June 27, 2007

Mr. Christopher M. Crane
President and Chief Nuclear Officer
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3
NRC INITIAL LICENSE EXAMINATION REPORT 05000237/2007301(DRS); 05000249/2007301(DRS)

Dear Mr. Crane:

On April 30, 2007, U.S. Nuclear Regulatory Commission (NRC) examiners completed initial operator licensing examinations at your Dresden Nuclear Power Station. The enclosed report documents the results of the examination which were discussed on April 27, 2007, with Mr. R. Gadbois and other members of your staff. An exit meeting was conducted on May 29, 2007, between Mr. C. Symonds of your staff and Mr. B. Palagi, Senior Operations Engineer, to review the resolution of the station's post examination comments and the proposed final grading of the written examination for the license applicants.

The NRC examiners administered an initial license examination operating test during the week of April 23, 2007. The written examination was administered by Dresden training department personnel on April 30, 2007. Four Senior Reactor Operator and three Reactor Operator applicants were administered license examinations. The results of the examinations were finalized on June 11, 2007. Four applicant failed the written examination and were issued proposed license denial letters. Three applicants passed all sections of their respective examinations and were issued senior operator licenses.

In accordance with 10 CFR 2.390 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 System (PARS) component of NRC's Agencywide Documents Access and Management System (ADAMS), 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 by N. Valos acting for/

Hironori Peterson, Chief
Operations Branch
Division of Reactor Safety

Docket Nos. 50-237; 50-249
License Nos. DPR-19; DPR-25

Enclosures:
1. Operator Licensing Examination Report 05000237/2007301(DRS); 05000249/2007301(DRS)
2. Simulation Facility Report
3. Post Examination Comments and Resolutions
4. Written Examinations and Answer Keys (RO & SRO)

cc w/encls 1 & 2:
Site Vice President - Dresden Nuclear Power Station
Dresden Nuclear Power Station Plant Manager
Regulatory Assurance Manager - Dresden
Chief Operating Officer
Senior Vice President - Nuclear Services
Senior Vice President - Mid-West Regional Operating Group
Vice President - Mid-West Operations Support
Vice President - Licensing and Regulatory Affairs
Director Licensing - Mid-West Regional Operating Group
Manager Licensing - Dresden and Quad Cities
Senior Counsel, Nuclear, Mid-West Regional Operating Group
Document Control Desk - Licensing
Assistant Attorney General
Illinois Emergency Management Agency
State Liaison Officer
Chairman, Illinois Commerce Commission

cc w/encls 1, 2, 3, and 4: C. Symonds, Training Manager

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Sincerely,

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Hironori Peterson, Chief
Operations Branch
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Chief Operating Officer
Senior Vice President - Nuclear Services
Senior Vice President - Mid-West Regional Operating Group
Vice President - Mid-West Operations Support
Vice President - Licensing and Regulatory Affairs
Director Licensing - Mid-West Regional Operating Group
Manager Licensing - Dresden and Quad Cities
Senior Counsel, Nuclear, Mid-West Regional Operating Group
Document Control Desk - Licensing
Assistant Attorney General
Illinois Emergency Management Agency
State Liaison Officer
Chairman, Illinois Commerce Commission

cc w/encls 1, 2, 3, and 4: C. Symonds, Training Manager

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Letter to Christopher M. Crane from Hironori Peterson dated June 27, 2007.

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3
NRC INITIAL LICENSE EXAMINATION REPORT 05000237/2007301(DRS); 05000249/2007301(DRS)

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REGION III

Docket Nos:	50-237; 50-249
License Nos:	DPR-19; DPR-25
Report No:	000237/2007301(DRS); 05000249/2007301(DRS)
Licensee:	Exelon Generation Company
Facility:	Dresden Nuclear Power Station
Location:	6500 North Dresden Road Morris, IL 60450
Dates:	April 23 through April 30, 2007
Examiners:	B. Palagi, Senior Operations Engineer N. Valos, Senior Operations Engineer D. Reeser, Operations Engineer
Approved by:	Hironori Peterson, Chief Operations Branch Division of Reactor Safety

SUMMARY OF FINDINGS

ER 05000237/2007301(DRS); 05000249/2007301(DRS); 04/23/2007-04/30/2007; Dresden Nuclear Power Station; Initial License Examination Report.

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

Examination Summary:

- Three Reactor Operator and four Senior Reactor Operator examinations were administered.
- Four of the seven applicants failed the written examination and were issued proposed license denials. Three applicants passed all sections of their respective examinations and were issued senior operator licenses.

REPORT DETAILS

4. OTHER ACTIVITIES (OA)

4OA5 Other

.1 Initial Licensing Examinations

a. Examination Scope

The licensee used the guidance established in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9, to prepare the examination outline and to develop the written examination and operating test. The NRC examiners validated the proposed examination during the week of April 2, 2007, at the Dresden Site with the assistance of members of the licensee training staff. During the on-site validation week, the chief examiner audited two license applications for accuracy. The NRC examiners conducted the operating portion of the initial license examination during the week of April 23, 2007. Members of the Dresden training department staff administered the written examination on April 30, 2007. The NRC examiners and the Dresden training department staff used the guidance established in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9, to prepare, validate, revise, administer, and grade the examination.

b. Findings

Written Examination

The licensee developed the written examination. During their internal review, the NRC examiners determined that the examination, as submitted, was within the range of acceptability expected for a proposed examination. Written examination comments developed during review by the NRC staff, and as a result of examination validation were incorporated into the written examination in accordance with the guidance contained in NUREG-1021.

