U.S. NUCLEAR REGULATORY COMMISSION

REGION III

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Facility:	Byron Nuclear Generating Station, Units 1 and 2
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EXECUTIVE SUMMARY

Byron Nuclear Generating Station
NRC Examination Reports 50-454/98301; 50-455/98301

A licensee developed and NRC approved initial operator licensing examination was administered to five license applicants; two Reactor Operators (RO) and three Senior Reactor Operators (SRO). In addition, the examiners observed a period of routine operations in the control room.

Results:

- Only one SRO applicant passed all portions of his respective examination and was issued an SRO license.

- Two RO applicants and two SRO applicants failed portions of the examination and were denied operator licenses.

Operations Summary:

- The inspectors noted that shift turnover was effectively performed. Operator awareness of plant conditions, including response to annunciators, was considered satisfactory. Based only on a limited opportunity to observe the new briefing process, including the Heightened Level of Awareness briefing, the inspectors considered the briefing well coordinated and effective. Overall, the inspectors concluded that an appropriate level of awareness and professionalism existed in the control room. (Section O1.1)

- The examiners identified two procedure errors and relayed this information to the licensee to properly assure that the required and appropriate actions were taken to correct these procedure problems to preclude other operators from taking incorrect actions or unnecessarily delaying procedure implementation. Following the completion of the NRC examination, the licensee initiated a problem identification form (PIF) to correct the procedure problems. In conclusion, the examiners determined that the two procedure errors, one related to post emergency recovery operations, had potential weaknesses, but that these concerns alone did not constitute an inadequate procedure. (Section O3.1)

- Overall, the licensed operators, who were involved in the examination validation and as a surrogate operator to augment one applicant crew, provided excellent support to complete the verification and validation of the examination material and the administration of the initial license examination. The licensed operators were very knowledgeable and provided good insight and information to the examiners to validate the examination. The surrogate licensed operator demonstrated satisfactory performance during the dynamic simulator scenarios. Only one item to note was that the surrogate operator overreacted to the plant conditions and initiated bleed and feed when it was not considered necessary. (Section O4.1)

Examination Summary:

- The training staff's knowledge of the examination development guidelines, attention to detail during examination development, and the ability to develop technically accurate and challenging simulator scenarios, Job Performance Measures (JPM), and written examination material in accordance with the examination guidelines were considered satisfactory. Still, some changes to the submitted examination material were recommended by the examiners. Specifically, 40% of the written examination questions required some changes. The majority of these changes included editorial (grammar correction, typographical errors) and clarification or enhancement to question stems and distractors; however, five written examination questions were either rewritten or replaced. Some JPMs, and simulator scenarios were also changed to improve the level of difficulty and to better conform to the NUREG 1021 guidance. The operating test required the enhancement five JPMs, and the replacement of two JPMs. Also, some minor changes were made to two simulator scenarios. Overall, the examiners concluded that the examination as a whole was average to above average in difficulty and satisfactorily discriminated between competent and less than competent operators. (Section O5.2)

- The inspectors determined the training staff properly prepared the simulator for administration of the operating test. Although some minor simulator setup errors and equipment deficiencies did occur, no unnecessary delays in examination administration were observed. No examination compromise issues were identified. Examination security was considered excellent. (Section O5.3)

- The high failure rate and below average grades on the written examination suggested that the training program did not ensure that the applicants were adequately prepared for the examination. Several apparent knowledge deficiencies were identified through the written examination, including some understanding of system response, knowledge of operator actions, and understanding of operating procedures and technical specifications. Applicants' performance in the operating examination was generally satisfactory. However, the examiners also identified similar knowledge deficiencies during the administration of systems JPM open reference questions. Although the two RO applicants passed the systems JPM portion of the test, they both demonstrated weaknesses in answering systems related questions by incorrectly answering eight out of 20 questions. In addition, one SRO applicant displayed inadequate use and understanding of the Generating Station Emergency Plan emergency classification activity. He incorrectly classified an event due to his failure to take into account plant conditions for both units to determine the overall plant emergency status. During the simulator examination, the applicants' control and supervision of reactivity manipulations and communications, in particular the shift briefing, was considered a strength. However, the examiners also identified that some of the applicants' performance in event diagnosis, system response interpretation (understanding the abnormal effects to the plant from instrument malfunctions), and the use and understanding of operating procedures were, at times, weak and needed some improvement. (Section O5.4)

- Overall, the licensee's submittal of the post examination documents was considered satisfactory. Also, the licensee's technical review and the synopsis of identified knowledge weaknesses were considered satisfactory. The examiners accepted all five of the licensee's post written examination comments. (Section O5.5)

- On one occasion, the licensee's simulation facility experienced a minor equipment problem. The RM11 failed to properly update radiation monitoring information. The failure was for a short duration and the cause was unknown. In general, the simulation facility continued to perform well even with the problem and it did not deter the applicant's evaluations. (Section O5.6)

Reports Details

I. Operations

O1 Conduct of Operations

O1.1 Control Room Observations

a. Inspection Scope (71707)

Using Inspection Procedure 71707, "Plant Operations," inspectors observed actual control room operations. The inspectors observed routine control room activities during full power operations, performed a panel walk-down, reviewed control room logs, and questioned operators about plant and equipment status. In addition, a shift turnover, shift briefing, and a Heightened Level of Awareness (HLA) briefings were observed.

b. Observations and Findings

The inspectors observed control room activities during the examination and preparation weeks. The observed period included a shift turnover, shift briefing, and an HLA briefing. The HLA briefing concerned the planned activity to troubleshoot and potentially replace the Main Generator auto voltage regulator. Control room activities were observed to be professional during and after the shift turnover. The shift briefing following shift turnover was observed to be conducted in a new format. The briefing was conducted by the Shift Manager and held in a separate briefing room just outside the control room. The control room operators listened in and participated in the shift briefing via a telephone conference. This new format was to reduce the noise level and crowding of the control room area during shift turnover briefings.

Operator performance concerning response to plant conditions were also observed. The operators properly responded to all annunciators and took the appropriate actions based on the annunciator response procedures, as necessary. No annunciators were left unattended for any length of time. When asked by the examiners, the control room operators readily answered questions and were knowledgeable of plant conditions.

c. Conclusions

The inspectors noted that shift turnover was effectively performed. Operator awareness of plant conditions, including response to annunciators, was considered satisfactory. With just one observation of the new briefing process it may be presumptuous to make a broad and precise assessment. But, based only on a limited opportunity to observe the briefing process, including the HLA briefing, the inspectors considered the briefing well coordinated and effective. Overall, the inspectors concluded that an appropriate level of awareness and professionalism existed in the control room.

O3 Operations Procedures and Documentation

a. Inspection Scope

During both the examination validation and administration weeks, the examiners reviewed numerous procedures to ascertain procedural adequacy. Examples of procedures reviewed included, emergency operating procedures (EOP), Generating Station Emergency Plan (GSEP) procedures, annunciator response procedures, abnormal operating procedures, system operating procedures, and administrative procedures.

b. Observations and Findings

The examiners identified two procedural concerns.

1. The fuse control procedure, Byron Administrative Procedure (BAP) 350-6, "Fuse Control Program," Revision 6 dated January 17, 1997, was not adequately updated to reflect a change to another referenced procedure. During the verification of the administrative JPM for fuse replacement for Non-Like-For-Like fuses, the examiners noted that a nonexisting document was referenced in the procedure text. This determination was made after the examiners questioned the licensee of the requirement of the applicants' use of the referenced procedure (TID-E/I&C-09, "Fuse Selection Guidelines") to assist the applicant in determining a proper substitute fuse. The Fuse Control Coordinator informed the examiners that the procedure no longer existed, and that the operators were to use the existing procedural steps in BAP 350-6 or call the Fuse Control Coordinator for assistance to properly identify a substitute fuse. During administration of this particular JPM, the applicants were able to adequately follow BAP 350-6 and satisfactorily completed the JPM.

2. The Byron operating procedure for Component Cooling Water (CC) CC-14, "Post LOCA Alignment of the CC System," Revision 3, dated May 31, 1994, had erroneous information. During the verification of this procedure as a systems JPM, the examiners found that the location of one valve was incorrectly noted in the procedure. In performing this JPM the applicants demonstrated some unfamiliarity in CC valve locations. Some applicants had difficulty in readily locating the required valves to align the CC system, in particular, valve 2CC9506B, the 2A residual heat removal (RH) pump CC inlet isolation valve. In section G, "Checkoff List," of Byron operating procedure CC-14, the procedure listed the designation and the plant location of each valve referenced in the text of the procedure. Valve 2CC9506B was incorrectly designated as a valve for the 2B RH pump and listed the incorrect plant location. Although this procedure was an infrequently used procedure, it was a support procedure important for post-LOCA recovery of the plant. The erroneous information resulted in applicants' confusion in locating the valve and significantly delayed completion of the JPM task.

c. Conclusions

The examiners identified two procedure errors and relayed this information to the licensee to properly assure that the required and appropriate actions were taken to correct these procedure problems to preclude other operators from taking incorrect actions or unnecessarily delaying procedure implementation. Following the completion of the NRC examination, the licensee initiated a problem identification form (PIF) to correct the procedure problems. In conclusion, the examiners determined that the two procedure errors, one related to post emergency recovery operations, had potential weaknesses, but that these concerns alone did not constitute an inadequate procedure.

O4 Operator Knowledge and Performance

O4.1 Simulator Scenario Crew Observation

a. Inspection Scope

During the examination verification and validation week of August 31, 1998, a crew of licensed operators was assigned to support the examination effort. The licensed operators performed validation and verification of Administrative and System Job Performance Measures (JPM) and performed mitigating actions for simulator scenarios. In addition, a licensed Senior Reactor Operator (SRO) was used as a surrogate operator to augment one applicant crew during the dynamic simulator scenario examination. The performance of the licensed operators during their support effort for the examination validation and the surrogate's participation in the simulator scenarios were not generally evaluated; however, if significant performance concerns were identified, the licensee was notified of any weaknesses.

b. Observations and Findings

During validation of the dynamic simulator scenarios and JPMs, the examiners noted good participation and performance by the licensed operators. These operators were very helpful in verifying and validating the examination material. Although the examination material was processed through the licensee's quality assurance review and previously verified prior to NRC submittal, the examiners with assistance of the licensed operators were able to identify additional items needing correction and clarification prior to exam administration. For example: (1) procedure problems with the CC system JPM; (2) clarification of simulator scenarios expected actions; and (3) clarification and correction to JPM questions. In addition, the licensed operators provided assistance to help the examiners get familiarized with the simulator and the plant.

During the dynamic simulator test, one licensed operator participated as a surrogate operator in the positions of SRO and Balance of Plant (BOP) operator to augment one applicant crew. It was noted during one scenario concerning a loss of heat sink casualty, that the surrogate operator as the SRO (also, along with another license applicant in the other operating crew) directed actions to initiate bleed and feed based only on the pressurizer power operated relief valve (PORV) cycling open between 2310 and 2335 psig. Although the bleed and feed action was an option per the functional recovery procedure for loss of heat sink, the examiners determined that the plant conditions warranted the expeditious reestablishment of feedwater flow to the intact SGs, thereby restoring secondary heat sink capability, rather than a bleed and feed that degraded and challenged another fission product barrier, the primary containment. The examiners observed that the surrogate operator overreacted to the EOP's operator action summary page which noted an initiation condition into bleed and feed when pressurizer pressure was greater than or equal to 2335 psig due to loss of heat sink. However, the examiners, based on actual exam validation, scenarios expected actions, Emergency Response Guideline (ERG) based critical task, and EOP bases, determined that the plant conditions were not as irrevocable as interpreted by the SRO and did not require the immediate initiation of bleed and feed.

c. Conclusions

Overall, the licensed operators, who were involved in the examination validation and as surrogate operator to augment one applicant crew, provided excellent support to complete the verification and validation of the examination material for the administration of the initial license examination. The licensed operators were very knowledgeable and provided good insight and information to the examiners to validate the examination. The surrogate licensed operator demonstrated satisfactory performance during the dynamic simulator scenarios. Only one item to note was that the surrogate operator overreacted to the plant conditions and initiated bleed and feed when it was not considered necessary.

O5 Operator Training and Qualification

O5.1 General Comments - Initial Operator License Examination

Initial operator licensing examinations were administered to three SRO applicants and two RO applicants. The operating examination was administered by the examiners during September 15-18, 1998. The written examination was administered by the licensee's training staff with approval from and observation by the NRC examiners on September 14, 1998.

The licensee developed the initial operator license examination in accordance with guidance prescribed in NUREG 1021, "Operator Licensing Examination Standards for Power Reactors," Interim Revision 8. The examiners reviewed and approved all examination material that the licensee developed prior to its administration.

O5.2 Examination Development and Validation

a. Examination Scope

The licensee developed the examination material in accordance with the prescribed examination development guidelines. The examiners reviewed, revised, and validated the written and operating examination material during the week of August 31, 1998.

b. Observations and Findings

1. Examination Outline and Quality Assurance Review:

The licensee satisfactorily submitted the proposed examination outline based on the quantitative requirements of ES-201-2, "Examination Outline Quality Assurance Checklist," in NUREG 1021, Interim Revision 8. The examiners identified the following items based on the outline review:

- The Administrative Topics were generally performance-based JPMs, with only one area covered by two open reference type questions.
- The proposed simulator scenarios (including one spare scenario) had the appropriate number of malfunctions and had significant (realistic) initial conditions, (e.g., existing technical specification (TS) limiting condition for operation (LCO), and multiple equipment out-of-services (OOS)). Overall, the scenarios were considered good, but some minor enhancements and clarifications were needed. For example: (1) one scenario needed a malfunction after the major transient; (2) the licensee was reminded that one event could not be credited as both a normal and reactivity evolution for one applicant; and (3) sequence of and separation of events was adjusted for better scenario progression.
- The reference material submittal was satisfactory. Additional references including administrative procedures could have been useful; however, any questions relating to those procedures were readily answered by the licensee.
- In reviewing the JPM walkthrough outline, the examiners noted that a JPM question had a potential for double jeopardy with the written examination. The question covered the same topic of pressurizer PORV tailpipe temperatures and determination of temperature based on PRT conditions.
- The proposed written examination knowledge and ability (K/A) outline met the minimum requirements of NUREG 1021. A few enhancement recommendations were made to the K/A outline. For example: (1) avoid only having the minimum of one item in a K/A category tier, to prevent the possibility of not adequately covering the K/A category tier if that one question happens to be deleted due to post examination comments; (2) K/A s listed for at least two questions were changed to reflect a better K/A that was more representative of the topic covered in the question; and (3) changed one question based on K/A and topic due to potential double jeopardy with another written examination question.

The examination outline met the quantitative minimal requirements of NUREG 1021. Only minor enhancements or clarifications were needed during the outline review. Prior to and during the examination prep week, the examiners requested the licensee to upgrade the examination material based on the proposed outline. A more qualitative review of the actual examination material as submitted by the licensee was conducted and noted in subsequent sections.

2. Dynamic Simulator Examination Material

The licensee submitted seven dynamic simulator scenarios for NRC consideration. The examiners reviewed and assessed the material content of the seven scenarios and selected three simulator scenarios for examination administration with one additional scenario as a spare. The examiners noted the following regarding the dynamic simulator scenarios:

- The dynamic simulator scenario events were generally well integrated and would enable all the applicants to be evaluated on all the required competencies and rating factors.
- In general, the expected operators' actions documented in the scenarios were sufficient to provide an objective method for evaluating the applicants' performance, in accordance with NUREG 1021, ES-301, Section D.4.f.

