March 8, 2007

Mr. J. Conway
Site Vice President
Monticello Nuclear Generating Plant
Nuclear Management Company, LLC
2807 West County Road 75
Monticello, MN 55362-9637

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT
NRC INITIAL LICENSE EXAMINATION REPORT 05000263/2007301(DRS)

Dear Mr. Conway:

On February 19, 2007, the Nuclear Regulatory Commission (NRC) completed initial operator licensing examinations at your Monticello Nuclear Generating Plant. The enclosed report presents the results of the examination which were discussed on February 16 and March 1, 2007, with you and Mr. Earl, respectively, and with other members of your staff.

The NRC examiners administered initial license examination operating tests during the week of February 12, 2007. Members of the Monticello Nuclear Generating Plant Training Department administered an initial license written examination on February 19, 2007, to the applicants. Four senior reactor operator and four reactor operator applicants were administered license examinations. The results of the examinations were finalized on March 6, 2007. Six applicants passed all sections of their examinations resulting in the issuance of two senior reactor operator and four reactor operator licenses. Two applicants failed the written examination and will not be issued licenses. The applicants who failed the NRC examination were issued proposed license denial letters.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (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).

We will gladly discuss any questions you have concerning this examination.

Sincerely,

/RA/

Hironori Peterson, Chief
Operations Branch
Division of Reactor Safety

Docket No. 50-263
License No. DPR-22

Enclosures:
1. Operator Licensing Examination Report 05000263/2007301(DRS)
2. Simulation Facility Report
3. Post Examination Comments and Resolutions
4. Written Examinations and Answer Keys (RO/SRO)

cc w/encls 1 and 2:
M. Sellman, President and Chief Executive Officer
Manager, Nuclear Safety Assessment
J. Rogoff, Vice President, Counsel, and Secretary
Nuclear Asset Manager, Xcel Energy, Inc.
State Liaison Officer, Minnesota Department of Health
R. Nelson, President
Minnesota Environmental Control Citizens Association (MECCA)
Commissioner, Minnesota Pollution Control Agency
D. Gruber, Auditor/Treasurer,
Wright County Government Center
Commissioner, Minnesota Department of Commerce
Manager - Environmental Protection Division
Minnesota Attorney General's Office

cc w/encls 1, 2, 3, and 4: S. Halbert, Training Manager

We will gladly discuss any questions you have concerning this examination.

Sincerely,

/RA/

Hironori Peterson, Chief
Operations Branch
Division of Reactor Safety

Docket No. 50-263
License No. DPR-22

Enclosures:
1. Operator Licensing Examination Report 05000263/2007301(DRS)
2. Simulation Facility Report
3. Post Examination Comments and Resolutions
4. Written Examinations and Answer Keys (RO/SRO)

cc w/encls 1 and 2:
M. Sellman, President and Chief Executive Officer
Manager, Nuclear Safety Assessment
J. Rogoff, Vice President, Counsel, and Secretary
Nuclear Asset Manager, Xcel Energy, Inc.
State Liaison Officer, Minnesota Department of Health
R. Nelson, President
Minnesota Environmental Control Citizens Association (MECCA)
Commissioner, Minnesota Pollution Control Agency
D. Gruber, Auditor/Treasurer,
Wright County Government Center
Commissioner, Minnesota Department of Commerce
Manager - Environmental Protection Division
Minnesota Attorney General's Office

cc w/encls 1, 2, 3, and 4: S. Halbert, Training Manager

DOCUMENT NAME: C:\FileNet\ML070680188.wpd
☐ Publicly Available ☐ Non-Publicly Available ☐ Sensitive ☐ Non-Sensitive
To receive a copy of this document, indicate in the concurrence box "C" = Copy without attach/encl "E" = Copy with attach/encl "N" = No copy

OFFICE	RIII		RIII					
NAME	NValos:co		HPeterson					
DATE	03/07/07		03/08/07					

OFFICIAL RECORD COPY

DISTRIBUTION:
TEB
PST
RidsNrrDirsIrib
GEG
KGO
GLS
CST1
CAA1
LSL (electronic IR's only)
C. Pederson, DRS (hard copy - IR's only)
DRPIII
DRSIII
PLB1
TXN
ROPreports@nrc.gov

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-263
License No:	DPR-22
Report No:	05000263/2007301(DRS);
Licensee:	Nuclear Management Company, LLC
Facility:	Monticello Nuclear Generating Plant
Location:	Monticello, Minnesota
Dates:	February 6 through February 19, 2007
Examiners:	N. Valos, Chief Examiner B. Palagi, Examiner C. Zoia, Examiner
Approved by:	Hironori Peterson, Chief Operations Branch Division of Reactor Safety

Enclosure 1

SUMMARY OF FINDINGS

ER 05000263/2007301(DRS); 02/12/2007 - 02/19/2007; Nuclear Management Company, LLC; Monticello Nuclear Generating Plant; Initial License Examination Report.

The announced initial operator licensing examination was conducted by regional NRC examiners in accordance with the guidance of NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9.

Examination Summary:

- Eight examinations were administered (four senior reactor operator and four reactor operator).
- Six applicants passed all sections of their examinations resulting in the issuance of two senior reactor operator and four reactor operator licenses.
- Two applicants failed the written examination and will not be issued licenses. The applicants who failed the NRC examination were issued proposed license denial letters.