A total of nine post-examination comments on the written examination were submitted by the applicants and station training department personnel on May 7, 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 for a proposed examination. The examiners validated the operating test during the validation week and replaced or modified several items in the proposed operating test. Test changes, agreed upon between the NRC and the licensee, were made in accordance with NUREG-1021 guidelines. The NRC examiners completed operating test grading on May 4, 2007.

Examination Results

Four applicants at the Senior Reactor Operator (SRO) level and three applicants at the Reactor Operator (RO) level were administered written and operating tests. One of the SRO applicants was previously licensed as a RO at Dresden. Four applicants failed the written examination and were issued proposed license denials. Three initial SRO applicants passed all portions of their examinations and were issued operating licenses.

.2 Examination Security

a. Scope

The NRC examiners reviewed and observed the licensee's implementation of examination security requirements during the examination validation and administration to assure compliance with 10 CFR 55.49, "Integrity of Examinations and Tests." The examiners used the guidelines provided in NUREG 1021, "Operator Licensing Examination Standards for Power Reactors" to determine acceptability of the licensee's examination security activities.

b. Findings

No findings were noted in this area.

4OA6 Meetings

Debrief

The chief examiner presented the examination team's preliminary observations and findings on April 27, 2007, to Mr. R. Gadbois and other members of the Dresden Operations and Training Department staff. The examiners asked the licensee whether any of the material used to develop or administer the examination should be considered proprietary. No proprietary or sensitive information was identified during the examination or debrief/exit meetings.

Exit Meeting

The chief examiner conducted an exit meeting with Mr. C. Symonds, Dresden Training Director and other members of the Dresden Operations and Training Department staff on May 29, 2007. The NRC's final disposition of the station's post-examination comments were discussed and revised preliminary written examination results were provided.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
R. Gadbois, Operations Director
C. Symonds, Training Director
G. Graff, Operations Training Director
F. Ferrero, Operations Training

NRC
B. Palagi, Chief Examiner
N. Valos, Examiner
D. Reeser, Examiner

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened, Closed, and Discussed

None

LIST OF DOCUMENTS REVIEWED

None

LIST OF ACRONYMS USED

ADAMS	Agency-Wide Document Access and Management System
DRS	Division of Reactor Safety
NRC	Nuclear Regulatory Commission
IR	Inspection Report
RO	Reactor Operator
SRO	Senior Reactor Operator

SIMULATION FACILITY REPORT

Facility Licensee: Dresden Nuclear Power Station

Facility Docket No.: 50-237; 50-249

Operating Tests Administered: April 23-27, 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
None	

Post Examination Comments and Resolutions

Question Number 10

Why is the required quantity of boron GREATER for COLD shutdown boron weight than it is for HOT shutdown boron weight?

A. To overcome a greater RPV water level.

B. To overcome the reduction in Xenon.

C. To overcome the reduction in Samarium.

D. To overcome a reduction in voids present in the core.

Answer: B

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

Hot Shutdown Weight is the least weight of soluble boron which, if injected into the Reactor Pressure Vessel (RPV) and mixed uniformly, will maintain the reactor shutdown under Hot Standby conditions. In accordance with Dresden Technical Specifications, Hot Shutdown is greater than 212°F where voiding would be present in the core resulting in decreasing power.

Cold Shutdown Weight is the least weight of soluble boron which, if injected into the RPV and mixed uniformly, will maintain the reactor shutdown under all conditions. This weight is utilized to assure the reactor will remain shutdown irrespective of control rod position or RPV temperature. In accordance with Dresden Technical Specifications, Cold Shutdown is less than 212°F where voiding would no longer be present in the core resulting in increasing power, and the need for increased boron to ensure the reactor would remain shutdown.

Facility Proposed Resolution:
The Training Department agrees with the challenge on question #10, that there are two correct answers. The question asks why is the required quantity of boron greater for Cold shutdown boron weight than it is for Hot shutdown boron weight. The answer in the key for the question is to overcome the reduction in Xenon. Based on the support document used by the author, (EPG Appendix B sections 17.2 and 17.6) the calculations assume no Xenon present for Cold shutdown and full power Xenon equilibrium for Hot shutdown. Therefore distractor 'B' to overcome the reduction in Xenon remains correct.

The definition for Cold Shutdown Boron Weight on page B-17-9 of BWROG EPG's/SAG's, Appendix B states "The Cold Shutdown Boron Weight (CSBW) is that amount of soluble boron which, if injected into the RPV and mixed uniformly, will maintain the reactor shutdown under all conditions. The CSBW is utilized to assure the reactor will remain shutdown irrespective of control rod position or RPV water temperature." With the reactor less than 212 degrees voids would no longer be present and the need for a greater quantity of Boron would be needed to maintain the reactor shutdown. With the reactor at Hot Standby conditions, (i.e., greater than 212 degrees), the void coefficient and temperature coefficient will be present. Without a distractor for the temperature coefficient, the candidates chose the distractor that included voids.