The three scenarios submitted and chosen by the examiners required little or no changes. The minor enhancements that were made included expanding the already existing malfunctions and additionally clarify expected responses. The scenarios included more than enough malfunctions and required in-depth entry into multiple EOPs.

3. Job Performance Measure (JPM) Walkthrough Examination

The licensee submitted one set of ten system JPMs and a set of administrative JPMs specific for SRO and RO applicants. The examiners found the administrative and systems JPM material lacked some minor qualitative attributes, as described in Form ES301-3, "Operating Test Quality Assurance Checklist." This resulted in several items identified by the examiners, during examination validation and administration. The minor discrepancies included the following:

- Several JPMs needed changes including correcting cues, identifying critical steps, and adding additional cues to clarify and assist the examiners during exam administration. For example: (1) the CC alignment JPM procedure had erroneous information that delayed JPM completion; (2) notes throughout the JPM were more appropriately referenced as initiating cues to the applicants; (3) corrected a cue to note that the auxiliary feedwater pump does not crank when the start pushbutton was depressed; and (4) identified that depressing the emergency stop push button was a critical step associated with the local abnormal start of an emergency diesel generator.

- One JPM required the selective elimination of steps to appropriately expedite JPM performance. The enhancement required selecting only one bank of control rods to perform the surveillance, rather than performing the surveillance on all control rod banks.

- One systems JPM was replaced with a more discriminating operational task. The JPM to establish a shutdown electrical lineup that only required closing and opening breakers without significant synchronizing actions was replaced with a JPM to synchronize a station auxiliary transformer to a bus being fed by an emergency diesel generator following a surveillance test. The new JPM required significant synchronizing actions to match voltage, frequency, and adjust KVARS.

- Out of 20 system JPM questions reviewed, only eight questions were modified or found to have some problems. For example: (1) direct look up questions were enhanced by limiting the use of references to electrical prints only; (2) there were no specific SRO type questions, so questions involved in using technical specifications were added for SROs only; (3) enhanced several questions to ask why; and (4) clarified the expected answers to a few questions.

- One administrative JPM was replaced with a more operationally based and discriminating JPM task. A JPM that only required an RO to copy given information of a bomb threat on an approved form (K/A of 2.3) was replaced with a JPM to perform the normal and alternate offsite AC power availability weekly surveillance.

In general, the systems and administrative JPMs lacked some minor qualitative attributes that required some changes and enhancements. However, the JPMs were sufficiently corrected by the licensee to appropriately evaluate the license applicants.

4. Written Examination

The examiners reviewed all 127 questions from the originally submitted written examination. The examiners identified that some questions reviewed required some enhancements to meet the written examination question development guidance stated in NUREG 1021, Interim Revision 8, Appendix B, "Written Examination Guidelines." The licensee was informed of the potential changes and improvements needed on the written examination. During the on-site validation week, additional effort was made to ensure the required changes and enhancements were made to properly reflect the examination guidance to allow for exam administration. Following the validation week, the licensee had one week to incorporate the changes and enhancements to the written examination. The identified deficiencies included:

- Ten questions had specific determiners contrary to the guidance of NUREG 1021, Appendix B, Section C.2.m. For example: (1) selected distractors referenced specific information in the stem of the question which reflected the most correct choice; and (2) some distractors used such words as "only" and "never" when it appeared unnecessary and would potentially suggest a wrong option (per Section C.2.m(8)).

- Five questions were either rewritten or replaced. For example: (1) one question was replaced due to duplication from a certification exam; (2) two questions were replaced because they were too simple and did not sufficiently discriminate between a competent and less than competent operators; and (3) two questions gave supporting information and examined similar concepts, the contents of these questions could have potentially influenced the answer selected to other questions and was considered potential "double jeopardy" questions.

- Approximately thirty questions required some type of stem and/or distractor clarifications, enhancements, or corrections. Also, one question was identified to have multiple correct answers.

- One question was identified to have the wrong K/A reference.

- Four questions, identified by the examiners, appeared not to meet the indicated level of knowledge. The questions were categorized as comprehensive level of knowledge, but were determined to be memory type questions. The categorization was changed.

The above changes ranged from minor editorial to questions that required rewriting or replacement. These types of changes were made to approximately 40% of the 127 originally submitted written examination questions. However, only five questions had to be either rewritten or replaced to increase the discrimination factor. Furthermore, the as-administered examination, comprising a total of 127 questions, contained some deficiencies identified by the licensee during the post examination review. See Section O5.5 and Enclosure 2 of this report.

c. Conclusions

The training staff's knowledge of the examination development guidelines, attention to detail during examination development, and the ability to develop technically accurate and challenging simulator scenarios, JPMs, and written examination material in accordance with the examination guidelines were considered satisfactory. Still, some changes to the submitted examination material were recommended by the examiners. Specifically, 40% of the written examination questions required some changes. The majority of these changes included editorial (grammar correction, typographical errors) and clarification or enhancement to question stems and distractors; however, five written examination questions were either rewritten or replaced. Some JPMs, and simulator scenarios were also changed to improve the level of difficulty and to better conform to the NUREG 1021 guidance. The operating test required the enhancement five JPMs, and the replacement of two JPMs. Also, some minor changes were made to two simulator scenarios. Overall, the examiners concluded that the examination as a whole was average to above average in difficulty and satisfactorily discriminated between competent and less than competent operators.

O5.3 Examination Administration

a. Examination Scope

The examiners administered the operating examination during September 15-18, 1998. The written examination was administered on September 14, 1998, by the licensee's training staff with approval from and observation by the NRC examiners.

b. Observations and Findings

1. Written Examination

The licensee administered the written examination with the approval from and observation by the NRC. The testing facility was appropriate to assure proper examination security. Licensee's examination proctors appropriately implemented their responsibilities in accordance with the guidance of NUREG 1021, Section ES-402. All appropriate documentation for written examination administration was completed. No examination compromise issues were identified.

2. Job Performance Measure Walkthrough Examination

The examiners identified no major/significant examination administration deficiencies during the JPM walkthrough portion of the operating test. The simulator setup and coordination of administering the JPMs were performed satisfactorily.

3. Dynamic Simulator Examination

The simulation facility performed satisfactorily; however, on one occasion the simulator displayed problems associated with the radiation monitoring computer, RM11. The RM11 failed to properly update radiation monitoring information for a short period; however, it was satisfactorily resolved to allow for continued simulator operation. The temporary malfunction did not deter the applicant evaluation.

c. Conclusions

The inspectors determined the training staff properly prepared the simulator for administration of the operating test. Although some minor simulator setup errors and equipment deficiencies did occur, no unnecessary delays in examination administration were observed. No examination compromise issues were identified. Examination security was considered excellent.

O5.4 License Applicant Performance

a. Examination Scope

Operator initial licensing examinations were administered to three SRO and two RO applicants. The examiners evaluated the applicants' performance in four general areas (dynamic simulator, plant systems walkthrough, administrative JPM walkthrough, and written examination).

b. Observations and Findings

1. Dynamic Simulator Examination

The examiners observed two license applicant crews' performance during the initial license examination. Each crew consisted of three license applicants occupying the positions of SRO, RO, and BOP operator. One crew consisted of a surrogate licensed operator to perform the duties of the SRO and BOP operator positions. The dynamic simulator examination consisted of routine and emergency activities evaluated on two or three dynamic simulator scenarios on the plant specific simulation facility.

The crews performed satisfactorily during routine activities, including the performance of system testing and normal reactor power changes. Shift briefs were numerous and conducted professionally. Communications between the applicants were satisfactory to ensure that appropriate mitigating actions were performed; however, there were some identified weaknesses pertaining to procedure use and understanding, event diagnosis, and recognition of abnormal system/plant response. In general, the crews adequately performed their associated tasks identified in the simulator scenarios.

All applicants passed the dynamic simulator examination; however, there were some areas of weaknesses. The examiners had the following observations (both strengths and weaknesses) regarding the applicants' performance:

Strengths

- During all reactivity changes, the applicants ensured that a good brief was conducted, and that everyone was aware of and concurred with boration or dilution changes, turbine load changes, and control rod movements prior to the evolution.

- Communications between applicants were good. Three way communications were generally consistent; however, on some occasions operators failed to repeat back information. These infrequent communications oversights, however, did not deter from satisfactorily completing the intended task. The number and quality (details) of shift briefings held by the SROs during the simulator scenarios were also good.

Weaknesses

- Some applicants displayed unfamiliarity and weakness in recognizing and diagnosing abnormal operating trends and system operation. For example: (1) failure to recognize a pressurizer level instrument and incorrectly reported that it was an RCS leak; (2) low charging flow rate due to a faulted charging pump was incorrectly diagnosed as a charging line RCS leak; and (3) following a pressurizer pressure controller failure, applicants misdiagnosed the high charging flow and low pressurizer level due to a cooldown as a primary leak when none existed.

- Some applicants displayed weaknesses in using and understanding abnormal and emergency operating procedures (EOPs). For example: (1) the applicants' failure to timely enter an annunciator response procedure for a faulted charging pump delayed restoration of charging flow to the plant; (2) the absence of using the abnormal operating procedure following a loss of instrument air; and (3) two SROs during a loss of heat sink simulator event overreacted to plant conditions and initiated bleed and feed when it was not necessary (see Section 4.1 for additional details).

- Some applicants demonstrated weaknesses to ensure correct compliance with Technical Specifications (TS) and limiting conditions for operation (LCO) action statements. For example, the failure to recognize that with two pressurizer PORVs being inoperable meant an LCO time limit of 6 hours instead of 72 hours.

2. Job Performance Measure Walkthrough Examination

One SRO applicant failed the systems JPM and one RO applicant failed the administrative JPM sections of the examination. All other applicants passed the JPM (systems and administrative) walkthrough sections of the operating examination, but demonstrated some weaknesses. The examiners had the following observations regarding the applicants' performance:

- One SRO applicant demonstrated a lack of understanding the GSEP emergency classification in determining the required methodology of classifying emergencies. He did not classify the event correctly. He failed to take into account the plant conditions as a whole based on the casualty situation for both units. He classified the JPM as an Alert instead of a Site Area Emergency, and identified the wrong protective action recommendation.

- Also, a minor clarification was in order for filling out the nuclear accident reporting system (NARS) form. All SRO applicants noted wind speed in miles per hour, but the NARS form specifically noted that wind speed information was to be in meters per second, with exception of Quad Cities facility to annotate the information in miles per hour.

- During JPM questions, the applicants displayed knowledge weaknesses. This was based on significant number of questions answered incorrectly by the applicants. For example: (1) all five applicants were unfamiliar with circuit breaker operation and interlocks pertaining to the diesel generator; and (2) the two RO applicants incorrectly answered eight out of twenty JPM questions.

3. Written Examination

Two SRO and two RO applicants failed the written examination, only one SRO applicant passed the test. Following the examination grading, the licensee submitted a list of questions that were missed by 50% or more of the applicants. The examiners, taking into account the licensee's post examination comments, determined that there were 22 written examination questions (nine on the RO examination only, six on the SRO examination only, and seven which were common to both examinations) that a significant number (50% or more) of the applicants answered incorrectly. Of the 127 questions, three general areas were considered by the Byron training staff as potential generic knowledge weaknesses.

Licensee Identified Knowledge Weaknesses from the Written Examination:

Fundamentals
1. Knowledge of antipump relays/use of electrical prints

Systems
1. Administrative determination of equipment operability during surveillance tests
2. Knowledge of loop isolation valve interlocks
3. Knowledge of Engineering Safety Features Activation System (ESFAS) design features
4. Knowledge of mode change requirements related to ESFAS instrumentation
5. Knowledge of rod bottom alarm operation
6. Effect of power range nuclear instrument on source range nuclear instruments
7. Evaluation of electrical supply status on ESF pump operability
8. Knowledge of fuel handling building crane interlocks
9. Effect of loss of DC on CO_2 actuation system
10. Knowledge of TS limitations on equipment outages
11. Effects of failure of ESF under voltage relay

Procedures
1. Reason for power reduction to recover a misaligned rod
2. Knowledge of reactor coolant pump trip criteria

c. Conclusions

The high failure rate and below average grades on the written examination suggested that the training program did not ensure that the applicants were adequately prepared for the examination. Several apparent knowledge deficiencies were identified through the written examination, including some understanding of system response, knowledge of operator actions, and understanding of operating procedures and technical specifications. Applicants' performance in the operating examination was generally satisfactory. However, the examiners also identified similar knowledge deficiencies during the administration of systems Job Performance Measure open reference questions. Although the two RO applicants passed the systems JPM portion of the test, they both demonstrated weaknesses in answering systems related questions by incorrectly answering eight out of 20 questions. In addition, one SRO applicant displayed inadequate use and understanding of the GSEP emergency classification activity. He incorrectly classified an event due to his failure to take into account plant conditions for both units to determine the overall plant emergency status. During the simulator examination, the applicants' control and supervision of reactivity manipulations and communications, in particular the shift briefing, was considered a strength. However, the examiners also identified that some of the applicants' performance in event diagnosis, system response interpretation (understanding the abnormal effects to the plant from instrument malfunctions), and the use and understanding of operating procedures were, at times, weak and needed some improvement.

O5.5 Post Examination Activities

a. Scope

The examiners reviewed the written examination grading that was performed by the licensee in accordance with Form ES-403-1, "Written Examination Grading Quality Assurance Checklist," contained in NUREG-1021, Interim Revision 8. The examiners also reviewed the post written examination comments submitted by the licensee.

b. Observations and Findings

The post examination submittal included the necessary documentation as required per the guidance of NUREG-1021, ES-501. The licensee submitted an analysis of the written examination results, which was a list of missed questions by 50% or more of the applicants and a synopsis of identified knowledge weaknesses.

The examiners reviewed the licensee's submitted post written examination comments. All five comments were accepted by the examiners, and the written examination was graded accordingly. The licensee's comments and the NRC resolution of the comments are detailed in Enclosure 2, "Post Written Examination Facility Comments and NRC Resolution."

c. Conclusions

Overall, the licensee's submittal of the post examination documents was considered satisfactory. Also, the licensee's technical review and the synopsis of identified knowledge weaknesses were considered satisfactory. The examiners accepted all five of the licensee's post written examination comments.

O5.6 Simulator Fidelity

a. Scope

The examiners observed the operation and fidelity of the plant specific simulation facility during the performance of the operating examination. These observations were only noted as information and not considered a detailed assessment of the simulation facility.

b. Observations and Findings

During this examination, the examiners identified one potential simulator fidelity issue. The simulator discrepancy was identified as described and also noted in Enclosure 3. The simulator displayed problems associated with the radiation monitoring computer, RM11. During one scenario, the RM11 failed to properly update radiation monitoring information for a short period; however, it was satisfactorily resolved to allow for continued simulator operation. The cause was unknown.

c. Conclusions

On one occasion, the licensee's simulation facility experienced a minor equipment problem. The RM11 failed to properly update radiation monitoring information. The failure was for a short duration and the cause was unknown. In general, the simulation facility continued to perform well even with the problem and it did not deter the applicant's evaluations.

V. Management Meetings

X1 Exit Meeting Summary

The examiners conducted an exit meeting with members of licensee management on September 22, 1998, and the licensee was contacted by telephone on October 20, 1998, to inform licensee management of the examination results. The licensee acknowledged the findings presented and indicated that the materials reviewed were not considered proprietary.