REPORT DETAILS

4. OTHER ACTIVITIES (OA)

4OA5 Other

.1 Initial Licensing Examinations

a. Inspection Scope

The NRC examiners conducted an announced initial operator licensing examination during the week of February 12, 2007. The facility licensee's training staff used the guidance prescribed in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9, to prepare the outline and develop the written examination and operating test. The examiners administered the operating test, consisting of job performance measures and dynamic simulator scenarios, during the period of February 12 through February 16, 2007. The facility licensee administered the written examination on February 19, 2007. Four senior reactor operator and four reactor operator applicants were examined.

b. Findings

Written Examination

The NRC examiners determined that the written examination, as originally submitted by the licensee, was within the range of acceptability expected for a proposed examination. All changes made to the submitted examination were made in accordance with NUREG-1021, "Operator Licensing Examination Standards for Power Reactors.

A total of ten post-examination comments (8 RO; 2 SRO comments) were submitted by the applicants to the facility training department. One of the post-examination comments was associated with a clarification made to a question by the facility during the administration of the examination. Of the ten post-examination comments, the facility agreed with two of the comments. The post-examination comments were submitted to the NRC on February 26, 2007. The results of the NRC's review of the comments are documented in Enclosure 3, Post Examination Comments and Resolutions.

Operating Test

The NRC examiners determined that the operating test, as originally submitted by the licensee, was within the range of acceptability expected for a proposed examination. All changes made to the submitted examination were made in accordance with NUREG-1021, "Operator Licensing Examination Standards for Power Reactors."

Examination Results

Six applicants passed all sections of their examinations resulting in the issuance of two senior reactor operator and four reactor operator licenses. Two applicants failed the

written examination and will not be issued licenses. The applicants who failed the NRC examination were issued proposed license denial letters.

.2 Examination Security

a. Inspection Scope

The NRC examiners briefed the facility contact on the NRC's requirements and guidelines related to examination physical security (e.g., access restrictions and simulator considerations) and integrity in accordance with 10 CFR 55.49, "Integrity of Examinations and Tests," and NUREG-1021, "Operator Licensing Examination Standard for Power Reactors." The examiners reviewed and observed the licensee's implementation and controls of examination security and integrity measures (e.g., security agreements) throughout the examination process.

b. Findings

The licensee's implementation of examination security requirements during examination preparation and administration were acceptable and met the guidelines provided in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors." No violations of 10 CFR 55.49 occurred during the examination preparation and administration.

4OA6 Meetings

Exit Meeting

The chief examiner presented the examination team's preliminary observations and findings with Mr. J. Conway, Site Vice President, and other members of the licensee management on February 16, 2007. A subsequent exit via teleconference was held on March 1, 2007, with Mr. J. Earl, General Supervisor Operations Training, following review of the site post-examination comments. No proprietary items were identified during the administration of the examination nor during the exit meeting with the licensee. The licensee acknowledged the observations and findings presented.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
J. Conway, Site Vice President
B. MacKissock, Operations Manager
S. Halbert, Training Manager
J. Earl, General Supervisor Operations Training
G. Allex, Supervisor Operations Training - Continuing
O. Olson, Supervisor Operations Training - Initial
J. Ruth, Operations Training

NRC
N. Valos, Chief Examiner
B. Palagi, Examiner
C. Zoia, Examiner
S. Thomas, Senior Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Closed

None

Discussed

None

LIST OF ACRONYMS

ADAMS	Agency-Wide Document Access and Management System
CFR	Code of Federal Regulations
CR	Condition Report
DRS	Division of Reactor Safety
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records System
RO	Reactor Operator
SRO	Senior Reactor Operator

SIMULATION FACILITY REPORT

Facility Licensee: Monticello Nuclear Generating Plant

Facility Licensee Docket No.: 50-263

Operating Tests Administered: February 12 through February 16, 2007

The following documents observations made by the NRC examination team during the initial operator license examination. These observations do not constitute audit or inspection findings and are not, without further verification and review, indicative of non-compliance with 10 CFR 55.45(b). These observations do not affect NRC certification or approval of the simulation facility other than to provide information which may be used in future evaluations. No licensee action is required in response to these observations.

During the conduct of the simulator portion of the operating tests, the following items were observed:

ITEM	DESCRIPTION
1	There was one simulator exam scenario delay of approximately 30 minutes on the morning of February 14, 2007, until three Process Computer screens could be restored in the simulator. CAP01077256 was written associated with the issue.

Post Examination Comments and Resolutions

Question Number 4

A LOCA is in progress. Both Core Spray pumps are injecting at 3000 gpm each to maintain RPV water level above the top of active fuel. Which of the below listed plant parameters may be an indication of pump cavitation in the Core Spray pump(s)?

A. Steadily lowering of Core Spray pump amps

B. Steadily lowering of Core Spray discharge pressure

C. Annunciator 3-A-29 (CORE SPRAY 1 PRESS VLV LEAKING) in alarm

D. Repeated alarming and subsequent clearing of annunciator 3-A-41(AC INTERLOCK)

Answer: D

Applicant Comment:

An applicant commented that answer "A" should also be accepted as correct.

Answer "A" should also be accepted. The discharge head of the Core Spray (CS) pumps is approximately 300 psig. The setpoint for the annunciator alarm "AC Interlock" is 100 psig. The stem of the question states both CS pumps are in service. Both CS pumps must be below 100 psig to clear the alarm. Even with severe cavitation, the applicant does not believe that both CS pumps would drop in discharge pressure at the same time to clear the alarm. Even though the Bases for C.4-B.04.01G, "ECCS Suction Control During LOCA" stated that the alarm may be an indication of ECCS suction plugging, this statement assumed only one CS pump was running. That was NOT the case in the question stem. Step 1.f of C.4-B.04.01G stated that the pump motor amperage would be erratic or decreasing for plugging strainer. Therefore, answer "A" is the most correct answer.

