV electrical supply breaker not be charged, it is not possible to close the The breaker's charging springs were striking the cubicle floor-mounted inter within 15 KV non-safety related 15 HI interference plate, which was in its de a different electrical current rating car misalignment of the racking rails cont the interference plate. The physical of mechanism and can intermittently res to actuate the control device and ene design deficiency.	mp failure to s eing charged. breaker and not charged c erference plate K ABB electric esign location, n not be instal ributed to the contact dissipa sult in the brea rgize the closi	start was due to the close With the breaker close start the pump from the lue to the closing spring a (also known as a reject cal supply breaker cubic is positioned to ensure led in its cubicle. Addit contact of the spring cl ates some kinetic energy let timing cam failing to ng spring charging mot	sing spring ng spring e control s g charging ction plate cles. This e that a bra ionally, narging lin y to the o o rotate s tor. This i	gs in the s not switch. g link arm e) used eaker with eaker with perating ufficiently s a legacy
A lack of full organizational commitme Solving and Decision Making Process cause of the December 23, 2004 RR the January 6, 2005 RRC pump dowr	ent for the pro s contributed t C pump down nshifting even	gram implementation o o the ineffectiveness in shifting event and is a o t.	f the Prob determin contributin	lem ing the ig cause of
IV. EVENT ANALYSIS				
The primary purpose of the RRC syst core to achieve full power operation a movement. Control interlocks are pro pump from fast to slow speed. These system components and mitigate the	em is to provi and permit var ovided for the e controls are effects of vari	de forced circulation th iations in power level w RRC pumps to automa provided to prevent cav ous operational transie	rough the rithout cor tically dov ritation in nts.	reactor htrol rod vnshift the RRC
Analysis of plant conditions at the tim confirmed that no process parameters were present. With the reactor at full a region of potential instability on the observed. Actual power after the dow percent. Following off normal operati control rods. However, the reactor wa decision by the control room crew give	e of the RRC s or transients power, the pu power-flow m vnshift transie ng instruction as subsequen en both the ui	pump fast to slow spee requiring initiation of the mp speed downshift pl ap, although no power nt was determined to b s, the control room crew thy manually scrammed mexpected downshift of	d downsh ne above aced the oscillation e approxin v began ir l as a con both RRC	ift interlocks reactor in is were mately 44 nserting servative C pumps

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The plant response was consistent w (USAR) event analyses for "Decrease shows that a downshift of both RRC p event. The plant response was also for "Recirculation System Single Loop The recirculation System single loop of operate with a single recirculation loo rated thermal power. The plant respondent Section 15.2.3 for a turbine trip. Thus evaluation limits. A risk assessment was also performed core damage for the January 6, 2005 release was computed to be 4.9E-09. 1.0E-06 and a large early release pro- significant events.	ith and bound in Reactor R pumps event i consistent with o Operation" in operation eval p out-of-servic onse following s, this event w ed. The comp , scram was 8 . Transients v obability less th	ed by th eccircula s bounds n and bo n Chapte uation in ce at up the mar as deter uted res .7E-07 a vith a co nan 1.0E	e Uj tion ed b bund er 15 ndica to a nual rmin ults and - re d. 2-07	pdated Safe Flow" in Ch by the trip of led by the U 5.0.5.3 and i ates that PN pproximatel scram was ed to be wit were that th the probabil amage prot are not con	ety Analys hapter 15. f both RR ISAR even in Append IPP can s ly 67 perc bounded thin desig he probab lity of a la bability les hisidered t	sis Report 3 which C pumps ent analysis dix 15F. safely cent of by USAR in bility of irge early ss that o be risk
Based upon the above information, the vertice of the second secon	is event is ve	ry low sa	afety	v significanc	æ.	
The failed optical isolator in the RRC place of the resistor/capacitor surge s that had been subjected to similar con modified cards.	logic was repl suppression ne nditions as the	aced wil etwork. e failed c	th a Add ard	modified ca itional optic were replac	ard using al isolato ced with t	diodes in r cards he
The LFMG A voltage regulator was re degrading LFMG voltage regulator tre	placed. Actio ands prior to e	ns were quipmer	initi nt fai	iated to ider ilure.	ntify and o	correct
The interference plate was removed f electrical supply cubicle. Interference also removed.	rom the floor (plates in othe	of the mo er 15 KV	otor ' bre	feedwater p akers that e	oump's 1 exhibited	5 KV wear were
Training will be provided to the appro Decision Making process "NOP-ER-3 management's expectations and the is implemented as written.	priate site per 001", their role standards req	sonnel o es and re uired to l	n th espo be fo	e Problem S onsibilities in ollowed to e	Solving a n the proc ensure the	nd cess, e process
The above events and corrective action program.	ons have beer	n entered	d in	the plant co	prrective a	action

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VI. PREVIOUS SIMILAR EVENTS

Manual Scram due to unexpected Reactor Recirculation Pump downshift, LER 1993-015. This manual scram was initiated in accordance with procedures in effect at that time when entering the power-to-flow region of core instability. The cause of the downshift was a failure of both RRC pump suction resistance thermal detectors, which is not similar to the January 6, 2005 event. Optical isolators were specifically evaluated as not being the cause.

Manual Scram due to unexpected Reactor Recirculation Pump downshift, LER 1994-002. This manual scram was initiated in accordance with procedures in effect at that time when entering the power-to-flow region of core instability. The cause of the downshift was a failure of a K1 relay on an alarm card, which is not similar to the January 6, 2005 event.

Automatic Scram following unexpected Reactor Recirculation Pump downshift, LER 2001-005. This automatic scram occurred following a high reactor pressure vessel water level that was initiated by a RRC pump downshift. The cause of the downshift was a failure of a feedwater system level summer card, which is not similar to the January 6, 2005 event.

The corrective actions from the above events would not have precluded occurrence of this event.

Automatic Scram following unexpected Reactor Recirculation Pump downshift, LER 2004-002. This automatic scram occurred due to an oscillation power range monitor (OPRM) actuation. The cause of the downshift was thought to be a failed low feedwater flow or low reactor water level sensing card. The subsequent pump downshift, which occurred on January 6, 2005, was determined to be due to a failed optical isolator, which was also subsequently determined to be the cause for the prior pump downshift on December 23, 2004. Although the cause of the downshift was the same, the corrective actions for the OPRM scram were judged effective since the correct operator actions were in progress to exit the immediate exit region of the power to flow map. No OPRM alarms were received indicating that an automatic scram was unlikely to have reoccurred. The RRC pump tripped to off, causing the control room crew to manually scram the reactor in this event.

Energy Industry Identification System Codes are identified in the text as [XX].