The definition for Hot Shutdown Boron Weight on page B-17-22 of BWROG EPG's/SAG's, Appendix B states "The Hot Shutdown Boron Weight (HSBW) is the least weight of soluble boron which, if injected into the RPV and mixed uniformly, will maintain the reactor shutdown under hot standby conditions. The HSBW is utilized to assure the reactor will remain shutdown irrespective of control rod position when RPV water level is raised to uniformly mix the injected boron." The definition section of the Technical Specifications 1.1 table 1.1-1 states that the difference between Hot Shutdown and Cold Shutdown is whether Reactor coolant temperature is greater than or less than 212 degrees. This is also stated in the DEOP bases B-6-49 and B-14-45. From an Operational standpoint with coolant greater than 212 degrees and all rods in, voiding is still occurring in the core providing negative reactivity. When temperature goes below 212 degrees, voiding in the core will cease and coolant density will increase. Formatting of the question emphasizes COLD and HOT, **NOT** the design basis for Cold Shutdown Boron Weight and Hot Shutdown Boron Weight. This could lead the students to utilize the Tech. Spec./Operational definition of Hot Shutdown per Table 1.1-1, and Section 9.3.5.3 of the UFSAR, which would have voiding. In addition, Lesson Plan 295L-S08 DEOP 400-5, refers to pressure and temperature conditions but does not mention an assumption of no voiding. Due to this circumstance and the fact that the question does not ask "per the bases," answer 'D' is also correct. The boron would also have to overcome a reduction in voids present in the core. Ten percent of the examination specifically asks the student to answer a question based on the basis, FSAR, or governing procedure. Question 2, 18, 27, 33, 50, 56, 65, 77, 80, and 90 all follow this format. The current program does not teach the students the design basis calculations for CSBW and HSBW. TR 07-0829 was written to include the material in lesson plan 295LC01.

Students are trained during the DEOP phase while performing DEOP 400-5 "Failure to SCRAM" to raise level when the HSBW is reached to allow for mixing in the core. They are also taught that commencing a cool down is only allowed after the CSBW is reached. DEOP 400-5 only allows for a cool down to commence once the CSBW is reached thus the assurance of maintaining a shutdown condition under all conditions. Students have executed scenarios where injection of HSBW and CSBW, have occurred and the time between the two evolutions is less than 30 minutes. With the short time frame, students are not concerned with Xenon from an operational standpoint, which further emphasizes the void effect as a potential answer.

The Xenon-135 Peak after Shutdown curve (BWR/Reactor Theory/Chapter 6/TP 6-1/Rev 3 Fig-6-3) illustrates that Xenon concentration will increase after a scram peaking approximately 10 hours after the scram adding negative reactivity during that time span. Figure 6-5 illustrates that after a power reduction, Xenon peaks 8-10 hours, which would be the case for an ATWS situation. Due to Xenon building in after a scram and temperature still above 212 degrees the candidates could assume that voiding is the reason for the additional boron concentration requirements. The addition of positive reactivity during Xenon burnout does not occur until after 10 hours of reactor shutdown.

The question stem did not emphasize for the student that it was looking for the design basis calculations but instead led them to consider the operational concern of temperature.

Based on the discussion above, 'B' and 'D' are both correct.

<u>NRC Resolution</u>:
Upon review of the question, the applicant comment, and the facility proposed resolution, it was decided to only accept the original correct answer.

In the LEVEL control leg of DEPO 400-5 FAILURE TO SCRAM it is directed that if reactor power is above 6%, water level should be lowered in an attempt to drop below 6% reactor power. The reactivity effect that is being utilized when dropping water level to lower reactor power is increased voiding of the reactor core. After water level has been lowered DEPO 400-5 directs "**WAIT** until Hot Shutdown Boron injected" before raising water. The reason it is directed that water level can be increased at this point is that sufficient boron has been injected into the reactor that the core should remain shutdown at current conditions with NO voids present. Therefore, a license candidate with a adequate understanding of the reason for waiting for Hot Shutdown Boron Weight would know answer "D" is incorrect.

The major negative reactivity effects maintaining subcriticality for the conditions assumed in the Hot Shutdown Boron Weight calculation are that: full-power Xenon is present; control rods are inserted to their minimum full power position at the most reactive time in core life; and the reactor water temperature corresponds to a pressure of 1100 psia. Each of these effects are minimized in the Cold Shutdown Boron Weight calculation and would be reasons for greater cold than hot shutdown boron weight. However, the only one offered as a possible answer was a reduction in Xenon.

The Cold Shutdown Boron Weight (CSBW) is defined in the BWR Owners Group (BWROG) document BWROG EPGs/SAGs, Appendix B, Section 17.2. The assumptions used to calculate the CSBW are also listed in the same section. For the CSBW, one of the assumptions is that "No voids are present in the reactor core." An additional assumption is that "No Xenon is present in the reactor core."

The Hot Shutdown Boron Weight (HSBW) is defined in Section 17.6 of the same document. The assumptions used to calculate the HSBW are also listed in the same section. For the HSBW, one of the assumptions is that "No voids are present in the reactor core." For the HSBW, an additional assumption is that "Full power equilibrium Xenon is present in the reactor core."

Since, the CSBW calculation states that no Xenon is present in the reactor core, whereas the HSBW calculation states that full power equilibrium Xenon is present in the reactor core, an additional quantity of boron is required when calculating the CSBW to overcome the positive reactivity addition caused by the decay of the full power equilibrium Xenon. Thus, distractor "B" is a correct answer.

Since the CSBW and the HSBW calculations each assume that no voids are present in the reactor core, distractor "D" (to overcome a reduction in voids present in the core) is not a correct answer.

Since there is only one definition for the CSBW and only one definition for the HSBW (from the BWROG EPGs/SAGs, Appendix B), and these definitions state that no voids are present in the reactor core for the CSBW and the HSBW, distractor “B” was retained as the only correct answer.

Question Number 14

Unit 2 is in a refuel outage, with the following conditions:

- Divers are needed to enter the Unit 2 Torus for the 4-year check for plugging of the ECCS strainers.
- Operations, Contract Personnel, and Engineering all have responsibilities associated with the performance of this evolution.

Of the positions listed below, who is the HIGHEST level of authority required to approve this evolution?