Facility Proposed Resolution:

The question grading for the exam should not change. The effects of cavitation include fluctuations in discharge pressure and motor current. This is supported in Lesson Plan M-8120L-114, "Fluid Statics and Dynamics," which was part of the Initial License Training (ILT) Generic Fundamentals Course. Answer "A", "Steadily lowering of Core Spray pump amps," may be an indication of suction plugging, but is not an indication of pump cavitation; the question asks for indication of pump cavitation.

NRC Resolution:

Upon review of the question, the applicant comment, and the facility proposed resolution, it was decided to accept only the original correct answer.

Post Examination Comments and Resolutions

The applicant stated that the Bases for C.4-B.04.01G assumed that only one CS pump was running, when stating that the "AC INTERLOCK" alarm may be an indication of ECCS suction plugging. However, the Bases for C.4-B.04.01G does not stipulate how many ECCS pumps may be running. The applicant stated that pump motor amperage would be erratic or decreasing for plugging strainer. Though this is valid, the question did not ask for an indication of a plugging strainer, the question asked for an indication of pump cavitation in the Core Spray pump(s). The Bases for C.4-B.04.01G stated that one indication of an ECCS suction strainer plugging is: "Erratic and dramatic fluctuations in pump discharge pressure. One alarm that may clue the operator to this condition is the alarming and subsequent clearing of AC INTERLOCK (3-A-41)."

The indications that a pump is cavitating include fluctuations in discharge pressure and motor current. These effects of cavitation are supported in Lesson Plan M-8120L-114, "Fluid Statics and Dynamics," which was part of the Initial License Training Generic Fundamentals Course. Per Lesson Plan M-8120L-114, the indication specified in distractor "A," "Steadily lowering of Core Spray pump amps," is not an indication of cavitation of the CS pumps. Since distractor "D," "Repeated alarming and subsequent clearing of annunciator 3-A-41(AC INTERLOCK)," provided the only indication of pump cavitation of the CS pumps, distractor "D" was retained as the only correct answer.

Post Examination Comments and Resolutions

Question Number 17

During a Reactor startup, when do plant conditions support the design limitations of the Reactor Water Level Control System allowing it to be placed in 3 Element Control?

A. When the first Feedwater Control Valve is placed in automatic.

B. When the Master Feedwater Level Controller is placed in automatic.

C. When the second Feedwater Control Valve is placed in automatic.

D. When feedwater flow is sufficient to clear Reactor Recirc pump low flow interlock.

Answer: C

Applicant Comment:

An applicant commented that answer "D" should also be accepted as correct.

Answer "D" should also be accepted as correct. Ops Manual B.05.07-01, "Reactor Level Control" states that the steam and feedwater flow signals lose their accuracy below 30% power. Also, on page 7 of B.05.07-01, it states that a separate single element control scheme is used when reactor power is less than 20% of rated. The question asked per design limits of the water level control system, when can the Digital Feedwater Control System (DFCS) be placed in 3 element control. The 3 element would work at greater than 20% feed flow and with one Feedwater Fegulating Valve (FRV) in service. Although the start up procedure C.1, " " places the DFCS in 3 element control at approximately 40% power and after the second FRV is in service, the design of DFCS allows it to be placed in service with feedwater greater than 20% per its' design.

Facility Proposed Resolution:

The question grading for the exam should not change. Per Ops Man B.05.07-01, "Steam flow and feedwater flow signals (used in three-element control) lose their accuracy below 30% power and, therefore, become less desirable as controlling inputs". Answer "C" (exam correct answer) is the only choice describing an action above 30% power. The conditions cited by the applicant "with feedwater greater than 20%" are for transition from the Feedwater Low Flow Regulating Valve to the Main Regulating Valves.

NRC Resolution:

Upon review of the question, the applicant comment, and the facility proposed resolution, it was decided to accept only the original correct answer. Ops Manual B.05.07-01, "Reactor Level Control" states that the steam and feedwater flow signals lose their accuracy below 30% power. Both the steam and feedwater flow signals are used when in 3 Element Control.

Post Examination Comments and Resolutions

The design limitations of the Reactor Water Level Control System would not allow placing 3 Element Control in service with inaccurate steam and feedwater flow signals. The applicant also stated that a separate single element control scheme is used when reactor power is less than 20% of rated power. However, this statement is related to control of the Feedwater Low Flow Regulating Valve, not to 3 Element Control of the Main Feedwater Regulating Valves. Thus, distractor "D," "When feedwater flow is sufficient to clear Reactor Recirc pump low flow interlock," which occurs at approximately 20% reactor power would not be a correct answer.

Since, per C.1, "Reactor Startup," at approximately 40% power, the second Main Feedwater Regulating Valve is placed in service, and then Reactor Level Control is transferred to 3 Element Control, distractor "C" was retained as the only correct answer.

Post Examination Comments and Resolutions

Question Number 20

What would be the effect on 4.16 KV breaker operation if all D.C. control power is lost?

Breaker operation with the control switch would be lost to…

A. all 4.16 KV breakers.

B. all 4.16 KV breakers EXCEPT for buses 13 and/or 14.

C. all 4.16 KV breakers EXCEPT for buses 15 and/or 16.

D. all 4.16 KV breakers EXCEPT for buses 17 and/or 18.

Answer: D

Applicant Comment:

An applicant commented that answer "A" should also be accepted as correct.

A clarification was made to question 20 which changed the acceptable answer. Without the clarification, answer "D" would be correct since the 17 and 18 buses use AC control power and are not affected by loss of DC control power. However, the proctor stated to add "**Control Room** Control Switch" to the stem of the question. Since there are no bus 17 or bus 18 control switches in the control room, there would be no control switches that could operate 4 KV breakers in the control room due to the loss of all DC control power. This would make answer "A" the only correct answer. Based upon whether a student already completed the question or did not apply the additional verbal clarification, then both answer "A" and "D" should be accepted.