X2 Post Exam Management Meeting Summary

A management meeting was conducted with members of Byron Station and ComEd Corporate management at the Region III office on October 21, 1998. The purpose of this meeting was to hear and discuss the licensee's post examination root cause evaluation concerning the high failure rate on the written examination. The licensee noted that there were precursors to this training problem, but timely and adequate corrective actions were not taken. In general, the licensee indicated the need to increase the training objectives to the higher cognitive knowledge level, that they failed to keep up with industry's rising standards, and that they failed to perform an overall comprehensive evaluation of the candidates. The licensee indicated its concern for the Byron training program and relayed intentions to change the demeanor at Byron Station. A copy of the licensee's presentation handout is briefly described below and is detailed in Enclosure 4, "Byron Nuclear Station Initial License Training Examination Meeting."

The following details the licensee's presentation of the root cause and proposed corrective action efforts.

Root Cause

Testing and evaluation did not ensure the initial license operator training (ILT) students had mastered the subjects at a high cognitive level.

The root cause was characterized by three primary causes:

1. ILT program material and examination bank did not contain sufficiently higher order learning criteria.
2. Line ownership/oversight of the ILT program was lacking.
3. Certification process that ILT candidates were ready to take the examination was flawed.

Byron Specific Corrective Actions

1. Current ILT classroom training was temporarily suspended, as of October 20, 1998.
2. The ILT examination materials and quality of ILT examination question bank will be upgraded prior to it being presented to the current ILT class.
3. Implemented mentoring process between operations and ILT candidates.
4. ILT program will include interim checkpoints to evaluate candidate performance and progress prior to certification.
5. Enhance evaluation of ILT candidate readiness for NRC examination.
6. Process of individualized remediation packages will be developed and implemented commencing with the current ILT class.
7. Effectiveness reviews of corrective actions will be performed for the current ILT class.

Nuclear Generating Group (NGG) Corporate Corrective Actions

1. Immediate interim action was completed to evaluate current ComEd ILT program candidates.
2. Thorough evaluation of other ILT programs will be completed by the end of 1998.
3. Increase training oversight of ILT program to include the accountability that site examination question banks will be upgraded to a higher cognitive level.
4. Take steps to improve candidate selection process.
5. Issue the Common Work Practice Instruction (CWPI) which includes Byron lessons learned.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

K. Graesser, Site Vice-President
W. Levis, Station Manager
B. Adams, Regulatory Assurance Manager
E. Campbell, Support Services Manager
T. Gierich, Operations Manager
B. Kouba, Engineering Manager
T. Schmidt, Training Manager
G. Teeter, Nuclear Oversight Manager
M. Snow, Maintenance Manager
J. Nuernberger, NGG Training Manager (Acting)
C. Cerovac, Braidwood Operations Training Superintendent
T. Benton, ILT Group Lead - Braidwood
M. Brown, Training Instructor - Exam Developer
S. Pettinger, Operations Training Superintendent
D. Chrzanowski, Corporate Nuclear Licensing
R. Colglazier, NRC Coordinator
E. Bendis, Shift Operations Supervisor
R. Franklin, ILT Coordinator
B. Gorwood, Executive Assistant to Site Vice-President
J. Heaton, Operations Support Manager
B. Subalusky, Corporate Training Vice-President
P. DiGiovanna, NGG Operations Training
P. Hippley, NGG Exam Developer

NRC

E. Cobey, Senior Resident Inspector
N. Hilton, Resident Inspector
B. Kemker, Resident Inspector

INSPECTION PROCEDURES USED

IP 71707, "Plant Operations"

ITEMS OPENED, CLOSED, AND DISCUSSED

NONE

LIST OF ACRONYMS USED

AC	Alternating Current
BAP	Byron Administrative Procedure
BOP	Balance of Plant
CC	Component Cooling Water
CFR	Code of Federal Regulations
CWPI	Common Work Practice Instruction
DC	Direct Current
DRS	Division of Reactor Safety
EAL	Emergency Action Level
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
ERG	Emergency Response Guideline
ES	Examination Standards
ESF	Engineering Safety Feature
ESFAS	Engineering Safety Feature Actuation System
GSEP	Generating Station Emergency Plan
HLA	Heightened Level of Awareness
ILT	Initial License Operator Training
IP	Inspection Procedure
JPM	Job Performance Measure
K/A	Knowledge and Abilities
LCO	Limiting Condition for Operation
LOCA	Loss of Coolant Accident
NARS	Nuclear Accident Reporting System
NGG	Nuclear Generating Group
NRC	Nuclear Regulator Commission
NRR	NRC Office of Nuclear Reactor Regulation
OL	Operator Licensing
OOS	Out-of-Service
PDR	Public Document Room
PIF	Problem Identification Form
PORV	Power Operated Relief Valve
PRT	Pressure Relief Tank
RCA	Radiologically Controlled Area
RCS	Reactor Coolant System
RH	Residual Heat Removal
RM	Radiation Monitor
RO	Reactor Operator
SAT	Systematic Approach to Training
SG	Steam Generator
SRO	Senior Reactor Operator
TS	Technical Specification

Facility Post Written Examination Comments and NRC Resolution

1. EXAMINATION QUESTION RO #3/SRO #3

LICENSEE COMMENT

The question asked how was a procedure change procedurally conveyed to members of the operating crew. Procedure BAP 350-2 Rev 7, "Daily Order Book," requires "licensed operators" to read and initial the Daily Orders and the Shift Manager (SM) was responsible to ensure others were notified, as necessary. The question distractor (A) was incorrect because a "memo" was not issued to all crew personnel and the SM may not be the person who places the information in the Daily Order Book. Distractor (B) was incorrect because the SM was not necessarily informed by memo and it wasn't proceduralized. Distractor (C) was incorrect because non-licensed operators were briefed by the SM, by procedure. Distractor (D) was also incorrect because the Shift Operation Supervisor does not make an announcement at the shift briefings. Therefore, there was no definitive correct answer.

Recommend deleting the question.

NRC RESOLUTION

The comment was accepted and the question was deleted. Although answer C was intended to be the correct choice, it did not specifically note "licensed" operators. Therefore, it could be interpreted that non-licensed operators were not required procedurally to review and initial the Daily Order Book and make answer C incorrect.

Question History

No changes were made to the original question as submitted.

2. EXAMINATION QUESTION SRO # 10

LICENSEE COMMENT

This question asked for the correct response based on Nuclear Operations Policy (NOP) OP-19, "Operations Control of Critical Activities," Revision 3. This revision of the NOP had not yet been implemented at Byron station. Nuclear Operations Policies are implemented through station administrative procedures. NOP OP-19 was implemented through BAP 2010-2, "Reactivity Management Controls." This procedure still contains the requirements of NOP OP-19 Revision 2. The administrative procedure, which takes precedence over the policy, stated:

> "For LCOs that lead to a reactor shutdown action statement upon expiration, reactor shutdown shall commence NO later than eight (8) hours prior to the end of the Technical Specification time clock."

This revision of the BAP and Policy does not contain the guidance to begin a shutdown as soon as it was realized that the equipment cannot be restored within the allowable outage time.

Answer (A) is nine (9) hours prior to the LCO expiration and answer (B) is eight (8) hours prior to the LCO expiration.

Recommend changing answer key to accept answer (B) as the correct answer.

NRC RESOLUTION
The comment was accepted, and the correct answer was changed from (A) to (B). Accept answer (B) as the correct choice.

Question History
No changes were made to the original question as submitted.

6. EXAMINATION QUESTION RO # 13/SRO # 17
LICENSEE COMMENT
The question stem stated that the alarm has been in for the last hour implying that the alarm was not cycling. However, the stem also stated that maintenance was troubleshooting the alarm which means that it may be cycling. The stem contradicts itself.

Per BAP 380-2, "Handling of Main Control Board and Radwaste Panel Annunciator Alarms," section C.1 noted that alarms shall be silenced, acknowledged, and reset as soon as practical. Per section C.2, it noted that the SER points must be evaluated. Also, per section C.5, it noted that an alarm which requires continuous acknowledgment MAY be silenced without acknowledgment.

The question asked which of the following actions would be appropriate. This BAP gives the operator the option of silencing and acknowledging every cycle of the alarm or just silencing the alarm without acknowledgment.

Recommend changing the answer key to accept both (A) and (C) as correct answers.

NRC RESOLUTION
The comment was accepted, and both (A) and (C) was noted as correct answers.

Question History:
No changes were made to the original question as submitted.

7. EXAMINATION QUESTION SRO # 59
LICENSEE COMMENT
The question asked for the largest reactivity addition from a 20 step rod withdrawal. Typical fundamental rules say that the largest addition from a rod withdrawal would be from an EOL condition (rods worth more due to less competition) and from a higher power condition (higher flux at the tip of the rod increases the rod worth due to less dense moderator).

Since rod positions and ro worth curves were not provided to the candidates they are required to answer this question based on fundamental knowledge.

The original question was asking for the highest reactor power. The original answer was based on the highest power being reached at BOL 4% power since the negative reactivity effects from MTC would be smaller and take a longer time to counteract the reactivity addition from the rod pull.

Recommend changing the answer key to accept choice (D) as the correct answer.

NRC RESOLUTION
The comment was accepted, and the correct answer was changed from (A) to (D). Accept the answer (D) as the correct choice.

Question History
No changes were made to the original question as submitted. The original question referred to in the licensee's comments was the original question as submitted to the licensee by the contractor. The stem wording change of power to reactivity was made prior to examination submittal to the NRC.

8. EXAMINATION QUESTION RO # 96/SRO # 96
LICENSEE COMMENT
The question stated that RCS activity was increasing due to corrosion product activation and to identify the effects of placing the cation bed demineralizer in service.

The cation bed will remove Lithium from the RCS causing a reductio in pH. The lowering of pH could lead to an increase in the production of corrosion products so the use of the cation bed must be carefully controlled. While it will cause the activity level to decrease as soon as it is placed in service (Answer B) it should not be used due to the effects on Lithium and pH (Answer A). This is the reason that BOA PRI-4, "Abnormal Primary Chemistry," has the operators check the effectiveness of the mixed bed demineralizer and does not place the cation ben id service for high activity.

Recommend changing the answer key to accept both (A) and (B) as correct answers.

NRC RESOLUTION
The comment was accepted, and both (A) and (B) was noted as correct answers.

Question History
No changes were made to the original question as submitted.

Enclosure 3

SIMULATION FACILITY REPORT

Facility Licensee: Byron

Facility Licensee Dockets No: 50-454; 50-455

Operating Tests Administered: September 15-18, 1998

This form is to be used only to report observations. These observations do not constitute audit or inspection findings and are not, without further verification and review, indicative of a noncompliance with 10 CFR 55.45(b). These observations do not affect NRC certification or approval of the simulation facility other than to provide information that may be used in future evaluations. No licensee action is required in response to these observations.

While conducting the simulator portion of the operating test, the following items were observed (if none, so state):

ITEM	DESCRIPTION
RM11 display computer	The simulator displayed problems associated with the radiation monitoring computer, RM11. The RM11 failed to properly update radiation monitoring information for a short period; however, it was satisfactorily resolved to allow for continued simulator operation. The cause was unknown

Byron Nuclear Station Initial License Training Examination Meeting

October 21, 1998

ComEd
A Unicom Company

Agenda

Introduction	W. Levis
Root Cause Team Findings	K. Rach
Byron Corrective Actions	T. Gierich
NGG Corrective Actions	R. Holbrook
Conclusions	K. Graesser

Introduction

William Levis

Byron Station Manager

Introduction

- Examination Results Clearly Unacceptable
- Byron/NGG Management Responded Swiftly With Immediate Corrective Action
- Purpose of Meeting is to Communicate Root Cause and Corrective Actions
- Lessons Learned From Byron Site Issue Being Acted On by Other ComEd Sites

Root Cause Team Findings

Ken Rach

NGG Operations

Training Superintendent

ComEd
A Unicom Company

Root Cause Team

- Investigative Team of Select NGG Personnel Immediately Formed to Determine Root Cause
- Team Charter Included:
  - Review of Relevant Training Data and Records
  - Conduct of Extensive Interviews
  - Completion of Root Cause Analysis
- Team Presented Findings to ComEd NGG Senior Management on September 28, 1998

Root Cause

- Testing and Evaluation Did Not Ensure The Initial License Operator Training (ILT) Students Had Mastered the Subjects at a High Cognitive Level

Root Cause Team Findings

- Root Cause Was Characterized By Three Primary Causes
  - ILT Program Material and Examination Bank Did Not Contain Sufficiently Higher Order Learning Criteria
  - Line Ownership/Oversight of the ILT Program Was Lacking
  - Certification Process That ILT Candidates Were Ready to Take Examination Was Flawed

Primary Cause #1
ILT Program Material

- Training Materials Contain Limited Number of Higher Cognitive Level Objectives
- Limited Number of Examination Bank Questions at a Similar Higher Cognitive Level
- Limited Number of Instructors Are Trained on the Standards of NUREG-1021, Rev. 8

Primary Cause #1 (cont.) ILT Program Material

- Examinations Administered to ILT Candidates Did Not Contain the Same Cognitive Level As That Required for the NRC Examination
- Deficiencies Were Previously Identified, However, Corrective Actions to Address Were Not Timely
- No Structured NGG Approach to Upgrade Cognitive Level of All Examination Banks at Stations

Primary Cause #2
Line Ownership/Oversight

- Overconfidence Based on Past Success
- Limited Management Involvement in ILT Classroom Training
  - Line Management Did Not Effectively Respond to Indications of Candidates' Performance Weaknesses
  - Line Management Did Not Act On the Need for Increasing Standards
- Corporate Training Oversight Not Effective in Escalating Program Deficiencies at Byron

Primary Cause #3
Certification of ILT Candidates

- Failure of Five Candidates During Initial Certification Examination Did Not Result in Adequate Remediation of Candidate Weaknesses
- Second Certification Examination Was Not at the Same Cognitive Level as First Examination
- Comprehensive and Objective Review of Candidate Performance Was Not Performed Prior to NRC Examination

Contributing Factor

- Student Selection Process for Entry Into ILT Program

Byron Station Corrective Actions

Tom Gierich

Byron Station Operations Manager

Byron Specific Actions

- Current ILT Classroom Training Temporarily Suspended
- Upgrade ILT Examination Materials and Quality of ILT Examination Question Bank Prior to Being Presented to Current ILT Class
  - Train Byron Operations Instructors on Increased Standards
  - Increase Learning Objectives to a Higher Cognitive Level
  - Increase Cognitive Level of Questions Commensurate With Current Industry Standards
  - Increase Amount of Review Material and Length of Examinations Throughout the Program

Byron Specific Actions (cont.)

- Implement Mentoring Process Between Operations and ILT Candidates (Complete)
  - Byron Operations Department Assigned Mentors to Each Individual in New ILT Class and Assigned an Overall Class Sponsor
- ILT Program Will Include Interim Checkpoints to Evaluate Candidate Performance and Progress Prior to Certification (10/31/98)

Byron Specific Actions (cont.)

- Enhanced Evaluation of ILT Candidate Readiness For NRC Examination (10/31/98)
  - Comprehensive Review of All Examinations Throughout Program
  - Solicitation of Input From Various Training and Station Personnel Based on Observed Candidate Performance and Behaviors
- Process of Individualized Remediation Packages Will Be Developed and Implemented Commencing With the Current ILT Class (10/31/98)
  - Include Structured Training to Address Individual Problems
  - Address Generic ILT Class Weaknesses

Byron Specific Actions (cont.)