A. Nuclear Station Operator

B. Unit Supervisor

C. Shift Manager

D. Operations Director

Answer: D

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

Realizing procedure HU-AA-1211, Sections 3.1 and 4.4.3 state that the Operations Director approves evolutions requiring a High Level Awareness (HLA) or Infrequent Plant Activity (IPA), the question asks who approves the evolution, not the evolution requiring a HLA or IPA. The Unit Supervisor has control of the Unit and approves all evolutions on the Unit per procedure OP-AA-101-111, Section 4.2.7.

The question is misleading. During a refueling outage, the Unit Supervisor is in control of the Unit. Planning for a 4-year surveillance would have been done prior to the outage start. Approval to perform (start) this job would come from the Unit Supervisor, after reviewing all briefs and packages.

Depending on the situation for this question, the applicant believes there are two correct answers ("B" and "D").

Facility Proposed Resolution:
The question grading for the exam should not change.

Procedure HU-AA-1211 states that Torus diving operations require an IPA briefing, and that Senior Line Management (Operations Director) are responsible for approving evolutions that require an IPA briefing.

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 the Unit Supervisor has control of his Unit and approves all evolutions on his Unit per procedure OP-AA-101-111, Section 4.2.7.

Procedure HU-AA-1211, Section 4.4.3 states that Torus Diving Operations is an activity that requires an IPA briefing. Section 3.1 of the same procedure states that Senior Line Management (which includes the Operations Director) approves evolutions requiring IPA briefings.

Since the Operations Director (distractor “D”) is the only distractor that includes Senior Line Management, and the question stem asks who is the HIGHEST level of authority who is required to approve the evolution, distractor “D” was retained as the only correct answer.

Question Number 18

What is the operational concern with excessive moisture traveling downstream of the Off Gas Preheater per DOP 5400-23, 2A STEAM JET AIR EJECTOR/RECOMBINER STARTUP, OPERATION AND SHUTDOWN?

A. Off Gas fire

B. Overheating of the Recombiner

C. Overpressurization of the Recombiner

D. Reduction in Main Condenser vacuum

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

The question asks "What is the operational concern with excessive moisture (in the recombiner) per DOP 5400-23. The discussion section writes about the catalyst being quenched, stopping the hydrogen/oxygen combination, resulting in high hydrogen concentrations. This condition would result in both possibilities of an Off Gas fire and the reduction of Main Condenser vacuum (answer "D"). Procedure DOP 5400-14, "Extinguishing an Off Gas Fire," Prerequisites, Precautions, and Procedure sections, all give evidence of symptoms of Off Gas fire as a "Sudden Reduction in flow," and "All available Circulating Water Pumps should be running ... due to reduced main condenser vacuum." The only reference in DOP 5400-23 to Off Gas fires is a Precaution stating "Valving operations ... should be conducted slowly to minimize pressure/flow perturbations, which can cause Off Gas fires," referencing the "Discussion" section, which also states "At too low or high a pressure, SJAE flow will be lost."

Facility Proposed Resolution:
The training department agrees with the challenge on question #18, that there are two correct answers. The question asks what is the Operational concern with excessive moisture traveling downstream of the Off Gas Preheater per DOP 5400-23 STEAM JET AIR EJECTOR/RECOMBINER STARTUP, OPERATION AND SHUTDOWN. If water is forced through the catalyst, the quenching will stop the hydrogen/oxygen combination resulting in high hydrogen concentrations downstream of the recombiner. This results in H2O2 downstream that can exceed flammable or explosive limits.

The following documents all reference that changes in flow, temperature, and pressure will result based on the conditions in the stem of the question:

DOP 5400-23 Steam Jet Air Ejector/Recombiner Startup Operation and Shutdown
DOP 5400-14 Extinguishing an Off Gas Fire
DAN 902-7 C-14 Off Gas Sys Flow Hi/Lo
DAN 902-7 B-13 Off Gas Temp Hi
DAN 902-7 A-13 Off Gas Press Hi
DAN 902-7 C-13, GE SIL No. 150 Revision 2 Supplement 1
DOA 3300-2 Loss of Condenser Vacuum
Specific Dresden Station Operating Experience

DOP 5400-23 Steam Jet Air Ejector/Recombiner Startup Operation and Shutdown: Having moisture downstream of the Off Gas preheater will force water through the catalyst. This quenching will stop hydrogen/oxygen combination resulting in high hydrogen concentrations downstream of the recombiner. There is no mention in DOP 5400-23, of this causing Off Gas fires. What it discusses is that valving operations (including manual isolation of components, valving in/out pressure controllers and pressure controller manual bypass valves) should be conducted slowly to minimize pressure/flow perturbations, which **CAN** cause Off Gas fires. Since the question clearly states per DOP 5400-23 2A STEAM JET AIR EJECTOR/RECOMBINER STARTUP, OPERATION AND SHUTDOWN, the operator is left to determine if an Off Gas Fire would occur or if the system performance would be degraded leading to a reduction in Main Condenser Vacuum. Whether or not a fire occurs would still make Reduction in Vacuum an "operational concern" for the conditions given.

DOP 5400-14 Extinguishing an Off Gas Fire: Throughout the procedure there are cautions which state, do not allow flow to exceed 200 scfm which may cause a loss of the Main Condenser Vacuum. The caution leads the operator to believe that the major operating concern with the Offgas system is a loss of condenser vacuum.

DAN 902-7 C-14 Off Gas Sys Flow Hi/Lo: Step B.1.a states in part to check Main Condenser Vacuum. This is the first action of the DAN.

DAN 902-7 B-13 Off Gas Temp Hi: Step B.1 states to reduce power to reduce condenser vacuum drop. This is the first action of the DAN.