Facility Proposed Resolution:

The question grading for the exam should be changed to accept both "A" and "D" as correct answers. This is due to the information given during exam implementation, as stated above, that changed the intent of the question. This question is correct, as written, and does not require any changes prior to incorporation into the exam bank.

NRC Resolution:

Upon review of the question, the applicant comment, and the facility proposed resolution, it was decided to accept the facility's comment and accept both answer "A" and "D" as correct answers.

During the administration of the examination, a question was asked by an applicant as to which control switch the question stem was referring to (i.e., the control switch in the Control Room or

at the breaker). The facility provided a clarification that the question stem was asking "Breaker operation with the **Control Room** control switch would be lost to…," with the words underlined added for clarification. This clarification of the question was communicated to all of the applicants. The clarification provided changed the intent of the question. Buses 17 and 18 are in the plant discharge structure, with control power supplied alternating current (AC) from one of two control power transformers. There are no bus 17 or bus 18 control switches in the control room.

In answering the clarified question, since there are no bus 17 or bus 18 control switches in the control room, there would be no control switches that could operate 4.16 KV breakers in the control room due to the loss of all DC control power. This would make distractor "A," "all 4.16 KV breakers," a correct answer. However, in answering the clarified question, distractor "D" is still correct, since there are no control room switches for bus 17/18 breakers. Distractors "A" and "D" thus reduce to the same answer "all 4.16 KV breakers." Therefore, the answer key was modified to accept both "A" and "D" as correct answers.

Post Examination Comments and Resolutions

Question Number 31

A plant transient was in progress with the following conditions present:

- D/W pressure 15 psig and rising
- D/W temperature 285°F and rising
- RPV water level -175 inches and slowly lowering
- RPV pressure 850 psig
- All Control rods are fully inserted
- EOPs 1100 and 1200 have been entered and are being executed
- The SAMGs have not been entered
- Drywell spray is directed to be placed in service

Which of the following predicts the impact on Drywell temperature and why?

A. Drywell temperature would continue to rise; Drywell Spray is NOT ALLOWED as plant conditions are outside the Drywell Spray Limit curve.

B. Drywell temperature would lower; as Drywell Spray IS ALLOWED to be initiated by placing the Containment Spray/Cooling LPCI Initiation switch in Bypass and opening the Drywell Spray inboard and outboard isolation valves.

C. Drywell temperature would continue to rise since neither Drywell Spray inboard or outboard isolation valve is allowed to be opened under these plant conditions.

D. Drywell temperature would lower as drywell spray is allowed to be initiated by placing both the Containment Spray 2/3 Core Height Bypass and Containment Spray/Cooling LPCI Initiation switches in BYPASS, and then opening the Drywell Spray inboard and outboard isolation valves.

Answer: C

Applicant Comment:

An applicant commented that answer “D” should also be accepted as correct.

Answer “D” should also be accepted. The RHR logic allows drywell sprays to be placed in service after bypassing the 2/3 core height interlock and the Containment Spray(CS)/Cooling Low Pressure Coolant Injection (LPCI) initiation switch. Although the Drywell (DW) spray procedure (Part C of procedure Ops Man C.5-3502, “Drywell Spray - RHR-A”) stated that this should be done when in the Severe Accident Management Guidelines (SAMGs), the logic would allow answer “D” to be correct. The applicant assumed other pumps (like High Pressure Coolant Injection and CS) were available to recover level. The basis did state DW spray and injection can be alternated to provided adequate cooling. Therefore, answer “D” also correctly answered the question.

Facility Proposed Resolution:

Post Examination Comments and Resolutions

The question grading for the exam should not change. Site procedures, other than SAMGs, do not allow bypassing of the interlock for DW spray; the question states "The SAMGs have not been entered". Although system design *provides the means* to bypass the interlock with the given conditions, utilization is clearly not allowed by site procedure.

NRC Resolution:

Upon review of the question, the applicant comment, and the facility proposed resolution, it was decided to accept only the original correct answer.

The question asks, for a given set of conditions during a plant transient, to predict the impact on Drywell temperature and why? Distractor "D" states "Drywell temperature would lower as drywell spray is allowed to be initiated by placing both the Containment Spray 2/3 Core Height Bypass and Containment Spray/Cooling LPCI Initiation switches in BYPASS, and then opening the Drywell Spray inboard and outboard isolation valves." Procedure Ops Man C.5-3502 only allows placing the "Containment Spray 2/3 Core Height Bypass" switch to MANUAL OVERRIDE and the "Containment Spray/Cooling LPCI Initiation" switch in BYPASS, if required to spray the containment by the SAMGs. The question stem states "The SAMGs have not been entered." Although system design *provides the means* to bypass the interlock with the given conditions, bypass of the interlock is clearly not allowed by site procedure. Therefore, distractor "C" was retained as the only correct answer.

Post Examination Comments and Resolutions

Question Number 46

With the reactor operating steady state in MODE 1, what effect, if any, will a loss of instrument air have on the Reactor Building to Suppression Pool Vacuum Relief Dampers, AO-2379 and AO-2380?

A. These valves would fail OPEN.

B. These valves would fail CLOSED.

C. No effect as these valves are normally aligned to air and use nitrogen as a backup supply.

D. No effect as these valves are normally aligned to nitrogen and use bottled nitrogen as a backup supply.

Answer: D

Applicant Comment:

An applicant commented that answer "C" should also be accepted as correct.