- Effectiveness Reviews of Corrective Actions Will Be Performed for the Current ILT Class
  - At the End of Systems Phase of Training
  - At the End of All Phases of Training

NGG Corrective Actions

Robert Holbrook

NGG Training Manager

NGG Corrective Actions

- Immediate Interim Action Was Completed to Evaluate Current ComEd ILT Program Candidates
- Thorough Evaluation of Other ILT Programs Will Be Completed by the End of 1998

NGG Corrective Actions (cont.)

- Increase Training Oversight of ILT Program to Include the Accountability That Site Examination Question Banks Will Be Upgraded to a Higher Cognitive Level
- Take Steps to Improve Candidate Selection Process
- Issue the Common Work Practice Instruction (CWPI) Which Includes Byron Lessons Learned

Recent Dresden Insights

- NRC Examination Candidates Passed a Management Review Board
- ILT Class Mentor
- ILT Program Contains Higher Order Objectives and Examination Questions
- Certification Examination More Representative of NRC Examination
- NRC Preparation Phase Includes Individualized Remediation Plans

CWPI Enhancements

- Specify the Number of Tests for Course Length
- Require Operations Department Mentors
- Review Examination Material and Increase Comprehension Level As Appropriate
- Set Minimum Limits of Periodic Performance Reviews
- Add Formal Checkpoints for Line Management Review
- Establish Formal Review and Approval Board
- Require Structured Remediation After Certification Examination

Conclusions

Kenneth Graesser

Byron Station Site Vice President

U.S. Nuclear Regulatory Commission
Site-Specific
Written Examination

Applicant Information

Name: MASTER EXAMINATION	Region: III
Date: SEPTEMBER 14, 1998	Facility/Unit: Byron Units 1 and 2
License Level: RO	Reactor Type: W
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. The passing grade requires a final grade of at least 80.00 percent. Examination papers will be collected four hours after the examination starts.

Applicant Certification

All work done on this examination is my own. I have neither given nor received aid.

______________________ Applicant's Signature

Results

Examination Value	99 ~~100~~ Points
Applicant's Score	______ Points
Applicant's Grade	______ Percent

SUBJECTIVE SCORE INSTRUCTOR USE ONLY

100	90	80	70	60
50	40	30	20	10
9	8	7	6	5
4	3	2	1	0

IMPORTANT

- MAKE DARK MARKS
- ERASE COMPLETELY TO CHANGE
- EXAMPLE A B ● D E

TO USE SUBJECTIVE SCORE FEATURE

- Mark total possible subjective points
- Only one mark per line on key
- 150 points maximum

EXAMPLE OF STUDENT SCORE

PART 1

SCANTRON® FORM NO. 888-E

TO REORDER CALL 1-800-826-7196

NAME

SUBJECT

TEST NO.

DATE

HOUR

TEST RECORD

PART 1

PART 2

TOTAL

← FEED THIS DIRECTION

	(T)	(F)			KEY
	%	●	3		●
1	A	B	●	D	E
2	A	B	●	D	E
~~3~~	~~A~~	~~B~~	~~●~~	~~D~~	~~E~~ Delete ✓
4	●	B	C	D	E
5	A	B	●	D	E
6	●	B	C	D	E
7	A	●	C	D	E
8	●	B	C	D	E
9	A	●	C	D	E
10	A	B	C	●	E
11	A	B	C	●	E
12	A	●	C	D	E
13	A	B	●	D	E A also correct (A or C) ✓
14	●	B	C	D	E
15	A	B	●	D	E
16	A	B	●	D	E
17	A	●	C	D	E
18	A	B	C	●	E
19	A	B	C	●	E
20	A	B	●	D	E
21	A	●	C	D	E
22	A	B	●	D	E
23	●	B	C	D	E
24	A	B	C	●	E
25	A	●	C	D	E
26	●	B	C	D	E
27	A	B	C	●	E
28	A	●	C	D	E
29	●	B	C	D	E
30	A	B	●	D	E
31	A	B	C	●	E
32	A	B	●	D	E
33	A	●	C	D	E
34	A	B	●	D	E
35	A	B	C	●	E
36	A	B	C	●	E
37	A	●	C	D	E
38	A	B	●	D	E
39	A	B	●	D	E
40	A	B	C	●	E
41	A	B	C	●	E
42	A	B	●	D	E
43	A	●	C	D	E
44	●	B	C	D	E
45	A	B	C	●	E
46	A	B	●	D	E
47	A	B	●	D	E
48	A	B	●	D	E
49	A	B	C	●	E
50	A	●	C	D	E

BYRON

1998 INITIAL LICENSE TRAINING

Description: NRC REACTOR OPERATOR FINAL EXAMINATION

Total Points: 100.0 Points Received: ______

NRC APPROVED

Name/Date: ANSWER KEY ___/___/___

Instructions

1. Use black ink only on all portions of the exam package EXCEPT for the scantron answer selections.
2. Print your name and date in the space provided above.
3. If you have any questions or need clarification during the examination, notify the proctor.
4. Conversation during the examination is prohibited except when asking for question clarification from the proctor.
5. Cheating on the examination will result in failure of the examination and may result in further disciplinary action.
6. Use only #2 pencil to mark your selection on this exam sheet.
7. Completely darken the selected answer. If you make a mistake, completely erase the darkened selection.
8. Ensure you do not skip a question or answer which would place you out of sequence.
9. Do not place any extraneous marks on this exam sheet.
10. You have 4.0 hour(s) to complete this exam.
11. Prior to handing in your exam, verify that you have transferred your answers to this scantron sheet properly.

I have neither given, received, or observed any aid or information regarding this exam prior to or during its administration that could compromise this exam's integrity. I also understand my obligation to report any exam compromise by others prior, during, or subsequent to the exam administration.

______________________ ___/___/___

signature date

SUBJECTIVE SCORE
INSTRUCTOR USE ONLY

100	90	80	70	60
50	40	30	20	10
9	8	7	6	5
4	3	2	1	0

IMPORTANT

- MAKE DARK MARKS
- ERASE COMPLETELY TO CHANGE
- EXAMPLE: A B ● D E

TO USE SUBJECTIVE SCORE FEATURE:

- Mark total possible subjective points
- Only one mark per line on key
- 150 points maximum

EXAMPLE OF STUDENT SCORE:

PART 2

NAME:

SUBJECT: TEST NO.:

DATE: HOUR:

TEST RECORD

PART 1:

PART 2:

TOTAL:

	(T)	(F)			KEY
	%	●	3	●	●
51	A	B	●	D	E
52	A	●	C	D	E
53	●	B	C	D	E
54	A	●	C	D	E
55	A	B	●	D	E
56	A	●	C	D	E
57	A	B	●	D	E
58	A	B	C	●	E
59	A	●	C	D	E
60	A	●	C	D	E
61	A	●	C	D	E
62	A	B	C	●	E
63	A	●	C	D	E
64	A	B	C	●	E
65	●	B	C	D	E
66	●	B	C	D	E
67	A	B	●	D	E
68	●	B	C	D	E
69	A	B	●	D	E
70	●	B	C	D	E
71	●	B	C	D	E
72	●	B	C	D	E
73	A	B	C	●	E
74	A	B	C	●	E
75	A	B	●	D	E
76	A	●	C	D	E
77	A	●	C	D	E
78	A	B	C	●	E
79	A	B	C	●	E
80	A	●	C	D	E
81	A	B	C	●	E
82	A	●	C	D	E
83	●	B	C	D	E
84	A	B	C	●	E
85	●	B	C	D	E
86	●	B	C	D	E
87	●	B	C	D	E
88	A	●	C	D	E
89	A	●	C	D	E
90	A	●	C	D	E
91	A	●	C	D	E
92	A	B	C	●	E
93	A	B	●	D	E
94	●	B	C	D	E
95	A	●	C	D	E
96	A	●	C	D	E
97	●	B	C	D	E
98	A	B	●	D	E
99	●	B	C	D	E
100	A	B	●	D	E

50
A is also correct (A or B) ✓

FEED THIS DIRECTION

Reactor Operator Answer Key

1 .c
2 .c
~~3 .c~~ Deleted ✓
4 .a
5 .c
6 .a
7 .b
8 .a
9 .b
10 .d
11 .d
12 .b
13 .c A is also correct ✓
14 .a
15 .c
16 .c
17 .b
18 .d
19 .d
20 .c
21 .b
22 .c
23 .a
24 .d
25 .b
26 .a
27 .d
28 .b
29 .a
30 .c
31 .d
32 .c
33 .b
34 .c
35 .d
36 .d
37 .b
38 .c
39 .c
40 .d
41 .d
42 .c
43 .b
44 .a
45 .d
46 .c
47 .c
48 .c
49 .d
50 .b

Reactor Operator Answer Key

51 .c
52 .b
53 .a
54 .b
55 .c
56 .b
57 .c
58 .d
59 .b
60 .b
61 .b
62 .d
63 .b
64 .d
65 .a
66 .a
67 .c
68 .a
69 .c
70 .a
71 .a
72 .a
73 .d
74 .d
75 .c
76 .b
77 .b
78 .d
79 .d
80 .b
81 .d
82 .b
83 .a
84 .d
85 .a
86 .a
87 .a
88 .b
89 .b
90 .b
91 .b
92 .d
93 .c
94 .a
95 .b
96 .b A is also correct ✓
97 .a
98 .c
99 .a
100 .c

NOTE: DISREGARD THE PAGE NUMBERS AT THE BOTTOM ON EACH PAGE.

THE QUESTION NUMBER ORDER IS AT THE TOP LEFT ON EACH PAGE.

Question 1 Evaluation of requirement for "active" license

An "Active" licensed NSO (original license obtained in 1996) worked the following schedule at Byron:

- 9/4 - 0700 to 1500 as Unit 1 NSO
- 9/7 - 0700 to 1500 as Unit 2 NSO
- 9/8 - 0700 to 1500 as Unit 2 NSO
- 9/9 - 0700 to 1200 as Unit 2 NSO and 1200 to 1500 as WEC NSO
- 9/10 - 0700 to 1500 as WEC NSO
- 9/11 - 0700 to 1500 as Unit 1 NSO
- 9/14 - 1500 to 2200 as Unit 2 NSO
- 9/12 - 1500 to 2200 as Unit 2 NSO

The NSO...

a. meets the requirements for maintaining his/her license active for the next quarter.

b. needs to work an additional FOUR hour shift to maintain his/her license active for the next quarter.

c. needs to work an additional EIGHT hour shift to maintain his/her license active for the next quarter.

d. needs to work TWO additional EIGHT hour shifts to maintain his/her license active for the next quarter.

Answer C **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.1 Conduct of Operations

2.1.1 Knowledge of conduct of operations requirements. 3.7 3.8

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Operators' "License" Information	Policy 400-12	I.A.1	2	71	
OPERATING SHIFT TURNOVER AND RELIEF	BAP 335-1	C.1.e	3	21	
Administrative Procedures - BAP 335-1	335-1r4	I.C.1.e	6	4	1

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 2 Direction of NLO personnel

The following conditions on Unit 1:

- Reactor power 45%
- 1A and 1C Feedwater pumps are operating
- FW PUMP TURB BRNG OIL LEVEL HIGH LOW annunciator (1-16-D3) alarms and the SER monitor indicates a low level.
- An EA is dispatched and confirms a low level exists.

In performing actions to correct the condition (per BOP TO-08 "Filling a Turbine Feed Pump Oil Reservoir"), what is the normal relationship between the US, the NSO and the EA?

a. The US will direct the EA's activities, but will inform the Unit NSO before the job commences.

b. The US will direct the EA's activities, and need NOT inform the Unit NSO unless unit controls are affected.

c. The Unit NSO will direct the EA's activities, but will inform the US before the job commences.

d. The Unit NSO will direct the EA's activities, and need NOT inform the US unless unit load is affected.

Answer c **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.1 Conduct of Operations

2.1.1 Knowledge of conduct of operations requirements. 3.7 3.8

Explanation of Answer

Reference Title	**Facility Reference Number**	**Section**	**Page**	**Revisio**	**L. O.**
Conduct Of Operations	BAP 300-01	C.2.b.4) & C.4.a.2)	14 & 20	14	
ADMINISTRATIVE PROCEDURES LESSON 300-1 & 340-1	BAP 300-1 and 340-1	I.C.2.b).4)	34-35	1	7

Material Required for Examination **Number(s)** n

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 3 Operating Daily Orders

How is a procedure change, which significantly changes normal processes, procedurally conveyed to members of the operating crew?

a. The SM places the applicable information in the Daily Order Book, and issues an additional memo to all crew personnel that is initialed.

b. The SM is informed by memo of the addition to the Daily Order Book, and makes an announcement of the addition during the shift briefing.

c. The SOS places the applicable information in the Daily Order Book, and the individual operator is responsible for reviewing and initialing the Daily Order.

d. The SOS places the applicable information in the Daily Order Book, and makes an announcement of the addition during the shift briefing.

deleted

Answer C **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.1 Conduct of Operations

2.1.2 Knowledge of operator responsibilities during all modes of plant operation. 3.0 4.0

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
DAILY ORDER BOOK	BAP 350-2	C.3.h	1	7	
ADMINISTRATIVE PROCEDURES LESSON Selected Administrative Procedures	Selected Administrative Procedures	VII.B.1.h, 4	54	3	25

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type Comment

Question 4 Procedure required usage

An example of a licensed operator evolution that can be performed without having a procedure in hand is...

a. Adjusting rod position following a boration for delta-I control.

b. Starting the 1A Heater Drain Pump.

c. Placing excess letdown in service.

d. Latching and rolling up the main turbine following surveillance trip test.

Answer a **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.1 Conduct of Operations

2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation. 3.9 4.0

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Use Of Procedures For Operating Department	BAP 340-1	C.1.a.1), C.1.d	3, 6-7	9	
ADMINISTRATIVE PROCEDURES LESSON BAP 300-1 and 340-1	BAP 300-1 and 340-1	II.C.1.e	62	4	3

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type Comment

Question 5 Use of electrical prints

Assuming an auto-close signal is continuously present in the circuit for the 1A SI pump, which contact will be maintained open in order to prevent the starting relay (SR) from attempting repeated breaker closures onto a faulted bus?

(E 1-4030-SI01 is provided for use.)

a. LC SW

b. 52/b

c. Y

d. LS

Answer c **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.1 Conduct of Operations

2.1.24 Ability to obtain and interpret station electrical and mechanical drawings. 2.8 3.1

Explanation of Answer "Y" is an antipump relay that when prevented from energizing interrupts the circuit that energizes the START relay in the AUTO start circuit

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Schematic Diagram - Safety Injection Pump 1A 1SI01P	6E 1-4030-SI01				
Print Reading	Chap 3		34		2.c

Material Required for Examination E 1-4030-SI01

Question Source: Facility Exam Bank **Question Modification Method:** Editorially Modified

uestion Source Comments: Braidwood requal bank

Comment Type **Comment**

Question 6 MOV tagout

An operator is preparing an OOS that designates 1CC685, RCP Thermal Barrier CC Return CNMT Isolation valve, as an isolation point.

Vhat is the acceptability of using this isolation point?

The OOS is...

a. acceptable if the MOV is tagged at its control switch, power supply and valve handwheel.

b. acceptable if the MOV is tagged at its control switch, power supply and a blocking device is placed on the valve.

c. NOT acceptable because the MOV fails to meet isolation requirements.

d. NOT acceptable because the valve fails open on a loss of power.