DAN 902-7 A-13 Off Gas Press Hi: Step B.1 states to reduce power to reduce condenser vacuum drop. This is the first action of the DAN.

DAN 902-7 C-13, GE SIL No. 150 Revision 2 Supplement 1: Step B.1.a states in part to check Main Condenser Vacuum. This is the first action of the DAN.

The first action of DOA 3300-2 Loss of Condenser Vacuum is to check main condenser vacuum or Reduce load per DGP 03-1, Routine Power Changes, to minimize the resultant condenser vacuum drop.

Each of the DANs has you either check condenser vacuum or reduce power to conserve vacuum. While the DANs in some cases discuss and have actions regarding a potential fire and or explosion, the first steps performed regard loss of vacuum. Based on the primary response of handling vacuum, it is easy to conclude that the primary operating concern is a loss of vacuum.

OE 7600 BWR Offgas Fires specific to Dresden discusses in the abstract section that no damage was experienced during an Offgas fire that occurred. The OE also states under the safety significance section "There is no known short term harmful effect from Offgas fires". Based on the OE one can conclude that the operational concern again is related to a loss of vacuum and the fire and/or explosion is the symptom.

Based on the discussion above there are two correct answers "A" and "D."

NRC Resolution:
Upon review of the question, the applicant comment, and the facility experienced resolution, it was decided to accept two correct answers.

The question stem asks what is the operational concern with excessive moisture traveling downstream of the Off Gas Preheater. The catalytic recombiner is located just downstream of the Off Gas Preheater. In procedure DOP 5400-23, “2A Steam Jet Air Ejector/Recombiner Combined Startup, Operation, and Shutdown,” Section H.3, it states that if water is forced through the catalyst, the quenching will stop the hydrogen/oxygen combination resulting in high hydrogen concentrations downstream of the recombiner. Precaution E.1 of DOP 5400-23 mentions that valving operations should be conducted slowly to minimize pressure/flow perturbations, which can cause Off Gas fires. High hydrogen concentrations downstream of the recombiner due to excessive moisture in the catalyst would increase the potential for an Off Gas fire in the location downstream of the recombiner (answer A).

An Off Gas fire would also result in a disruption of normal air ejector flow and the potential for a “Reduction in Main Condenser vacuum” (answer D). In alarm procedure DAN 902-7 B-13 “OFF GAS TEMP HI” the first PROBABLE CAUSE is “off gas system fire/explosion” and the first step under OPERATOR ACTIONS is to reduce unit load “to reduce condenser vacuum drop.”

Additionally, Dresden and other BWRs have experienced Off Gas fires, this experience has shown that the Off Gas system can continue to operate for a significant length of time with a fire. However an Off Gas fire cannot be allowed to burn indefinitely. To extinguish the fire the either air would be injected into the stream to dilute the hydrogen concentration, or the idle air ejector train would started and air ejector train with the fire shutdown. Either of these actions have the potential to disrupt off gas flow. Therefore, knowing that operation of the plant could continue with an Off Gas fire in progress, a major operational concern becomes extinguishing an Off Gas fire while dealing with the potential for a reduction in Main Condenser vacuum (answer D).

Based on both an Off Gas fire and the resulting reduction in condenser vacuum as a result of the fire being valid operational concerns, the answer key was modified to accept both “A” and “D” as correct answers.

Question Number 21

Unit 2 has been shutdown for 30 hours, with the following set of conditions:

- 2A and 2C SDC pumps are running.
- 2A RBCCW pump is running.
- 2/3 RBCCW pump is running, lined up to Unit 2 and powered from Unit 2.

Then the following occurred:

- Due to a breaker malfunction, Bus 23-1 lost power and was subsequently re-powered.
- RBCCW parameters have stabilized two hours following the transient.

What will the current RBCCW pressure AND RBCCW temperature be compared to the pre-transient values?

Current RBCCW pressure . . . . .

A. will be the same and temperature will be the same.

B. will be lower and temperature will be higher.

C. will be the same and temperature will be lower.

D. will be lower and temperature will be the same.

Answer: A

<u>Applicant Comment</u>:
An applicant commented that answer “C” should also be accepted as correct.

At Dresden during the summer months, with the intake at higher temperatures, the temperature control valves (TCVs) for the Reactor Building Closed Cooling Water (RBCCW) system are full open. With a higher heat condition (since per the question stem, Shutdown Cooling (SDC) is on with a reactor still giving off decay heat), the temperature of RBCCW would be higher than set point due the TCVs already full open. With a loss of SDC, there would be much less heat transfer to the RBCCW, and the temperature could be controlled lower than the pre-transient values. The TCVs are set at 90°F year round per procedure DOP 3700-02. The TCVs can be adjusted per Unit Supervisor approval. The TCVs are normally lowered to 80 to 85°F during the summer due to intake temperatures being higher.

<u>Facility Proposed Resolution</u>:
The question grading for the exam should not change.

As stated in the question stem, the unit is shutdown, which means a reduced heat load to begin with. The candidate’s assumption that during the summer months the system TCV would be full open can not be assumed, since it is not stated in the stem. The question stem states that parameters have stabilized two hours after the transient. With the system parameters stabilized, the TCV would be controlling at its previous set point.

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 during the summer months, the RBCCW system TCVs would be full open. The applicant stated that with a loss of SDC, there would be much less heat transfer to the RBCCW, and the temperature could be controlled lower than the pre-transient values. He thus stated that distractor "C" (Current RBCCW pressure will be the same and temperature will be lower) was also a correct answer. The time of the year (i.e., summer months) and the position of the RBCCW TCVs were conditions 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 time of the year or the position of the RBCCW TCVs, distractor "C" is not correct, and distractor "A" was retained as the only correct answer.