Answer "C" should also be accepted. Loss of the instrument air will have no effect on the valves. Therefore, the first part of the question was true. As to the alignment, instrument air is normally aligned to the system, but pressure set points have this valve closed (CV-7478). The valve would automatically open if a problem existed with the Instrument Nitrogen system. The second part of answer "C" states that nitrogen is used as a backup supply. This is true. Alternate Nitrogen is used as the backup supply for these valves. Therefore, answer "C" is technically correct also.

Facility Proposed Resolution:

The question grading for the exam should not change. As stated in the applicant comment above, valve CV-7478, in its normal plant operating status, blocks instrument air from the supply line leading to the Vacuum Relief Dampers; therefore, the Vacuum Relief Dampers are NOT "normally aligned to air" as distractor "C" stated. Answer "D" is the only correct answer.

NRC Resolution:

Upon review of the question, the applicant comment, and the facility proposed resolution, it was decided to accept only the original correct answer. The normal supply to the Reactor Building to Suppression Pool Vacuum Relief Dampers (AO-2379 and AO-2380) is the Instrument Nitrogen supply from the Liquid Nitrogen Tank (reference drawing NH-36049-12). Bottled nitrogen from the Alternate Nitrogen System is the first backup supply to the Vacuum Relief Dampers (reference drawing NH-36049-10). The Instrument Air system supplies a secondary

backup supply to the Vacuum Relief Dampers via valve CV-7478, which opens on low nitrogen header supply pressure (reference drawing NH-36049-12). Therefore, the Vacuum Relief Dampers are not “normally aligned to air” as distractor “C” stated. Distractor “D,” ”No effect as these valves are normally aligned to nitrogen and use bottled nitrogen as a backup supply,” properly described the pneumatic supply to the Vacuum Relief Dampers, and distractor “D” was retained as the only correct answer.

Post Examination Comments and Resolutions

Question Number 59

The reactor was operating at rated conditions when the following events occurred: (Assume SPDS is NOT available)

- 0700: 5-B-46 (CONDENSER LOW VACUUM) alarm was received
- 0705: 7-B-16 (VAC 24 IN TRIP #1) alarm was received
- 0705: 7-B-17 (VAC 24 IN TRIP #2) alarm was received
- 0708: PR-1264, CONDENSERS A AND B VACUUM, on C-07 indicated 6.1" Hg Abs.
- 0710: PR-1264, CONDENSERS A AND B VACUUM, on C-07 indicates 7" Hg Abs. and remains stable

Given the above information, when is the EARLIEST time a manual Reactor Scram is required to be initiated?

A. 0721

B. 0726

C. 0729

D. 0731

Answer: C

Applicant Comment:

An applicant commented that answer "B" should also be accepted as correct.

Answer "B" should also be accepted. Annunciators 7-B-16 and 7-B-17 alarm at 24 inches. By typical engineering convention atmospheric pressure is 30 inches. Therefore, absolute pressure in the condenser would be 6 inches Hg Abs, which would place one in the "ALERT" region of Figure 1 at time 0705, requiring a scram 20 minutes later. Although procedure Ops Man C.4-B.06.03.A, "Decreasing Condenser Vacuum," stated that the operator should use recorder PR-1264 when the Safety Parameter Display System (SPDS) was not available, scramming the reactor at time 0729 would be 21 minutes later, which would not be in accordance with the C.4-B.06.03.A procedure.

Facility Proposed Resolution:

The question grading for the exam should not change. The referenced procedure, C.4-B.06.03.A, "Decreasing Condenser Vacuum," under "INDICATIONS," listed the annunciator 7-B-16 and 7-B-17 alarm setpoints of 24" Hg Vacuum as being equivalent to approximately 5" Hg Abs on recorder PR-1264; this procedure would override the stated "typical engineering convention." The procedure also required a reactor scram when in the "ALERT" region for

longer than 20 minutes, therefore scramming at T+21 minutes is in accordance with the C.4-B.06.03.A procedure.

NRC Resolution:

Upon review of the question, the applicant comment, and the facility proposed resolution, it was decided to accept only the original correct answer. Procedure Ops Man C.4-B.06.03.A, "Decreasing Condenser Vacuum," step 2.a under "INDICATIONS," listed the annunciator 7-B-16 (VAC 24 IN TRIP #1) and 7-B-17 (VAC 24 IN TRIP #2) alarm setpoints of 24" Hg (inches of Mercury) Vacuum as being equivalent to approximately 5 " Hg Abs (Absolute) on recorder PR-1264. Thus at time 0705, when the alarms for annunciators 7-B-16 and 7-B-17 were received, the plant was not in the "ALERT" region of Figure 1, "Turbine Exhaust Pressure Limits," of Ops Man C.4-B.06.03.A. For the reactor at rated conditions, as stated in the question stem, the "ALERT" region of Figure 1 extended from a Turbine Exhaust Pressure of 6.0" Hg Abs to 7.5" Hg Abs. (NOTE: Figure 1 was provided to the applicant with the examination.)

Procedure C.4-B.06.03.A stated that the operator should use recorder PR-1264 when the Safety Parameter Display System (SPDS) was not available (the question stem stated that SPDS was not available). The first instance at which the plant was in the "ALERT" region was at time 0708 when it was given in the question stem that recorder PR-1264, "CONDENSERS A AND B VACUUM," on C-07 indicated 6.1" Hg Abs. Since the C.4-B.06.03.A procedure required a reactor scram when the plant was in the "ALERT" region for longer than 20 minutes, time 0729 (as specified in distractor "C") was the earliest time at which a manual Reactor Scram was required to be initiated in accordance with the procedure (when the plant was in the "ALERT" region for 21 minutes).