Answer a **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98
Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC
2.2 Equipment Control
2.2.13 Knowledge of tagging and clearance procedures. 3.6 3.8

Explanation of Answer Valve is MOV and requirements include tagging control switch, electrical power supply and local handwheel if accessible.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
OUT OF SERVICE PROCESS	BAP 330-1	C.4.c NOTE	27	28	
Selected Administrative Procedures - BAP 330-1	Selected Administrative Procedures	I.E.8	26	3	2

ıaterial Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 7 RCS level discrepancy during refueling

The following conditions exist for Unit 1 in preparation for head removal:

- Unit shutdown and cooldown initiated 120 hours ago
- Lowering of RCS level to the reactor vessel flange is underway
- RCS temperature - 95°
- RCS level Control Room indicators: 1LI-RY046 - 401' 0"
  1LI-RY049 - 402' 1"
- RH loop 1A in operation with "normal" indications

What is the appropriate action for these conditions?

a. The lowering of RCS level can continue after verifying appropriate amount of water removed.

b. The level change must be stopped until the cause for the level discrepancy is determined.

c. The running RHR pump shall be immediately stopped to prevent cavitation.

d. The available SI Pump is immediately aligned for hot leg injection and shall be started.

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.2 Equipment Control

2.2.26 Knowledge of refueling administrative requirements. 2.5 3.7

Explanation of Answer With any level discrepancy, the reason for the discrepancy must be determined before further draining can continue.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
REACTOR COOLANT SYSTEM DRAIN	BOP RC-4a	Step 20 CAUTION	11	14	
Residual Heat Removal System	Chp 18	III.C.1.c & III.C.2.C	44 & 48	2	9.c

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: Zion exam bank

Comment Type	Comment
NRC	Significant Industry Event -

Question 8 RO duties in Control Room during refueling

What is a responsibility of the NSO during refueling operations in the main control room?

- a. Checking source range counts while a fuel assembly is being placed in the core.
- b. Verifying direct phone communication with the Fuel Handling Supervisor once per day during fuel movement.
- c. Maintaining a 1/M plot while reloading fuel during a core shuffle.
- d. Updating the Control Room tag board per the Nuclear Component Transfer List on an hourly basis.

Answer a **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.2 Equipment Control

2.2.32 Knowledge of RO duties in the control room during fuel handling such as alarms from fuel handling area, communication with fuel storage facility, systems operated from the control room in support of fueling operations, and supporting instrumentation. 3.5 3.3

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
ADMINISTRATIVE CONTROL DURING REFUELING	BAP 370-3	C.1.c	4	18	
ADMINISTRATIVE PROCEDURES LESSON BAP 2010-2	BAP 2010-2, Reactivity Management	II.F.2	36	1	1.d
ADMINISTRATIVE PROCEDURES LESSON BAP 370-3	Administrative Control During Refueling, BAP 370-3	C.1.c.2	8	2	13

'aterial Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 9 Radiation exposure determination

An operator has the following exposure history this year until today:

Deep Dose Equivalent (DDE)	-	210 mrem
Committed Effective Dose Equivalent (CEDE)	-	45 mrem
Shallow Dose Equivalent (SDE)	-	33 mrem
Committed Dose Equivalent (CDE)	-	28 mrem

Today the operator was required to make two entries into containment:

Entry 1: Gamma dose - 52 mrem; Neutron dose - 24 mrem
Entry 2: Gamma dose - 124 mrem

How much radiation exposure is available to the operator if he has to make additional entries?

His available margin based on the routine Administrative Exposure Control Levels is...

a. 100 mrem for that day; 2484 mrem for the year.

b. 100 mrem for that day; 2545 mrem for the year.

c. 124 mrem for that day; 2569 mrem for the year.

d. 124 mrem for that day; 2614 mrem for the year.

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.3 Radiation Control

2.3.1 Knowledge of 10 CFR: 20 and related facility radiation control requirements. 2.6 3.0

Explanation of Answer Limits are 300 mrem routine DDE/Day and 3000 mrem routine cumulative TEDE/year. C. Neutron rad not counted for daily & yearly; A. All counted for yearly; d. previous DDE+CEDE only counted for year.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
EXPOSURE REVIEW AND AUTHORIZATION	BRP 5300-2	F.1.a & F.5.a	4 & 9	6	
Radiation Protection	Chapter 3			1	4

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 10 Fuel Handling Accident Response

The following conditions exist on Unit 1:

- Refueling operations in progress
- A HIGH alarm received on radiation monitor 1RE-AR012, Containment Fuel Handling Incident

When should the NSO initiate action and what action should he/she take from the control room?

Indication of a fuel handling accident is considered when a...

a. report is received from personnel in containment. The operator starts the containment charcoal filter fans.

b. report is received from personnel in containment. The operator actuates Unit 1 CNMT evacuation alarm.

c. corresponding rise is indicated on monitor 1RE-AR011. The operator starts the containment charcoal filter fans.

d. corresponding rise is indicated on monitor 1RE-AR011. The operator actuates Unit 1 CNMT evacuation alarm.

Answer d **Exam Level** R **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.3 Radiation Control

2.3.10 Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure. 2.9 3.3

xplanation of
nswer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Fuel Handling Emergency, U-1	1BOA REF-1	B. Symptoms & step 1.a	1 & 2	54	
Fuel Handling Emergency	1BOA REF-1	I.E, II.B & I.C.1	1-3	1	2, 6

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 11 Performance of Status Trees/Function Restoration

The following conditions exist on Unit 1:

- A reactor trip has occurred and both reactor trip breakers are verified open
- The turbine has tripped
- BEP-0 "Reactor Trip OR Safety Injection" has been entered.
- BUS 141 ALIVE light is NOT lit with bus voltage at ZERO volts
- BUS 142 ALIVE light is lit with bus voltage at 4149 volts.

Which of the following describes the action(s) the operators is/are required to take?

a. Check SI status.

b. Turn on the synchroscope and manually close ACB 1412, SAT 142-1 feed breaker.

c. Manually start 1A D/G and verify ACB 1413, D/G output breaker, closes.

d. Initiate actions of BOA ELEC-3 and then check SI status.

Answer d **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.4 Emergency Procedures / Plan

2.4.16 Knowledge of EOP implementation hierarchy and coordination with other support procedures. 3.0 4.0

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Reactor Trip Or Safety Injection	BEP-0	Step 3 RNO(Step 1-4 IMMEDIATE ACTION)	4	29	
Use Of Procedures For Operating Department	BAP 340-1	C.4.a	12	9	
ADMINISTRATIVE PROCEDURES LESSON BAP 300-1 and 340-1	BAP 300-1 and 340-1	II.C.4.a, b & f	78, 80, 84	4	3

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 12 Applicability of EOP Foldout Page

Following transition to BEP-1 "Loss of Reactor Or Secondary Coolant", the US refers to the Operator Action Summary, and directs the operator to Cold Leg Recirculation Switchover Criterion. Which of the following describes the complete set of procedures for which the Transfer to Cold Leg Recirculation requirements are applicable?

(NOTE: The following procedures are in the E-1 or CA-1 series:
BEP-1 "Loss Of Reactor Or Secondary Coolant"
BEP ES-1.1 "SI Termination"
BEP ES-1.2 "Post-LOCA Cooldown And Depressurization"
BEP ES-1.3 "Transfer To Cold Leg Recirculation"
BEP ES-1.4 "Transfer To Hot Leg Recirculation"
BCA-1.1 "Loss Of Emergency Coolant Recirculation"
BCA-1.2 "LOCA Outside Containment)

a. BEP-1, BCA-1.1 and BCA-1.2 procedures.

b. BEP-1, BEP ES-1.1 and ES-1.2 procedures.

c. BEP-1 and BEP ES-1.2 procedures.

d. BEP-1 procedure.

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.4 Emergency Procedures / Plan

2.4.20 Knowledge of operational implications of EOP warnings, cautions, and notes. 3.3 4.0

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Loss Of Reactor Or Secondary Coolant	BEP-1	OAS	3	19	
USE OF EMERGENCY OPERATING PROCEDURES	BAP 340-1	G.4	9	5	
ADMINISTRATIVE PROCEDURES LESSON BAP 300-1 and 340-1	BAP 300-1 and 340-1	II.C.3.d	78	4	7

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 13 Identification of inoperable CR annunciators

The following conditions exist on Unit 1:

- Reactor trip breakers status - OPEN
- RCS Tave - 557°F
- Pzr pressure - 2235 psig

Annunciator RCFC VIBRATION HI (1-3-C5) has been in alarm for the past hour due to vibration condition while maintenance troubleshoots the vibration probe on RCFC 1C.

Which of the following actions is appropriate for this alarm window?

a. The alarm should be acknowledged for each actuation and the SER monitored for valid alarm inputs.

b. The alarm should be acknowledged for each actuation and operators stationed locally at each RCFC to monitor vibration.

c. The alarm should have been silenced without acknowledgement with US permission and the SER monitored for valid alarm inputs.

d. The alarm should have been silenced without acknowledgement with US permission and operators stationed locally at each RCFC to monitor vibration.

Answer c or A HP **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.4 Emergency Procedures / Plan

.4.31 Knowledge of annunciators alarms and indications, and use of the response instructions. 3.3 3.4

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
RCFC VIBRATION HI	BAR 1-3-C5	E.	1	51	
HANDLING OF MAIN CONTROL BOARD AND RADWASTE PANEL ANNUNCIATOR ALARMS	BAP 380-2	C.5	3	3	
Selected Administrative Procedures - BAP 300-2	Selected Administrative Procedures	XII.C.5	64	3	32

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 14 Effect of Xenon Transient & compensation

A feed pump trip occurred resulting in a rapid power reduction on Unit 1. Power was reduced from 100% steady-state conditions using a combination of rods and boration.

The following conditions exist for Unit 1 following stabilization:

- Reactor Power - 60%
- Delta-I target value - +2.0
- Control Bank D position - 160 steps withdrawn
- Tave - 572°F
- Delta-I - -10.5%
- Core Age - MOL

What actions will be required to maintain the current power level and maintain Delta-I within its normal operating band over the next FIVE hours?

a. Boration and control rod withdrawal, followed by dilution.

b. Boration and control rod insertion, followed by dilution.

c. Dilution and control rod withdrawal, followed by boration

d. Dilution and control rod insertion, followed by boration.

Answer a **Exam Level** B **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

001 Control Rod Drive System

A2. Ability to (a) predict the impacts of the following on the Control Rod Drive System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:

A2.06 Effects of transient xenon on reactivity 3.4 3.7

Explanation of Answer With delta-I near the negative limit of the band, boration would be initiated to allow rod withdrawal and hence shifting of power production toward positive delta-I (power shift toward top of core). Later as Xenon (neutron poison) builds in, dilution will be initiated to maintain power level

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
DELTA I CONSIDERATIONS AND GENERIC COASTDOWN GUIDELINES	BGP 100-8	E.4 & 5; F.3	2; 5-6	5	
BGP 100-8, I Considerations	BGP 100-8, I Considerations	II.B.5 & III.A.3	6 & 12-13	1	1
Curve Book - BYRON UNIT ONE CYCLE NINE Xenon Worth vs Time After Shutdown	BCB-1	Figure 8.c		19	

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 15 Application of DC Hold

A problem with the rod control system requires checking several rod bank circuits. The affected power cabinet repairs are to be made by supplying power from the DC hold supply cabinet.

Which statement describes the proper operation for DC Hold and the associated response in the event of a reactor trip?

a. ONE control rod bank group can be placed on DC HOLD, and these rods will drop if the controls are taken to OFF at the DC Hold cabinet.

b. ONE control rod bank group and ONE shutdown bank group can be placed on DC HOLD, and these rods will drop if the controls are taken to OFF at the DC Hold cabinet.

c. ONE control rod bank group can be placed on DC HOLD, and these rods will automatically drop.

d. ONE control rod bank group and ONE shutdown bank group can be placed on DC HOLD, and these rods will automatically drop.

Answer C **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

001 Control Rod Drive System

K1. Knowledge of the physical connections and/or cause-effect relationships between Control Rod Drive System and the following:

K1.03 CRDM 3.4 3.6

Explanation of Answer Only one GROUP of control rods can be placed on HOLD at a time in order to ensure the rods are held without falling. Opening the reactor trip breakers interrupts the power to the DC Hold cabinet.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
ROD DRIVE PLACEMENT IN D.C. HOLD	BOP RD-6	NOTE step F.4	1	1	
Rod Control System	Chapter 28	I.C.4 & II.A.6.a	8 & 44	1	9

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 16 Relationship of levels during refueling operations

The following conditions exist for Unit 2:

- Mode 5
- RCS is draining to Pzr level of 40%
- IM calibrations have been completed for LT-048, Refuel Cavity level, in preparation for further draining

- LI-462 indicates 40%

What is the relationship of Pzr level instrument LI-459 as compared to LI-048?

a. LI-459 and LI-048 will be offscale high.

b. LI-048 will be just onscale and LI-459 will be offscale low.

c. LI-459 will read higher than LI-462 and LI-048 will just be onscale.

d. LI-048 will be offscale high and LI-459 will read lower than LI-462.

Answer C **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

002 Reactor Coolant System

A1. Ability to predict and/or monitor changes in parameters associated with operating the Reactor Coolant System controls including:

A1.11 Relative level indications in the RWST, the refueling cavity, the PZR and the reactor vessel during preparation for refueling 2.7 3.2

Explanation of Answer LI-462 is the cold calibrated Pzr level instrument and will read lower (but more accurately) than the hot calibrated level instruments (LI-459/460/461) at lower RCS temperatures. The refueling cavity level instrument just comes onscale at 40% Pzr level.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
REACTOR COOLANT SYSTEM DRAIN OR MAINTENANCE	BOP RC-4a	E.4	4	14	
Chp 14, Pressurizer (RY)	Chp 14	II.B.1.a.5).b).(2) & III.B.4.b, c	37, 74-76	3	19, 25
BGP 100-6, Refueling Outage	BGP 100-6	II.A.7	24-26	2	5

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 17 RCS leak Detection Systems

The following conditions exist for Unit 1:

- Reactor power - 100%
- RCS activity is elevated, but below Technical Specification (ITS) levels
- Pzr pressure - 2225 psig
- Pzr level - 44%
- Leak rate - 10 gpm

In an attempt to isolate the leakage past the PORV, the Block Valve 1RY8000B was taken to close. The Block Valve failed to close and the operator placed 1RY456 in the CLOSE position. When conditions stabilize:

- Reactor power - 100%
- Pzr pressure - 2228 psig
- Pzr level - 44%

How would the operator be able to tell if the PORV has closed?

a. Position lights for PCV-456 showing CLOSE indication.

b. Verify stable VCT level indication.

c. Level change in RCDT.

d. Lower readings for containment radiation monitors RE-0011A/0012A.