Question Number 27

Per the UFSAR, what is the reason for having LPCI pumps operating with the Torus CLG/TEST valves throttled open following Reactor vessel flooding?

A. To ensure adequate mixing of the Torus water.

B. To maintain Torus level in the normal operating band.

C. To immediately terminate the increase in Torus temperature.

D. To terminate the increase in Torus temperature after several hours.

Answer: D

Applicant Comment:
One applicant commented that answer "A" should also be accepted as correct. The applicant commented that the Updated Final Safety Analysis Report (UFSAR) stated suppression pool cooling will be initiated to control suppression pool temperature. The applicant stated that the question stem does not state initiation of suppression pool cooling. The Low Pressure Coolant Injection (LPCI) heat exchanger (HX) bypass valves MO 2(3)-1501-11 A/B may still be open. If the LPCI HX bypass valves are open, no cooling will take place, and only mixing of the torus water will occur. If the LPCI HX bypass valves are closed, then cooling of the torus water will be accomplished.

A second applicant commented that answer "B" should also be accepted as correct. The applicant stated that when the plant is flooded to the Automatic Depressurization System (ADS) valves, one maintains reactor pressure between saturation pressure to 100 psi above saturation pressure. If full flow LPCI continues to go to the vessel, one would exceed the reactor pressure vessel (RPV) pressure limits. Opening the Torus Cooling/Test valves (MO 2(3)-1501-20A/B and MO 2(3)-1501-38A/B) would divert flow from LPCI injection to the reactor vessel, thus maintaining reactor pressure and maintaining torus level.

A third applicant commented that answer "C" should also be accepted as correct. The applicant stated that when the Torus Cooling/Test valves (MO 2(3)-1501-20A/B and MO 2(3)-1501-38A/B) are open, immediate cooling of torus temperature will occur, assuming that the LPCI HX bypass valves MO 2(3)-1501-11 A/B are closed.

Facility Proposed Resolution:
The question grading for the exam should not change.

The candidates claims that mixing of Torus water, maintaining Torus level in normal band, or immediately terminating increasing Torus temperature, may also be correct, but can be disregarded, since the question stem states "per the UFSAR." The UFSAR states that Suppression Pool Cooling will be initiated to control pool temperature after several hours.

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.

Throttling open the Torus CLG/TEST valves is done as part of placing LPCI in torus cooling mode following a LOCA to remove decay heat. In Section 6.2.1.3.4.1 (page 6.2-33) of the Dresden UFSAR, it states that “Following vessel flooding ... Operators will initiate suppression pool cooling to control suppression pool temperature. After several hours, the containment cooling heat exchangers will terminate the increase in the suppression pool temperature.” This statement makes distractor “D” (to terminate the increase in Torus temperature after several hours) a correct answer.

The question stem asked “**Per the UFSAR** (emphasis added), what is the reason for having LPCI pumps operating with the Torus CLG/TEST valves throttled open following Reactor vessel flooding?” Since the UFSAR does not mention having the LPCI pumps operating with the Torus CLG/TEST valves throttled open (1) to ensure adequate mixing of the Torus water (distractor “A”), (2) to maintain Torus level in the normal operating band (distractor “B”), or (3) to immediately terminate the increase in Torus temperature (distractor “C”), distractor “D” was retained as the only correct answer.

Question Number 34

You are a licensed NSO performing a JPM at a Unit 2 CRD accumulator as part of requalification training. You hear a continuous 2-minute siren followed by an announcement directing all personnel NOT having emergency assignments, to report to the CLOSEST assembly area.

To which of the following areas are you required to report?

A. Main Control Room

B. Operation Support Center (OSC)

C. Unit 2 Turbine Building Main Corridor

D. Administration Building Lunchroom/Foyer Area

Answer: C

Applicant Comment:
An applicant commented that there is no correct answer.

The applicant stated that the card readers are not in the Unit 2 Turbine Building Main Corridor. The card readers are located in the Unit 2 Turbine Building Trackway. Dresden Annex Procedure EP-AA-1004 has an assembly area location prior to relocation of Radiological Controlled Area (RCA) entrance.

Facility Proposed Resolution:
The question grading for the exam should not change.

The question stem asked the candidates which assembly area was the closest, not whether or not there was a card reader located in the assembly area. A side note is that the Unit 2 Turbine Building Trackway is adjacent to the Unit 2 Turbine Building Main Corridor.

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 there are no card readers in the Unit 2 Turbine Building Main Corridor. He stated that the card readers are located in the Unit 2 Turbine Building Trackway. However, the question stem asked the applicants which assembly area was the closest to the Unit 2 CRD accumulator area, not whether or not there was a card reader located in the assembly area. Thus, distractor “C” was retained as the only correct answer.

Question Number 45

A major fault occurs on the 2A Instrument Air Compressor (IAC), but it's feed breaker does NOT trip.

Bus ___(1)___ will de-energize and the ___(2)___ to re-energize the de-energized Bus.

A. (1) 20;
(2) Bus 20 to Bus 24 cross-tie breaker(s) will AUTOMATICALLY close

B. (1) 24;
(2) Operator will MANUALLY close the Bus 24 to Bus 24-1 cross-tie breaker(s)

C. (1) 25;
(2) Operator will MANUALLY close the Bus 25 to Bus 27 cross-tie breaker(s)

D. (1) 27;
(2) Bus 25 to Bus 27 cross-tie breaker(s) will AUTOMATICALLY close

Answer: C

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

Distractors "A" and "D" are not viable.