The applicant stated that distractor "B" should also be accepted as a correct answer. He stated that by "typical engineering convention" atmospheric pressure is 30" Hg. Since annunciators 7-B-16 and 7-B-17 alarm at 24" Hg Vacuum, the absolute pressure in the condenser would be 6" Hg Abs when annunciators 7-B-16 and 7-B-17 alarmed. This would place one in the "ALERT" region of Figure 1 at time 0705, requiring a scram 20 minutes later. The applicant stated that since distractor "B" was time 0726 (21 minutes after time 0705), this would make distractor "B" a correct answer.

However, procedure C.4-B.06.03.A listed the annunciator 7-B-16 and 7-B-17 alarm setpoints of 24" Hg Vacuum as being equivalent to approximately 5" Hg Abs on recorder PR-1264 (and not 6" Hg Abs), which provided a more accurate representation of the absolute pressure corresponding to 24" Hg Vacuum on recorder PR-1264 than that provided by the estimate of 6" Hg Abs per "typical engineering convention." The applicants were also provided with a book of Steam Tables in which the conversion factor of 2.0360 was given for obtaining pressure in "inches of Hg" from a pressure specified in "pounds per square inch (psi)." Using this conversion factor, a pressure of 14.7 psig would correspond to 29.93" Hg Abs (since Monticello is at approximately the 935 foot elevation above sea level, the nominal atmospheric pressure at the site would actually be reduced to approximately 14.25 psig or 29.0" Hg Abs per Mark's

Standard Handbook for Mechanical Engineers). Using the high end value of 14.7 psig (29.93" Hg Abs) for the pressure corresponding to atmospheric pressure at the Monticello site, when annunciators 7-B-16 and 7-B-17 alarmed at 24" Hg Vacuum, this would correspond to a pressure of 5.93" Hg Abs (NOTE: 24" Hg Vacuum would correspond to 5.0" Hg Abs for the elevation at the Monticello site). Both the values of 5“ Hg Abs (on recorder PR-1264) and 5.93" Hg Abs (from the calculation described above) would place one below the “ALERT” region entry of 6" Hg Abs (i.e., in the “ALLOWABLE OPERATION” region) of Figure 1. Therefore, the procedural requirement to scram when in the “ALERT” region for longer than 20 minutes would not begin when annunciators 7-B-16 and 7-B-17 alarmed at time 0705. Therefore, distractor “B” is not a correct answer, and distractor “C” was retained as the only correct answer.

Post Examination Comments and Resolutions

Question Number 62

While performing test 0081 (CONTROL ROD DRIVE SCRAM INSERTION TIME TEST) the incorrect control rod was withdrawn from position 14 to 48. The crew recognized the error. What action is required to ensure adequate margin to fuel thermal limits exists?

A. Notify Reactor Engineering to run a predictor and provide a recovery plan.

B. Obtain fuel thermal limit information before any rod recovery is performed.

C. Insert the control rod to its previous position without obtaining information on fuel thermal limits.

D. Fully insert the control rod to position 00 and obtain further guidance from Reactor Engineering.

Answer: C

Applicant Comment:

An applicant commented that answer "D" should also be accepted as correct.

Answer "D" should also be accepted as correct. Ops Manual B.01.03-05 H.10, "Inadvertent Control Rod Withdrawal/Rod Drift Out" is used for an inadvertent control rod withdrawal/rod drift out which is the condition in the question stem. This procedure stated that if annunciator 5-A-19, "ROD SELECT BLOCK TIMER MALFUNCTION" was not in alarm (which would be the case in accordance with the question stem), then to immediately insert the rod to position 00 using normal or emergency insert.

Facility Proposed Resolution:

The question grading for the exam should not change. The question asked for the required action following a control rod mispositioning event. The correct answer "C" was to "Insert the control rod to its previous position without obtaining information on fuel thermal limits" this is per site procedures. The applicants missed this question for various reasons, not recognizing the procedural guidance for a control rod inadvertently withdrawn due to operator error, and specifically, for expediting return of the rod to its original position. Distractor "D" would not be a correct action response to the given event; with no equipment problems, there is no reason to insert the control rod further than its original position. A "NOTE" in the same procedure step referenced in the applicant comment stated "In this case the timer malfunction circuit is deficient"; this reinforces that this is the incorrect procedure for this event. CAP 01078752 has been initiated to reinforce practical training on this type of event and to clarify procedural guidance.

Post Examination Comments and Resolutions

It is recommended that the wording for the correct answer be revised to state:

Insert the control rod to its previous position ***prior to*** obtaining information on fuel thermal limits. The current wording could be implied to mean that fuel thermal limits would not be checked even after rod re-positioning.

NRC Resolution:

Upon review of the question, the applicant comment, and the facility proposed resolution, it was decided to accept only the original correct answer. The applicant stated that the correct procedure to use for the conditions stated in the question stem was Ops Manual B.01.03-05 H.10, "Inadvertent Control Rod Withdrawal/Rod Drift Out," since this procedure is used for an inadvertent control rod withdrawal/rod drift. However, the Bases for the procedure states that the procedure is used for inadvertent control rod withdrawal or rod drifting during notch-out movement, for the situation where the rod continues to withdraw past the desired notch at the normal speed or drifts out with a slow erratic movement. These conditions were not those stated in the question stem.