Answer b **Exam Level** R **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

ler: Plant Systems **RO Group:** 2 **SRO Group:** 2

J02 Reactor Coolant System

A3. Ability to monitor automatic operations of the Reactor Coolant System including:

A3.01 Reactor coolant leak detection system 3.7 3.9

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
REACTOR COOLANT SYSTEM WATER INVENTORY BALANCE 72 HOUR SURVEILLANCE	2BOS 4.6.2.1.d-1	F.10	7	8	
Abnormal Operating Procedure, Excessive Primary Plant Leakage	BOA PRI-1	III.D	42	2	3
Chp 15a - Chemical and Volume Control System	Chp 15a	II.A.2.b.5) & III.B.2.c	45-46, 80	2	7 & 12.a

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 18 Use of Loop Isolation Valves

The following conditions exist on Unit 2:

- RCS Loop C is isolated for maintenance
- RCS Loop A had been isolated for maintenance
- RCS Loop A Hot Leg Stop Isolation Valve (LSIV) was opened at 1001
- RCS Loop A Bypass Stop Valve was opened at 1005 with relief line flow of 115 gpm verified
- RCS Loop A Cold Leg LSIV is closed
- RCS temperature - 110°F
- RCS Hot Leg Loop temperatures - 108°F (A); 119°F (B); 110°F (C); 125°F (D)
- RCS Cold Leg Loop temperatures - 103°F (A); 108°F (B); 90°F (C); 115°F (D)
- S/G levels (Narrow Range) - 20% (A); 30% (B); 15% (C); 32% (D)

What will occur when the operator takes the control switch for MOV-RC8002A (RCS Loop A Cold Leg LSIV) to OPEN at 1409?

The valve...

a. will travel fully open with NO automatic actuations.

b. will travel fully open, and the AFW pumps get a start signal.

c. remains closed because the temperature [illegible] interlock remains active.

d. remains closed because the timer interlock is still active.

Answer d **Exam Level** R **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

002 Reactor Coolant System

K4. Knowledge of Reactor Coolant System design feature(s) and or interlock(s) which provide for the following:

K4.09 Operation of loop isolation valves. 3.2 3.2

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Simplified RCS	RC-1	Valve Interlocks 1		3	
Chp 12, Reactor Coolant System	Chp 12	III.C.1.c	58	2	7

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: Question 30/35 on Braidwood 1996 NRC exam is about LSIV interlocks. Premise and answers significantly different. Question asked about interlock for opening HL LSIV.

Comment Type **Comment**

Question 19 RCP and Pzr spray operations

The following Unit 1 conditions exist:

- RCS temperature (Average CETC) - 140°F
- RCS pressure - 365 psig
- A bubble has just been drawn in the Pressurizer
- All loops are filled and vented
- Preparations are in progress to start the first RCP for continuous run
- 1C RCP is started

What is the effect on RCS pressure control?

RCS pressure will increase and...

a. both Pzr Sprays will function normally for Pzr pressure control.

b. manual cycling of the Pzr heaters will be required for Pzr pressure control.

c. PORV RY456 will open on high pressure from high pressure bistable PB456E.

d. Pzr spray will deliver minimal spray flow for Pzr pressure control.

Answer d **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

003 Reactor Coolant Pump System

A1. Ability to predict and/or monitor changes in parameters associated with operating the Reactor Coolant Pump System controls including:

A1.06 PZR spray flow 2.9 3.1

Explanation of
nswer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
PLANT SHUTDOWN AND COOLDOWN	1BGP 100-5	F.55	31	27	
Pressurizer	RY-1	Schematic		2	
Chp 14, Pressurizer (RY)	Chp 14	II.A.4.c.1)		3	8.c, d

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 20 RCP Breaker & interlocks

The following conditions exist on Unit 1:

- Reactor power 26%
- Pzr pressure - 2235 psig
- Pzr level - 35%

RCP 1A breaker trips due to sensed undervoltage from bus 157. What is expected as a result of the trip of the RCP?

a. The reactor will trip due to the open RCP breaker.

b. The reactor will trip due to RCS loop low flow condition.

c. The reactor will be manually tripped by the operator.

d. A normal plant shutdown will be initiated.

Answer C **Exam Level** R **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

003 Reactor Coolant Pump System

K2. Knowledge of electrical power supplies to the following:

K2.01 RCPS 3.1 3.1

Explanation of Answer No AUTO trip is expected due to power < P-8. Administrative direction for a RCP trip in these conditions is a manual trip will be initiated.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Chp 13, Reactor Coolant Pump	Chp 13	III.C.3	60-61	2	9.e
Chapter 60b/Reactor Protection System	Chapter 60b	II.B.3.c		2	4

.aterial Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 21 Charging & letdown flows (including seal injection)

The following conditions exist on Unit 1:

- Reactor power - 100%
- PZR pressure - 2235 psig
- PZR level - 44% stable
- CV121 - In MANUAL
- CVCS letdown - Isolated due to leak in Letdown Hx
- CVCS Excess Letdown - In service with maximum flow of 20 gpm
- RCP seal injection - 1A CV pump aligned to all RCPs
- RCP seal leakoff flow - 3 gpm (1A); 3.5 gpm (1B); 3 gpm (1C); 2.5 gpm (1D)

What flow is indicated on Charging Header Flow indicator, FI-121?

a. 20 gpm

b. 32 gpm

c. 55 gpm

d. 67 gpm

Answer b **Exam Level** R **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

004 Chemical and Volume Control System

A3. Ability to monitor automatic operations of the Chemical and Volume Control System including:

A3.11 Charging/letdown 3.6 3.4

Explanation of Answer FI-121 indicates total charging flow (chg header + RCP seal flow, less Chg pump recirc (60 gpm)). Flow balance - Letdown: 20 + 12 = 32 & Chg: 0 + 20 + 12 = 32.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
CVCS	CV-1	Schematic		2	
Chp 15a - Chemical and Volume Control System	Chp 15a	II.A.2.a	39-40	2	9, 10.a

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 22 Calculation of dilution

The following conditions exist on Unit 2:

- Unit is in MODE 5
- Unit burnup is 5700 EFPH in Cycle 7
- SDM - 1.3% DeltaK/K
- RCS pressure - 400 psig
- RCS average temperature - 195°F
- RCS boron concentration - 1006 ppm
- Differential boron worth - -10.75 pcm/ppm
- PZR level - 32.3%
- SR NIS countrate - 10 cps , BOTH channels are stable at "background levels"
- An inadvertent dilution at 70 gpm begins at 1300 hours

Assuming NO operator action is taken and PZR level remains constant over the time period, when would the HIGH FLUX AT SHUTDOWN alarm actuate?

a. NO action, because BDPS will actuate prior to receiving the annunciator.

b. 1430 hours.

c. 1505 hours.

d. 1734 hours.

Answer C **Exam Level** B **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

004 Chemical and Volume Control System

A4. Ability to manually operate and/or monitor in the control room:

4.07 Boration/dilution 3.9 3.7

Explanation of Answer Dilution rate dc/dt = (500)(C)(Y)/M where M is the RCS mass at the given temperature (200°F). M = 745,537 lbm; C = 1006 ppm (given); Y=70 gpm (given). The dil rate = 47.2 ppm/hr. HIGH FLUX AT SHUTDOWN alarms at 5 x background = 50 cps. With K1= 0.987 dK/K (p1=-0.01317), calculate K2 = 0.9974 DKr/K (p2=-0.00261). Delta-P = 1056 pcm. 1056/-10.75=-98.2 ppm change required. Therefore the time required for the 98 ppm dilution is 98.2/47.2 = 2 hours 5 min. Difference in time based on use of Nomograph for RCS at normal pressure & temperature conditions. 'd' would only occur if count rate doubled in any 10 minute period. Assuming count rate increase is linear, for given dilution rate counts would change by 3 every 10 minutes.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Boron Dilution Rate Nomograph	Byron Curve Book, Unit 2	BCB-2 Figure 12		1	
Chp 15b, Reactor Makeup Control System	Chp 15b	III.B.3.d & e	57-59	2	7.c
Chp 31, Source Range Nuclear Instrumentation	Chp 31	II.B.2	42	1	10.a, 11.a

Material Required for Examination CURVE BOOK CBC-2 Figure 12. And GFES Equation Sheet

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 23 Boron mixing

The following conditions exist on Unit 1:

- Reactor power was 95% prior to the event
- A turbine runback resulted in rod insertion with control rods in AUTOMATIC
- Annunciator ROD BANK LO-2 INSERTION LIMIT (1-10-A6) is lit

The operators initiated an emergency boration per BOA PRI-2 "Emergency Boration" and have verified control rods are now withdrawing. Why does the operator energize the Pzr Backup Heaters?

This action...

a. ensures Pzr boron concentration equalization with RCS by increasing normal spray flow.

b. counteracts RCS cooldown due to the boration by the additional heat from the backup heaters.

c. prevents loss of Pzr level by increasing the volume of fluid maintained in the Pzr.

d. guarantees adequate subcooling margin is maintained by raising the saturation temperature of the Pzr.

Answer a **Exam Level** R **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

004 Chemical and Volume Control System

K6. Knowledge of the of the effect of a loss or malfunction on the following will have on the Chemical and Volume Control System:

K6.01 Spray/heater combination in PZR to assure uniform boron concentration 3.1 3.3

xplanation of nswer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Emergency Boration	BOA PRI-2	Step 2	3	55B	
Abnormal Operating Procedures Emergency Boration	BOA PRI-2	2	7	1	6

Number(s) n

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 24 **Topic** Recirc interties to SI Pumps & CV Pumps

The following conditions exist on Unit 1:

- A LOCA has occurred
- Actions of 1BEP ES-1.3, 'Transfer To Cold Leg Recirculation, have been completed.
- During alignment, 1CV8804A, RH HX to CENT CHG Pumps Isolation Valve, failed to open and could NOT be manually opened.

What is the status of the ECCS system?

a. The RHR discharge headers are cross-tied with only RHR Pump 1B running and supplying suction to the SI pumps and Centrifugal Charging pumps from the B train connection.

b. The RHR discharge headers are cross-tied with both RHR pumps running and supplying suction to the SI pumps only from the B train connection. The Centrifugal Charging pumps are stopped.

c. RHR Pump 1B is discharging through the B Train cold leg injection headers and supplying suction to the SI Pumps. RHR Pump 1A and the Centrifugal Charging pumps are stopped.

d. RHR Pump 1B is discharging through the B Train cold leg injection headers and supplying suction to the SI pumps and Centrifugal Charging pumps. RHR Pump 1A is discharging through the A Train cold leg injection headers.

Answer d **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 3 **SRO Group:** 3

005 Residual Heat Removal System

K1. Knowledge of the physical connections and/or cause-effect relationships between Residual Heat Removal System and the following:

1.12 Safeguard pumps 3.1 3.4

Explanation of Answer CL recirc lineup has any ONE running RHR pump aligned to provide suction path to all other ECCS pumps (SI & CENT CHG). The discharge headers between RH trains are required to be separate so that the ONE running RH pump does not operate in runout condition.

Reference Title	Facility Reference Number	Section	Page Number(s)	Revision	L. O.
Transfer To Cold Leg Recirculation	1BEP ES-1.3	steps 3, 5, 6	3-5	1A WOG-1B	
Abnormal Operating Procedures - Loss of Reactor Or Secondary Coolant	BEP 1, BEP ES-1.1 - 1.4	BEP ES-1.3, 3-5	3-5	2	10
Chp 58, Emergency Core Cooling System	Chp 58	II.C.2, III.D.1.b	48-50, 74	2	8

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 25 Failure of Hx Outlet Valve

The following conditions exist on Unit 1:

- Unit is in MODE 4 during cooldown per 1BGP 100-5 following unit shutdown 38 hours ago
- RCS temperature - 340°F
- RCS pressure - 345 psig
- PZR level - 33%
- RHR pump 1A is operating in Shutdown Cooling mode
- RH-618 A Hx Bypass Flow Control Valve is in MAN at 3000 gpm
- RH-606 A HX Flow Control Valve controller demand is at 20%
- CV-128 RHR Ltdn Flow Contr Valve demand is at 100%
- PCV-CV-131 is in AUTOMATIC set to maintain 350 psig

A signal failure from the controller causes RH-606 to go fully closed. What is the system response to this failure without operator action?

a. PCV-131 will throttle open due to lower RH discharge pressure.

b. RCS pressure will increase due to RCS heatup.

c. Pressurizer level will decrease due to increased letdown flow.

d. RH-610 will throttle open due to lower RH flow.

Answer b **Exam Level** R **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 3 **SRO Group:** 3

005 Residual Heat Removal System

K4. Knowledge of Residual Heat Removal System design feature(s) and or interlock(s) which provide for the following:

K4.10 Control of RHR heat exchanger outlet flow 3.1 3.1

Explanation of Answer RCS pressure will rise as fluid temperature increases due to loss of cooling flow through HX. IF flow decreases system pressure downstream may decrease this will cause PCV-131 to throttle close in an attempt to raise pressure

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
RHR Cooldown	RH-1	Schematic		1	
Chp 18, Residual Heat Removal System	Chp 18	II.A.6 & 7, II.D.5 Appx A IV.A	24-26, 34, 68	2	4.g & h, 9.a
Chp 15a - Chemical and Volume Control System	Chp 15a	II.A.1.g.7)	23	2	5.c

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 26 Systems response to SI

The following conditions exist on Unit 1:

- A plant heatup is underway
- MODE 3 has just been entered
- RCS pressure 450 psig

SI Accumulator 1C was drained below required level during the outage for repair work. System configuration has NOT allowed refilling the Accumulator until now. The SI Accumulator line is being flushed in accordance with BOP SI-14 "SI Accumulator Fill Line Flush" (Valve lineup includes: 1SI-8964, SI Test Lines to Radwaste Isolation Valve, and SI-8888, SI Pps to Accumulator Fill Valve, are open. 1SI 8821A, SI Pump to Cold Leg Isolation Valve, and 1SI 8802A, SI to Hot Leg 1A & 1D Isol valve are closed). SI pump 1A running. During the flushing, an inadvertent SI signal is generated.

What is the status of the ECCS based on the current alignment without operator action?

a. 1B SI pump injection flow is directed to the RCS cold legs and 1A SI pump flow is directed to the Accumulator 1C fill line flush.

b. 1A SI pump flow is directed to the 1C Accumulator fill line flush and 1B SI Pumps is in PULL-TO-LOCK.

c. BOTH SI pump flows are directed to the RCS cold legs and to the Accumulator 1C fill line flush.

d. BOTH SI pump flows are directed to the RCS cold legs ONLY.

Answer a **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

06 Emergency Core Cooling System

A2. Ability to (a) predict the impacts of the following on the Emergency Core Cooling System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:

A2.13 Inadvertent SIS actuation 3.9 4.2

Explanation of Answer SI pumps are operable; SI8821A remains closed; SI8888 and SI8964 remain open.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
PLANT HEATUP	1BGP 100-1	F.49	38	29	
SI ACCUMULATOR FILL LINE FLUSH	BOP SI-14	F	2-3	4	
Chp 58, Emergency Core Cooling System	Chp 58	III.C.8	68	2	6.d & 9.b

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 27 10CFR50.46 Design Criteria

To meet the 10CFR50.46 criteria, the ECCS System is designed such that under accident conditions it will maintain...

a. total hydrogen production from zirconium-water reaction below maximum value of 5%.

b. maximum fuel temperature at the inside surface of the cladding NOT to exceed 2000°F.

c. the core at least 5% dK/K shutdown to prevent an inadvertent return to criticality.

d. fuel clad oxidation less than 17% of total clad thickness anywhere within the core.