Depending on the direction of cascading faults, either of the remaining two answers (distractors "B" and "C") are viable.

The fault would need to be corrected/cleared and then re-energization of the bus could be completed.

Facility Proposed Resolution:
The question grading for the exam should be changed to accept both "B" and "C" as correct answers.

With a fault on 2A Instrument Air Compressor (IAC) and a failure of its motor feed breaker to trip, a fault is sensed at the Bus feeding 2A IAC (Bus 26). Given the question stem statement that this is a "major fault," the candidates could have interpreted this as a cascading fault that would feed through the feed breaker and main feed breaker to Bus 24, which would cause Bus 24 to become de-energized, or through the Bus 26 to Bus 25 cross-tie breaker (after closing), which would cause it to become de-energized. With this in mind, either distractor "B" or "C" would be a correct answer.

Reference event 237-900116-1 where a similar event occurred. In this event, a pump powered from Bus 24 faulted, its feed breaker did not trip, the fault was sensed on Bus 24, but the feed breakers from the supply transformer did not trip in a timely manner, resulting in a loss of power and ultimately a reactor scram.

NRC Resolution:
Upon review of the question, the applicant comment, and the facility proposed resolution, it was determined that there was no correct answer.

Answer choices "A" and "D" are clearly wrong since there are no automatic closure signals for these breakers.

The question as written postulates a motor fault on the 2A Instrument Air Compressor (which is feed from Bus 26) and a failure of the motor feed breaker to open. This would result in over-current de-energization of Bus 26 by the opening of a feed breaker from Bus 24. Loss of power to Bus 26 would result in automatic transfer of Bus 26 to backup feed from Bus 25. With the fault still present on Bus 26, there are two possible out comes: 1) either the tie breaker between Bus 25 and Bus 26 trips on over-current, leaving Bus 25 energized; or 2) over-current on Bus 25 causes a Bus 25 feed breaker to open, de-energizing Bus 25 and Bus 26 If the breaker trip settings are properly coordinated then the Bus 25-26 tie breaker should trip open on a short current before the bus main feed breaker, and Bus feed breakers should trip before a suppling Bus is tripped. (i.e., breaker coordination should be designed such that a fault on a 480 V bus should not result in an over-current condition on the 4160 V bus, or the trip of an additional Bus.) Therefore, if the breakers are properly coordinated, Bus 25 and 24 should remain energized following a fault on Bus 26, therefore both answers "B" and "C" are also incorrect.

The facility references an event to support the position of the applicants that "cascading faults" could result in the trip of Bus 24. In the referenced Licensee Event Report (LER 237-1990-002) the 2D Condensate/Condensate Booster pump did trip and the faulted motor was separated from the associated 4160 V bus. A few seconds later the reactor scrammed due to a loss of feed. Approximately 2 minutes later the Reserve Auxiliary Transformer (RAT) tripped and that coupled with the Turbine Generator trip results in a loss of off-site power. The failure of the RAT was attributed to insulation breakdown caused by coils rubbing together under changes in electrical load. The coil looseness was attributed to insulation shrinkage from normal aging. While increased loading on the transformer during the event may have contributed to the transformer failure it was not a root cause. The internal RAT failure resulted in the de-energization of Bus 24, not the condensate/condensate booster pump failure. This event is significantly different and unrelated to the event postulated in the exam question. The assumption that Bus 24 could be de-energized by cascading faults is not supported as described by this event.

In NUREG-1021, Appendix E, "Policies and Guidelines for Taking NRC Examinations," it is 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. Nothing in the stem of the question supports the assumption of a cascading fault.

Since there are no correct answers, the question will be deleted from the exam.

Question Number 59

Unit 2 was operating at near rated power, when the following occurred:

- NSO increased Recirc flow slightly, using MASTER RECIRC FLOW CONTLR.
- Oil pressure on 2A MG Set decreased to 25 psig for 3 seconds then returned to normal.
- Oil pressure on 2B MG Set decreased to 29 psig for 7 seconds then returned to normal.

Which of the following describes the actions (if any) that are required to be taken concerning the Recirc Flow Control System?

A. NO action required.

B. Place BOTH RECIRC PP SPEED CNTLRs in MAN. Dial BOTH RECIRC PP SPEED CONTLRs potentiometers to 30%.

C. Place BOTH RECIRC PP SPEED CNTLRs in MAN. Dial the 2A RECIRC PP SPEED CONTLR potentiometer ONLY to 30%.

D. Place BOTH RECIRC PP SPEED CNTLRs in MAN. NO adjustment to the RECIRC PP SPEED CONTLR potentiometers are required.

Answer: C

<u>Applicant Comment</u>:
An applicant commented that answer “A” should also be accepted as correct.

The applicant stated that with no change to speed when the scoop tube locks out on low oil pressure, there are no immediate actions per alarm response procedure DAN 902-4 C-5, “2B Recirc M-G Scoop Tube Failure.” The Operations Actions per procedure DAN 902-4 C-5 direct one (per step B.7) to restore scoop tube operation per DOP 0202-12, “Recirculation Pump Motor Generator Set Scoop Tube Operation.” The applicant stated that he assumed that the question was asking for the immediate actions to take on the recirculation flow system, not the subsequent actions and the directions from DAN 902-4 C-5 to enter DOP 0202-12. Therefore, he determined that both distractors “A” and “C” could be correct answers.

<u>Facility Proposed Resolution</u>:
The question grading for the exam should not change.