The question was associated with a situation in which the incorrect control rod was withdrawn from position 14 to 48, with the crew subsequently recognizing that the wrong control rod was withdrawn. For this situation, procedure Ops Manual B.01.05-05, Section 3, "Recovery From an Inadvertent Control Rod Withdrawal" was the correct procedure to use. In this procedure, the control rod that was inadvertently withdrawn is inserted to its previous position without obtaining information on fuel thermal limits. The Bases for this procedure states that "check thermal margins prior to inserting the mispositioned rod would result in an unnecessary delay in returning the reactor to its desired configuration." Therefore, distractor "C," "Insert the control rod to its previous position without obtaining information on fuel thermal limits," was retained as the only correct answer.

Post Examination Comments and Resolutions

Question Number 70

Which of the following describes the MCPR safety limit?

A. With the reactor steam dome pressure < 785 psig AND core flow < 10% rated core flow: Thermal Power shall be ≤ 25% RTP.

B. With the reactor steam dome pressure < 785 psig OR core flow < 10% rated core flow: Thermal Power shall be ≤ 25% RTP.

C. With the reactor steam dome pressure ≥ 785 psig AND core flow ≥ 10% rated core flow: MCPR shall be ≥ 1.12 for two recirculation operation or ≥ 1.10 for single recirculation loop operation.

D. With the reactor steam dome pressure ≥ 785 psig OR core flow ≥ 10% rated core flow: MCPR shall be ≥ 1.12 for two recirculation operation or ≥ 1.10 for single recirculation loop operation.

Answer: B

Applicant Comment:

An applicant commented that there is no correct answer.

There was no correct answer. Answer "B" is a Reactor Core Safety Limit (SL) (or conditions) at which Minimum Critical Power Ratio (MCPR) limits do not apply. It is NOT the MCPR limit. The MCPR limits are 1.10 for two loop operations or 1.12 for single loops operations. The first two distractors ("A" and "B") only identifed when MCPR limits may or may not apply. The question specifically asked which of the following described the MCPR safety limits, NOT when the MCPR limits did not apply. The applicant thought the values were reversed in the chosen answer, and went with it, because the first two distractors ("A" and "B") did not answer the question.

Facility Proposed Resolution:

The question grading for the exam should not change.

The Bases for the Reactor Core SLs, BACKGROUND, states that the limits of 2.1.1.1, "Fuel Cladding Integrity," provide "a margin to the conditions that would produce onset of transition boiling (i.e., MCPR = 1.00)".

In the "APPLICABLE SAFETY ANALYSES" of Technical Specification Section B2.1.1.1, "Reactor Core SLs" (the Bases for Safety Limits), the following is stated: The fuel cladding must not sustain damage as a result of normal operation and Anticipated Operational Occurrences (AOOs). The reactor core SLs are established to preclude violation of the fuel design criterion

that a MCPR limit is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

The argument in the applicant comment contradicts these statements in the Bases.

NRC Resolution:

Upon review of the question, the applicant comment, and the facility proposed resolution, it was decided to delete the question from the examination. The question asked "Which of the following describes the MCPR safety limit?" In the Bases for the Reactor Core SLs (Section B 2.1.1), the terminology "MCPR SL" is only associated with reactor core SL 2.1.1.2, which is:

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.10 for two recirculation operation or ≥ 1.12 for single recirculation loop operation.

The above definition of the "MCPR SL" is not associated with any of the distractors.

In Section B 2.2.1.1, distractor "B" (the original correct answer) is associated with reactor core SL 2.1.1.1, "Fuel Cladding Integrity." The BACKGROUND for the Bases for the Reactor Core SLs, does state that "the fuel cladding SL is defined with a margin to the conditions that would produce onset of transition boiling (i.e., MCPR = 1.00)." However, the Bases does not define Section 2.1.1.1 as representing the "MPCR SL." Only section 2.1.1.2 in the Bases is defined as representing the "MPCR SL." Therefore, there is no correct answer, and it was decided to delete the question from the examination.

NOTE: The facility was contacted about the NRC Resolution stated above. The licensee acknowledged the NRC Resolution.

Post Examination Comments and Resolutions

Question Number 79

The reactor was operating in MODE 1 with the following conditions present:

- 11 Instrument Air Compressor, K-1A, is in LEAD
- 14 Instrument Air Compressor, K-1D, is in STBY
- 13 Instrument Air Compressor, K-1C, is taggout out for routine maintenance
- It is the 1900-0700 shift
- The Clearance Order Holder is NOT on site
- Instrument air pressure is 95 psig

If K-1A fails and K-1D is determined to NOT be able to maintain long term instrument air header pressure above the low pressure alarm point, what actions are required for the CRS to direct the removal of Danger Tags and place K-1C in service? (Assume instrument air pressure is lowering 1 psig per 30 minutes.)

A. Immediately sign off the Clearance Order Holder(s) and direct the tags be removed.

B. With concurrence of the Shift Manager, immediately sign off the Clearance Order Holder(s) and direct the tags be removed.

C. Attempt to contact the Clearance Order Holder(s) and if they cannot be reached, direct the tags be removed.

D. Attempt to contact the Clearance Order Holder(s) and if they cannot be reached, and with permission from the WCC, direct the tags be removed.

Answer: C

Applicant Comment:

An applicant commented that answer “D” should also be accepted as correct.

Per procedure FP-OP-TAG-01, “Fleet Tagging,” the “Note” in section 3.7 stated that “Ops Shift Supervision” may be carried out by an Senior reactor Operator (SRO) or persons designated by Ops Management in the Work Control center (WCC) acting for Ops shift supervision, provided the operating crew is kept informed and involved in decisions, when necessary. Ops Shift Supervision and the WCC should be considered to be the same. WCC personnel hours are different than the shift schedule and the WCC person would be there at some time during the shift. Since, in answer “D”, the applicant asked for the WCC permission, this told the applicant the applicant that a WCC person was available to perform the task. For normal day to day operation at the plant, as the CRS, the applicant stated that he would direct the WCC person to perform the necessary actions and have the tags removed; at a minimum, since most work started and ended with the WCC, the applicant stated that he would seek their concurrence and/or permission.