Answer d **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

006 Emergency Core Cooling System

K3. Knowledge of the effect that a loss or malfunction of the Emergency Core Cooling System will have on the following:

K3.02 Fuel 4.3 4.4

Explanation of Answer Third selection addresses design criteria for reactivity control per ITS.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
	10CFR50	47			
Chp 61,Engineered Safety Features	Chp 61	I.C.3	10	2	3
Chp 58, Emergency Core Cooling System	Chp 58	I.D.1	10	2	2

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments:

Comment Type **Comment**

Question 28 Evaluation of flow ECCS pumps

The following conditions exist on Unit 1:

- A LOCA has occurred
- 1B SI pump trips and cannot be restarted
- Transfer to Cold Leg recirculation is required
- RCS pressure is approximately 50 psig

What is the approximate total SI pump flow indicated on the main control board and how will this value change following transfer of BOTH trains of ECCS to cold leg recirculation?

	Total Flow	Flow Change
a.	400 gpm	Decrease
b.	650 gpm	Increase
c.	800 gpm	Decrease
d.	1300 gpm	Increase

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

006 Emergency Core Cooling System

K6. Knowledge of the of the effect of a loss or malfunction on the following will have on the Emergency Core Cooling System:

K6.03 Safety Injection Pumps 3.6 3.9

Explanation of Answer SI pump design values provide for 400 gpm flow per pump @ 1200 psig and 650 gpm @ 800 psig (or less). The flow from the pumps increases since the RH pumps are now providing a suction pressure of approximately 250 psig to the pumps instead of the lower pressure (30 psig or less) provided by the head associated with RWST level.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Chp 58, Emergency Core Cooling System	Chp 58	II.A.3.c & 5.c & d	22, 31	3	3, 8.a

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 29 PRT conditions causing alarm/response

During shift turnover for Unit 1, the NSO notes the following parameters:

RCS Tave - 566.5°F
Pzr pressure - 2235 psig
Pzr level - 38.3%
PRT pressure - 4 psig
PRT level - 74%
PRT temperature - 98°F

One hour later when annunciator 1-12-A7, PRT LEVEL HIGH LOW alarmed, the NSO notes the following parameters:

RCS Tave - 566.2°F
Pzr pressure - 2233 psig
Pzr level - 38%
PRT pressure - 5.9 psig
PRT level - 81%
PRT temperature - 96°F

What condition resulted in the change in parameters?

a. PRT PW Supply Inside Cnmt Isol Valve RY-8030 opened.

b. PRT to GW Comp Isol Valve RY-469 failed closed.

c. CVCS letdown relief valve CV-8117 lifted.

d. PORV RY-455A opened and reclosed.

Answer a **Exam Level** R **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98
Tier: Plant Systems **RO Group:** 3 **SRO Group:** 3
007 Pressurizer Relief Tank/Quench Tank System
2.4 Emergency Procedures / Plan
2.4.50 Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. 3.3 3.3

Explanation of Answer The only input provided that would give a level increase and a temperature decrease is the makeup from PW.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
PRESSURIZER RELIEF TANK FILLING AND VENTING	BOP RY-3			1	
PRT LEVEL HIGH LOW	BAR 1-12-A7	D	1	51	
Chp 14, Pressurizer (RY)	Chp 14	II.A.7.e-g, II.D.3		3	13, 14

Material Required for Examination

Question Source: New **Question Modification Method:** Editorially Modified

Question Source Comments: Ginna 9/90 NRC Exam

Comment Type **Comment**

Question 30 Determination of effect of valve positioning

Unit 1 is operating at 100% power in MOL conditions. All systems are functioning normally with rod control in manual.

Vhat is the effect on plant operations if instrument air supplied to the CVCS letdown Hx component cooling water outlet valve, TCV-CC-130A is lost?

TCV-CC-130A goes fully...

a. shut and reactor power decreases due to boration in the CVCS demineralizers.

b. shut and the CVCS demineralizers are automatically bypassed on temperature signal.

c. open and reactor power increases due to deboration in the CVCS demineralizers.

d. open and the CVCS demineralizers are automatically bypassed on temperature signal.

Answer C **Exam Level** R **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 3 **SRO Group:** 3

008 Component Cooling Water System

A2. Ability to (a) predict the impacts of the following on the Component Cooling Water System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:

A2.05 Effect of loss of instrument and control air on the position of the CCW valves that are air operated 3.3 3.5

Explanation of Answer The CVCS letdown flow is overcooled and will give up boron to the resins in the CVCS demins (until a new equilibrium value of boron reached in demins).

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Loss of Instrument Air	1BOA SEC-4	Table A; Component Cooling	6	52	
hp 15a - Chemical and Volume Control System	Chp 15a	II.A.1.h.5), II.D.1.b.10)	23, 69-71	2	5.a
Abnormal Operating Procedures BOA SEC-4 Loss of Instrument Air	BOA SEC-4	Table A CC	12	3	

Material Required for Examination

Question Source: New **Question Modification Method:**

Number(s) n

Question Source Comments:

Comment Type **Comment**

Question 31 Spray using Normal and Aux Spray

What are the parameters and values used by the operator to ensure the temperature difference between the PZR and the spray fluid are within the specified limit(s) in the PRESSURE AND TEMPERATURE LIMIT REPORT when initiating PZR spray?

a. For normal spray, the difference between RCS hot leg loop temperature and PZR vapor space temperature limit is 50°F, and for aux spray, the difference between Regenerative Hx charging inlet temperature and PZR vapor space limit is 320°F.

b. For normal spray, the difference between RCS cold leg loop temperature and PZR vapor space temperature limit is 50°F, and for aux spray, the difference between Regenerative Hx charging outlet temperature and PZR vapor space limit is 320°F.

c. For normal spray, the difference between RCS hot leg loop temperature and PZR vapor space temperature limit is 320°F, and for aux spray, the difference between Regenerative Hx charging inlet temperature and PZR vapor space limit is 320°F.

d. For normal spray, the difference between RCS cold leg loop temperature and PZR vapor space temperature limit is 320°F, and for aux spray, the difference between Regenerative Hx charging outlet temperature and PZR vapor space limit is 320°F.

Answer d **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

010 Pressurizer Pressure Control System

A1. Ability to predict and/or monitor changes in parameters associated with operating the Pressurizer Pressure Control System controls including:

A1.08 Spray nozzle DT 3.2 3.3

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
PLANT HEATUP	1BGP 100-1	E.3.d	11	29	
PRESSURIZER TEMPERATURE LIMIT SURVEILLANCE PRESSURIZER SPRAY WATER TEMPERATURE DIFFERENTIAL LIMIT SURVEILLANCE	1BOS 4.9.2-1 / 1BOS 4.9.2-2	7-10, 2-5	3, 2	3 / 1	
Chp 14, Pressurizer (RY)	Chp 14	III.A.6	70	3	26.b

Material Required for Examination

Question Source: New **Question Modification Method:** Significantly Modified

Question Source Comments: Kewaunee 2/94 NRC Exam

Comment Type **Comment**

Question 32 Evaluation of Pzr conditions

The following conditions exist on Unit 1:

- A load reject from 100% power has occurred
- Reactor power - 80%
- Pzr level - 56%
- Pzr vapor temperature - 655°F
- Pzr liquid temperature - 653°F
- RCS Tave - 578°F

What is the current status of the Pressurizer based on given conditions?

a. Backup and proportional heaters are fully on.

b. Proportional heaters are modulated on.

c. Pzr spray valves have modulated open.

d. Pzr spray valves and Pzr PORVs are open.

Answer C **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

010 Pressurizer Pressure Control System

K5. Knowledge of the operational implications of the following concepts as they apply to the Pressurizer Pressure Control System:

K5.01 Determination of condition of fluid in PZR, using steam tables 3.5 4.0

Explanation of Answer At 655°F, saturation pressure is 2272 psig. At this pressure, with current PZR level deviation <5% of program level(53%), the sprays are the only component "on".

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
zr Pressure Control	RY-2	Pzr Pressure Setpoints		3	
Chp 14, Pressurizer (RY)	Chp 14	II.C.1.c.4)	56	3	7 & 8
Steam tables		Saturation Table			

Material Required for Examination Steam Tables

Question Source: Facility Exam Bank **Question Modification Method:** Concept Used

Question Source Comments: Braidwood 1997 NRC exam

Comment Type Comment

Question 33 Pzr Level Reactor Trip

The following conditions exist on Unit 1 with all controls in normal lineup:

- Reactor power - 30% stable
- RCS Tave - 564.5°F
- Pzr pressure - 2230 psig
- Pzr level - 36% (LI-459), 37% (LI-460), 36% (LI-461)
- Pzr LVL CONT CH SELECT is in 459/460 position

The pressurizer level controller 1LK-459 output fails low. What automatic actions will occur as a result of this failure assuming NO operator action taken?

a. Pzr level will NOT change due to LT-460 being the controlling channel

b. The reactor will trip on high Pzr level due to letdown isolation.

c. Pzr level will control at 25% due to low output from the controller.

d. Pzr level will control at 60% due to low output from the controller.

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

011 Pressurizer Level Control System

K1. Knowledge of the physical connections and/or cause-effect relationships between Pressurizer Level Control System and the following:

K1 04 RPS 3.8 3.9

Explanation of Answer NOTE that this failure is like the failure of the controlling level channel high in that charging flow falls to minimum. At 17% level, letdown isolates charging continues at minimum (52 gpm) and Pzr level rises to high level trip setpoint.).

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Pzr Level Control	RY-3	Schematic, Pressurizer Level Setpoints		2	
Chp 14, Pressurizer (RY)	Chp 14	III.C.3.g	86-88	3	21

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments:

Comment Type **Comment**

Question 34 Operation of BOTH Bypass Trip Breakers

The following conditions exist on Unit 1:

- Mode 3 NOT NOP with reactor trip breakers (RTA and RTB) closed
- Testing of reactor trip bypass breakers underway
- Reactor bypass breaker B (BYB) is racked in and closed
- An operator begins to perform test with reactor bypass breaker A (BYA).

What occurs as the operator operates the breaker BYA?

When reactor bypass breaker BYA is...

a. locally closed, breaker BYB will trip. RTA and RTB remain closed.

b. racked in to the CONNECT position, breaker BYB will trip. RTA and RTB remain closed.

c. locally closed, all reactor trip and bypass breakers will trip.

d. is racked in to the CONNECT position, all reactor trip and bypass breakers will trip.

Answer C **Exam Level** R **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

012 Reactor Protection System

A3. Ability to monitor automatic operations of the Reactor Protection System including:

A3.07 Trip breakers 4.0 4.0

Explanation of Answer Closure of the second BYB results in SPSS generating a GENERAL WARNING on both trains which would open all trip and bypass breakers.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
ESF Setpoints	EF-2	Rx Trip Bypass Brkr trips		5	
Chp 60a - SSPS	Chp 60a			1	8

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments:

Comment Type **Comment**

Question 35 Input that can be bypass & condition

The following conditions exist on Unit 2:

- Unit shutdown is in progress
- Reactor power - 20%
- RCS Tave - 562°F
- Pzr pressure - 2235 psig
- Pzr level - 32%
- First stage turbine pressure channel PT-506 fails high

What affect does this failure have on operations as unit shutdown is continued, if NO action is taken for the channel failure?

a. At 10% power, the reactor will trip if the SR MAN BLOCK switches are taken to RESET.

b. At 9% power, the reactor will trip if an RCP trips.

c. At 7% power, the reactor will trip if the TURBINE TRIP pushbuttons are depressed.

d. At 5% power, the reactor will be manually tripped as required during a normal shutdown.

Answer d **Exam Level** B **Cognitive Level** Comprehension **Facility** **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

012 Reactor Protection System

A4. Ability to manually operate and/or monitor in the control room:

A4.03 Channel blocks and bypasses 3.6 3.6

Explanation of Answer PT-506 failure results in P13 interlock NOT clearing when turbine power falls below 10%. This also feeds into P7 "AT POWER TRIPS" interlock also remains active. Trips affected: 1) 2 loop loss of flow, 2) Pzr low press, 3) Pzr high level, 4) RCP brkr open, 5) RCP UV, 6) RCP UF. At 10% power, the SR NIS should still be auto blocked by P-10 (active). The turbine is normally tripped from ~65 Mwe at 5% power per BGP.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
POWER DESCENSION	1BGP 100-4	NOTE at step F.27	15	16	
ESF Setpoints	EF-1	Permissive, Reactor Trip		4	
Chapter 60b/Reactor Protection System	Chapter 60b	II.C.3.d, II.C.8	36, 42	2	4

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 36 OTdT inputs & effect of changes

The following conditions exist on Unit 1:

- Power range NIS reading - 100%
- Tcold - 553°F
- Thot - 608°F
- RCS total flow - 372,000 gpm
- Pzr pressure - 2245 psig
- Pzr level - 69%

How does the setpoint for Over Temperature Delta-T (OTdT) change when a listed parameter is changed? (Consider each change individually)

The setpoint...

a. increases if Power range NIS output rises to 102%.

b. decreases if total reactor flow increases to 370,000 gpm.

c. increases if pressurizer pressure decreases to 2235 psig.

d. decreases if the Thot rises to 612°F.

Answer d **Exam Level** R **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

012 Reactor Protection System

K5. Knowledge of the operational implications of the following concepts as they apply to the Reactor Protection System:

K5.01 DNB 3.3 3.8

Explanation of Answer Number(s) a - NIS input is only for exceeding +/- delta-I; b - Flow affects when DNB occurs, but is NOT an input to OTdT;
n
c - Pressurize rise increases OTdT. Thot input to dT power for OTdT determination

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
ESF Setpoints	EF-2	OTdT		5	
Chapter 60b/Reactor Protection System	Chapter 60b	I.B.3.c.2), II.B.3.b	8, 20-21	2	3 ,4

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 37 CNMT Spray/Phase B

A heatup is in progress on Unit 1.

At 0700, the following conditions are noted:

- RCS pressure - 1750 psig
- RCS temperature - 480°F
- S/G pressures - 565 psig

At 0730, the following conditions are noted:

- RCS pressure - 1850 psig
- RCS temperature - 485°F
- S/G pressures - 593 psig

If the current trend continues, the FIRST event that the operators should expect to see is the...

a. Pzr PORVs open

b. MSIVs close

c. Pzr sprays open.

d. S/G PORVs open

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

013 Engineered Safety Features Actuation System

4. Knowledge of Engineered Safety Features Actuation System design feature(s) and or interlock(s) which provide for the following:

K4.03 Main Steam Isolation System 3.9 4.4

Explanation of Answer RCS (Pzr) pressure rises above the P-11setpoint (1930 psig), which provides permissive for SI/Main Steam Line Isolation on low S/G pressure, and S/G pressure is less than MSL isolations.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
ESF Setpoints	EF-1	Permissive		3	
ESF Setpoints CS MCB Indications	EF-2	Steamline Isolation Signals		5	
Chp 61,Engineered Safety Features	Chp 61	II.C.17	41	2	7

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 38 FW Isolation - P14

The following conditions exist on Unit 2:

- RCS temperature - 340°F
- RCS pressure - 900 psig
- All MSIVs for the S/Gs are closed
- The MSIV Bypass valves are open
- The FW-035s, Feedwater Tempering Isolation Valves, are open
- The FW-034s, Feedwater Tempering Flow Control Valves, are closed (opened periodically for level control)
- Feedwater pump 2C is reset and latched on turning gear
- The Start Up Feedwater pump is running

The level in the S/G 2B rises to 90%. How is the plant affected?

a. No actuation occurs because of the position of the MSIVs.

b. The 2C Feedwater pump and Start Up Feedwater pump trip.

c. The 2C Feedwater pump trips and FW-035 valves close.

d. The 2C Feedwater pump and Start Up Feedwater pump trip, the FW-035 valves close, and the MSIV Bypass valves close.