Per procedure DOP 0202-12, step E.4, the Operator is directed to place the master and individual speed controllers in Manual mode if, anytime during operation, either scoop tube is locked out. Subsequently, when scoop tubes are locked or the Motor Generator (MG) sets are off, then target 30% demand on the 2A/B Recirculation Pump Speed Controllers.

<u>NRC Resolution</u>:
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 he assumed that the question was asking for the immediate actions to take on the recirculation flow system, not the subsequent actions and the directions from DAN 902-4 C-5 to enter DOP 0202-12. Therefore, he determined that distractors “A” (NO action required) could also be a correct answer. 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 only asking for the immediate actions to take on the recirculation flow system, distractor “A” is not correct, and distractor “C” was retained as the only correct answer.

Question Number 91

A LOCA has occurred on Unit 2, concurrently with a LOOP, with the following conditions:

- Reactor Pressure is 300 psig and LOWERING.
- HPCI and SBLC are the ONLY high pressure systems available AND injecting into the Reactor.
- RPV water level is -193" and LOWERING.
- BOTH loops of Torus Sprays are in operation.
- BOTH loops of Torus Cooling are in operation.
- Drywell sprays are NOT in operation due to valve binding on both loops.
- Drywell Pressure is 19 psig and RISING.
- Torus Bottom Pressure is 24 psig and RISING.
- Torus Level is 14 feet and STABLE.

Complete the following statements.

The SRO is required to direct the NSO to ___(1)___ and blowdown is required based upon ___(2)___.

A. (1) CONTINUE to operate Torus Cooling AND Torus Sprays;
(2) Torus Bottom Pressure ONLY

B. (1) CONTINUE to operate Torus Cooling AND Torus Sprays;
(2) Reactor Water Level AND Torus Bottom Pressure

C. (1) STOP Torus Cooling AND Torus Sprays;
(2) Reactor Water Level ONLY

D. (1) STOP Torus Cooling AND Torus Sprays;
(2) Reactor Water Level AND Torus Bottom Pressure

Answer: C

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

As given in the question stem, reactor level is -193" and lowering, which is below the top of the active fuel. Per the level leg in DEOP 100, "RPV Control," the Senior Reactor Operator (SRO) would direct reactor pressure vessel (RPV) blowdown before reactor level reaches -164".

Also, given that torus bottom pressure is 24 psig and rising with torus level at 14 feet and drywell sprays unavailable due to valve binding, using the DEOP 200-1, "Primary Containment Control," chart given at the time of the exam, 24 psig is on the Figure L, "Pressure Suppression Pressure," curve at 14 feet, and could be interpreted above or below the curve, and the examinee may or may not choose to blowdown on torus bottom pressure.

Facility Proposed Resolution:
The question grading for the exam should be changed to accept both “C” and “D” as correct answers.

With Figure L,’Pressure Suppression Pressure” just below the limit (shaded area), and indication in the question stem, that Torus Bottom Pressure is RISING (and with the Drywell Sprays inoperable, the ability to recover is gone), the SRO is allowed to make the determination that the limit will ultimately be exceeded, as allowed in the Dreseden Emergency Operating Procedure (DEOP) Bases document EPG/SAG, page B-3-3. This allows the SRO to blowdown based on not being able to stay inside Figure L. The SRO also may blowdown based on the requirements of the level leg of DEOP 100, “RPV Control.”

Subsequently, the Torus Cooling and Torus Sprays are to be stopped. DEOP 100 states when RPV water level drops to -143" and no subsystem (Detail F) is lined up with a pump running, to maximize injection with an alternate injection system (Detail E - Low Pressure Coolant Injection with Condensate Storage Tank suction).

Based on the flexibility referred to in DEOP 0010-00 discussion section, two different Operators are allowed to make a different decision based on analysis of the current situation, and both would be correct.

Therefore, distractors “C” and “D” are both correct since, based on the SRO’s determination on whether or not Figure L, “Pressure Suppression Pressure” will be exceeded, the Operating team can blowdown on both RPV water level and Torus Bottom Pressure.

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 “C” and “D” as correct answers.

For the condition of RPV water level at -193" and lowering as stated in the question stem, distractor “C” is a correct answer, since RPV blowdown is required in DEOP 100 when RPV level drops to -164." Per other conditions stated in the question stem, torus pressure does not exceed the Pressure Suppression Pressure limit in Figure L, “Pressure Suppression Pressure,” and so RPV blowdown per the Primary Containment Pressure leg of DEOP 200-1, “Primary Containment Control” is not required.

However, for the conditions stated in the question stem (i.e., torus level 14 feet and stable, torus bottom pressure at 24 psig and rising, and drywell sprays not in operation due to valve binding on both loops), one would be just below the Pressure Suppression Pressure limit in Figure L, “Pressure Suppression Pressure.” Since torus bottom pressure was at 24 psig and rising, and drywell sprays were not in operation due to valve binding on both loops, one could reasonably assume that the Pressure Suppression Pressure limit in Figure L would ultimately be exceeded. On page B-3-3 of Appendix B of the BWROG EPGs/SAGs (the DEOP Bases document), it states that if a parameter can not be maintained below a specified limit, then the appropriate action can be taken if it is anticipated that the limit will ultimately be exceeded. Based on this flexibility, distractor “D” is also a correct answer if the SRO determines based on parameter values and trends that the Pressure Suppression Pressure limit in Figure L would ultimately be exceeded.

Therefore, the answer key was modified to accept both “C” and “D” as correct answers.

WRITTEN EXAMINATIONS AND ANSWER KEYS (RO/SRO)

RO Initial Examination ADAMS Accession # ML071730551
SRO Initial Examination ADAMS Accession # ML071730559