Post Examination Comments and Resolutions

Facility Proposed Resolution:

The question grading for the exam should not change. The question specifically asked for actions required for the CRS to direct removal of the tags and place the unit in service. Although the WCC may also perform these actions, and the WCC may be communicated with prior to the CRS performing these actions, distractor "D" is incorrect as the permission of the WCC is not required.

NRC Resolution:

Upon review of the question, the applicant comment, and the facility proposed resolution, it was decided to accept only the original correct answer. The question asked what actions are required for the CRS to direct the removal of Danger Tags for the K-1C Instrument Air Compressor, in order to place the K-1C compressor in service, during a condition where instrument air pressure is lowering, no other Instrument Air Compressor is available, and the Clearance Order Holder is not on site.

An applicant commented that distractor "D" should also be accepted as the correct answer. The applicant stated that he would seek the WCC concurrence and/or permission prior to directing that the Danger Tags be removed, since for normal day to day operation at the plant, as the CRS, the applicant stated that he would direct the WCC person to perform the necessary actions and have the tags removed, since most work started and ended with the WCC. The applicant also stated that he assumed (based on the the description provided in distractor "D" about obtaining permission from the WCC) that a WCC person was available to perform the task.

However, the question specifically asked for the actions required for the CRS to direct removal of the tags and place the compressor in service. Although the WCC, if available, may also perform the actions to contact the Clearance Order Holder(s) and if they cannot be reached, direct the tags be removed, and the WCC may be communicated with prior to the CRS performing these actions, distractor "D" is incorrect as the permission of the WCC is not required to perform these actions. The applicant also stated that he assumed (based on the the description provided in distractor "D" about obtaining permission from the WCC) that a WCC person was available to perform the task. The question stem stated that it was the 1900-0700 shift. The availability of the WCC was a condition not specified in the question stem. In NUREG-1021, Appendix E, "Policies and Guidelines for Taking NRC Examinations," it stated, in part, that "When answering a question, do not make assumptions regarding conditions that are not specified in the question ...." The applicants were briefed verbatim on the contents of NUREG-1021, Appendix E prior to the administration of the written examination, and were provided a copy of Appendix E. The applicants did not ask for a clarification of the question during the administration of the written examination. Since there was no discussion in the question stem concerning the availability of the WCC, and since permission of the WCC is not required to perform these actions (to contact the Clearance Order Holder(s) and if they cannot be reached, to direct the tags be removed), distractor "B" was retained as the only correct answer.

Post Examination Comments and Resolutions

Question Number 87

During an ATWS event with SBLC injecting, the OATC reports SBLC tank level indicates 400 gallons. Predict the impact of this SBLC system condition on the plant and what actions you would direct as CRS associated with this SBLC condition?

A. The reactor cannot be determined to be shutdown under all conditions, refill the SBLC tank with boron solution.

B. Reactor depressurization can begin as Hot Shutdown Boron Weight has been achieved, secure the SBLC pump.

C. The reactor will remain shutdown under all conditions, secure the SBLC pump when level indicates 0 inches.

D. RPV water level must be retained between -126 inches and -33 inches until Cold Shutdown Boron Weight has been achieved, secure the SBLC pump when level indicates 0 inches.

Answer: C

Applicant Comment:

The applicant recommended removal of this question from the examination due to there being no correct answers.

The applicant stated that Answer "A" was the most correct, since there is procedural direction to refill the Standby Liquid Control (SBLC) tank with boron. Answer "B" was not correct, because Cold S/D Boron weight is required to depressurize. Answer "C" was not correct, because securing SBLC pump when reactor level indicates "0" inches was wrong. Securing the SBLC pump had nothing to do with Reactor Level. Answer "D" was not correct, because Hot S/D Boron weight is required to raise level and securing the SLBC pump had nothing to do with Reactor Level at "0" inches.

Facility Proposed Resolution:

The question should be removed from the exam due to there being no correct answer as worded. The correct answer "C", as stated on the exam, stated "… secure SBLC pump when level indicates 0 inches." This was not specific as to which "level" indicated 0 inches. The SBLC level instruments are in units of "gallons", not "inches." These inconsistencies could lead the applicants to assume that the level indication was referencing reactor water level. In this case, there was no correct answer.

The question should be revised, such that the correct answer states "…when SBLC tank level indicates 0 gallons", prior to inclusion to INPO Exam Bank.

NRC Resolution:

Upon review of the question, the applicant comment, and the facility proposed resolution, it was decided to delete the question from the examination. The original correct answer (distractor "C") stated "… secure SBLC pump when level indicates 0 inches." However, as described by the applicant and the facility, this statement was not specific as to which "level" indicated 0 inches. The SBLC level instruments are in units of "gallons," not "inches." This inconsistency of units could lead the applicants (as per the applicant comment) to assume that in distractors "C" and "D," the level indication specified of "0 inches" was referencing reactor water level, and not a SBLC tank level of 0 gallons. In any case, distractor "C" is incorrect as written to state to "secure the SBLC pump when level indicates 0 inches" instead of "secure the SBLC pump when level indicates 0 gallons." Therefore, there is no correct answer, and it was decided to delete the question from the examination.

WRITTEN EXAMINATIONS AND ANSWER KEYS (RO/SRO)

RO/SRO Initial Examination ADAMS Accession #ML070660539.