Answer c **Exam Level** R **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

013 Engineered Safety Features Actuation System

K4. Knowledge of Engineered Safety Features Actuation System design feature(s) and or interlock(s) which provide for the following:

4.13 MFW isolation/reset 3.7 3.9

Explanation of Answer Having Loop Isolation Stops closed does NOT defeat P-14.

Reference Title	**Facility Reference Number**	**Section**	**Page**	**Revisio**	**L. O.**
Feedwater Simple	FW-1	FWI Signals		4	
SGWLC	FW-2	Schematic - Flowpaths During Startup		0	
Chp 61,Engineered Safety Features	Chp 61	II.C.6.a	31-32	2	7.c

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 39 ROD BOTTOM Alarm operation

During a reactor startup, when does the ROD AT BOTTOM alarm become active for each control bank?

The alarm will actuate for a dropped rod for...

a. any Control Bank whenever Control Bank A DRPI output is above 9 steps.

b. each Control Bank whenever that Control Bank demand position is above 3 steps.

c. each Control Bank whenever that Control Bank DRPI output is above 9 steps.

d. Control Banks A, B and C whenever their Control Bank demand position is above 9 steps, and for Control Bank D whenever Control Bank D demand position is above 3 steps.

Answer C **Exam Level** R **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 1

014 Rod Position Indication System

2.4 Emergency Procedures / Plan

2.4.31 Knowledge of annunciators alarms and indications, and use of the response instructions. 3.3 3.4

Explanation of Answer Note that the ROD BOTTOM comes directly from the DRPI unit with a setpoint of 9 steps; the alarm actuates when rod position is detected at 3 steps (or less).

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
ROD AT BOTTOM	BAR 1-10-E6	SETPOINT	1	2	
Chp 29, Rod Position Indication System	Chp 29	II.B.6, II.C.1	20-22	2	4, 5

Material Required for Examination

Question Source: New **Question Modification Method:** Significantly Modified

uestion Source Comments: Millstone 3 11/90 NRC Exam

omment Type **Comment**

Question 40 SR NIS discriminator failure

How would the failure of the pulse height discriminator to a low value affect the indication of the affected Source Range channel?

'he output would increase due to...

a. electronic filtering which narrows the pulse height window.

b. failure in removing the higher amplitude neutron generated pulses.

c. increased gamma interaction inside the detector.

d. counting of the gamma generated pulses and decay-alpha generated pulses.

Answer d **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

015 Nuclear Instrumentation System

A2. Ability to (a) predict the impacts of the following on the Nuclear Instrumentation System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:

A2.02 Faulty or erratic operation of detectors or compensating components 3.1 3.5

Explanation of Answer Pulse height discriminator used to set window to detect those pulses with energy level high enough to be from event associated with neutron detection. Gamma and other interactions such as the alpha decay of fission product daughters is of lower height (energy) and discriminator normally electronically removes.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Source Range Detector	NI-4	Schematic - Pulse Amp & Discriminator		4	
Chp 31, Source Range Nuclear Instrumentation	Chp 31	II.A.2.b	16-17	1	3

.laterial Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 41 SR NIS - loss of control power

The following conditions exist on Unit 1:

- RCS at NOT NOP
- Reactor trip breakers - closed
- Source Range readings:
  N31 - 18 cps
  N32 - 22 cps

What indication would the operator observe if Control Power was lost to the N31 Drawer?

The N31 meter would read...

a. downscale, the associated drawer bistable lamps NOT lit, and reactor trip breakers closed.

b. downscale, the associated drawer bistable lamps lit, and reactor trip breakers open.

c. 18 cps, the associated drawer bistable lamps NOT lit, and reactor trip breakers closed.

d. 18 cps, the associated drawer bistable lamps lit, and reactor trip breakers open.

Answer d **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

015 Nuclear Instrumentation System

K2. Knowledge of electrical power supplies to the following:

K2.01 NIS channels, components, and interconnections 3.3 3.7

Explanation of Answer Control power loss affects bistables which trip but NOT drawer instrument indication which is from Instrument Power source.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Source Range Detector	NI-4	Loss Of Control Power		4	
Chp 31, Source Range Nuclear Instrumentation	Chp 31	II.A.2.g.5), III.C.2	18, 70	1	8.b

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 42 Eval for 1/M - Eightfold increase

The following conditions exist on Unit 1:

- A reactor startup is about to be performed
- All shutdown banks are fully withdrawn
- All control banks are fully inserted
- An ECC records the following:
  - Predicted Critical Position (ECP) - 130 steps on CBD
  - Max rod position - 231 steps on CBD
  - Min rod position - 58 steps on CBD

The following parameters were recorded during the rod withdrawal:

ROD HEIGTH	N31 cps	N32 cps
0 on CBA	25	23
178 on CBA	34	31
178 on CBB	80	82
178 on CBC	200	162
80 on CBD	237	184
92 on CBD	260	245

When was the first time the operator was required to determine the Predicted Critical Position?

a. At 50 steps on CBA, with N32 as the designated Source Range detector.

b. At 47 steps on CBC, with N31 as the designated Source Range Detector.

c. At 178 steps on CBC, with N31 as the designated Source Range detector.

d. At 80 steps on CBD, with N32 as the designated Source Range detector.

Answer C **Exam Level** R **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

015 Nuclear Instrumentation System

K5. Knowledge of the operational implications of the following concepts as they apply to the Nuclear Instrumentation System:

K5.06 Subcritical multiplications and NIS indications 3.4 3.7

Explanation of Answer During reactor SU, hold point for ICRR determination is performed for each Control Bank at 50 steps withdrawn. The actual determination of Predicted Critical Position is required at the eight-fold count increase on highest reading SR. Holdpoint occurs on CBC @178. $& steps on CBC is the 0% power RIL value.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Reactor Startup	1BGP 100-2A1	F.8.m	10	12	
BGP 100-2, Plant Startup and BGP 100-2A1, Reactor Startup	BGP 100-2A1	III.A.8.m	58	3	4, 6
BAP 2010-2, Reactivity Management	BAP 2010-2	II.F.5.d.2).e)	48	1	3

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Question 43 NR RTD Failure effects

The following conditions exist on Unit 1:

- Reactor power - 50%
- RCS Tave - 570°F (A); 569°F (B); 569°F (C); 570°F (D)
- RCS Thot - 585°F (A); 584°F (B); 583°F (C); 585°F (D)
- RCS Tcold - 555°F (A) 554°F (B); 555°F (C); 555°F (D)
- Pzr pressure - 2235 psig
- Pzr level - 43 %

If loop B Thot output channel fails LOW, what is the response of Pzr level ?

Pressurizer level will...

a. increases to 60%.

b. remains the same.

c. decreases to 25%.

d. decreases to the letdown isolation setpoint.

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

016 Non-Nuclear Instrumentation System

K3. Knowledge of the effect that a loss or malfunction of the Non-Nuclear Instrumentation System will have on the following:

K3.02 PZR LCS 3.4 3.5

Explanation of answer Thot fails to 510°F. With loop Tcold of 537°F, loop Tave is now 524°F. Auctioneered HIGH Tave is used for Pzr level program.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.	
Pzr Level Control	RY-3	Schematic - Level Program Controller		2		
Abnormal Operating Procedures, Operation with a Failed Instrument Channel		1BOA INST-2 low	I.B.2 - Th fails	15	1	1.a, 4
Chp 12, Reactor Coolant System	Chp 12	II.B.2.f.3), c.3)	23-24, 29	2	6.a	

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Concept Used

Question Source Comments: Zion 2/92 NRC Exam (along with several others). Change includes failure of Thot loop, failure low and conditions instead of dual condition.

Comment Type **Comment**

Question 44 CETC failure effect on Subcooling Monitor/Iconic Display

With Unit 1 at 100% power and with normal operating parameters, how would the failure of the HOTTEST Core Exit Thermocouple affect the reading of subcooling margin on the SPDS Iconics (CETC/SMM display) for each of the two situations below:

Situation 1 - The CETC output fails high slowly
Situation 2 - The CETC output fails low slowly

a. Situation 1: Subcooling margin will decrease to saturation then indicate superheated, and return to normal when CETC output reaches 2300°F.
Situation 2: Subcooling margin will increase, then stabilizes when the CETC output is smaller than TEN other TCs.

b. Situation 1: Subcooling margin will decrease to saturation then indicate superheated, and return to normal when CETC output reaches 1200°F.
Situation 2: Subcooling margin will remain constant.

c. Situation 1: Subcooling margin will increase to saturation then indicate superheated, and return to normal when CETC output reaches 1200°F.
Situation 2: Subcooling margin will decrease, then stabilizes when the CETC output is smaller than TEN other TCs.

d. Situation 1: Subcooling margin will Increase to saturation then indicate superheated, and return to normal when TC output reaches 2300°F.
Situation 2: Subcooling margin will remain constant.

Answer a **Exam Level** R **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98
Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

17 In-Core Temperature Monitor System
K4. Knowledge of In-Core Temperature Monitor System design feature(s) and or interlock(s) which provide for the following:

K4.01 Input to subcooling monitors 3.4 3.7

Explanation of Answer Fail high - Since it is initially the highest, its input will remain active in average until high setpoint reached at 2300°F. Fail low - subcooling margin will slightly increase as temperature falls and input to average remains valid. When it reaches the 11th highest value, the subcooling margin will stabilize and remain constant (assuming other 10 inputs do not change).

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Chapter 34b Inadequate Core Cooling Detection	Chapter 34b	II.A.3.a	22-23	2	6, 7

Material Required for Examination
Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified
Question Source Comments: Braidwood 1997 NRC Exam. Difference in all answer choices - similar premise in theory, but different wording.
Comment Type **Comment**

Question 45 RCFC operations requirements

The following conditions exist on Unit 2:

- RCS Temperature - 342°F
- Pzr pressure - 375 psig
- 2A, 2B, and 2D RCFCs are operating in high speed
- Unit 2 RCFC Dry Bulb temperatures are recorded as foilows:.
  - 2A RCFC - 119°F
  - 2B RCFC - 118°F
  - 2C RCFC - 127°F
  - 2D RCFC - 121°F

Which of the following identifies the equipment status and actions for the above conditions?

a. RCFC 2C must be started because the average of ALL the RCFC temperatures exceeds the limit.

b. RCFC 2C must be started because ONE of the operating RCFCs temperatures is above the limit.

c. NO action is necessary because ALL temperatures are within the required limit.

d. NO action is necessary because the average temperature of ALL operating RCFCs is below the limit.

Answer d **Exam Level** R **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

022 Containment Cooling System

2.1 Conduct Of Operations

2.1.32 Ability to explain and apply all system limits and precautions. 3.4 3.8

Explanation of Answer Limits on CNMT temperature determined by average of temperatures for OPERATING RCFC outlet temps.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
REACTOR CONTAINMENT FAN COOLER START-UP	BOP VP-5	E.1	2	2	
Unit 1 MODE 123 Shiftly And Daily Operating Surveillances	1BOS-0.1-1,2,3	F.8	6	58	
Chp 42 , Containment Ventilation and Purge System	Chp 42	II.C.2.a, IV.A.3, IV.B.b	14, 47	1	6, 10.a

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 46 Sequence for securing CNMT Spray

The following conditions exist on Unit 1:

- A LOCA has occurred
- Transition has been made to BEP ES-1.3 "Transfer To Cold Leg Recirculation"
- Containment Spray actuated due to high containment pressure
- All systems and components operating as expected

What conditions allow for termination of Containment Spray?

a. ONE pump is stopped when containment pressure is less than 15 psig. The other pump is stopped when RWST LO-3 level is reached.

b. ONE pump is stopped when containment pressure is less than 20 psig. The other pump is stopped after it has operated for a period of at least TWO hours

c. BOTH pumps are stopped when containment pressure is less than 15 psig and have operated for a period of at least TWO hours.

d. BOTH pumps are stopped when containment pressure is less than 20 psig and RWST LO-3 level is reached.

Answer C **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 1

026 Containment Spray System

A2. Ability to (a) predict the impacts of the following on the Containment Spray System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:

A2.08 Safe securing of containment spray when it can be done) 3.2 3.7

xplanation of nswer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Containment Spray	CS-1	CS Termination		3	
Loss Of Reactor Or Secondary Coolant	1BEP-1	Step 7.d	9	1A WOG-1 B	
Chp 59, Containment Spray System	Chp 59	III.E.4	54	2	12

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 47 Pump operation interlocks

The following conditions exist on Unit 1:

- LOCA is in progress
- Containment pressure - 17 psig
- Containment Spray actuated due to high containment pressure
- Containment Spray signal has been reset
- The actions of BEP ES-1.3 "Transfer To Cold Leg Recirculation" have been completed
- Offsite power is then lost and the D/G output breakers have just closed onto ESF buses

How are the Containment Spray Pumps re-started?

a. The pumps will auto start 15 seconds following closure of the D/G output breakers.

b. The pumps will auto start 40 seconds following closure of the D/G output breakers.

c. If the operator immediately places the CS & PHASE B ISOL switches for both trains to ACTUATE, the pumps will auto start 15 seconds following closure of the D/G output breakers.

d. If the operator immediately places the PP 1_ TEST switches for both pumps in TEST, the pumps will auto start 40 seconds following closure of the D/G output breakers.

Answer C **Exam Level** R **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 1

026 Containment Spray System

A4. Ability to manually operate and/or monitor in the control roo

`4.01 CSS controls 4.5 4.3

Explanation of Answer If the AUTO actuation input signal is absent and actuation input has been reset, manual actuation is required to get equipment restarted following a LOSP.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Abnormal Operating Procedures - BEP 1, Loss of Reactor or Secondary Coolant	BEP-1, BEP ES-1.1 - 1.4	7	9	2	1.c
Chp 59, Containment Spray System	Chp 59	II.C.1.a	34-36	2	9, 11.e

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 48 Charcoal Filters response to deluge

Annunciator 0-33-C3, FILTER 1VP05FA TEMPERATURE HIGH, alarms in the Control Room while 1VP02CA CNMT Charcoal Filter Fan is operating. The alarm condition is verified locally.

Vhich of the following describes the actions taken and/or system response for the Containment Ventilation system?

a. The deluge valve FP244A will automatically open and the fan will automatically stop.

b. The control room operator will open the deluge valve FP244A and the local operator will then stop the fan.

c. The local operator will open the deluge valve FP244A and the fan will automatically stop.

d. The local operator will open the deluge valve FP244A and the control room operator will then stop the fan.

Answer c **Exam Level** R **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 3 **SRO Group:** 2

027 Containment Iodine Removal System

A4. Ability to manually operate and/or monitor in the control room:

A4.03 CIRS fans 3.3 3.2

Explanation of Answer Operation of FP components associated with charcoal filter is local. But fan trips when deluge system activated.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
FILTER 1VP05FA TEMPERATURE HIGH	BAR 1VP01J-1-A1	D.3	1	1	
Chp 42 , Containment Ventilation and Purge System	Chp 42	I.B.3, III.A.2.g, C.5, IV.C.2	10-11, 22-23,43,	1	8.b

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 49 RWST Purification Loops

The following conditions exist:

- Unit 1 - 20% power with load increase in progress
- Unit 2 - MODE 5 following refueling outage
- Unit 2 Spent Fuel Pool Cooling Loop is in service.
- Spent Fuel Pool Pump 1FC01P is OOS.

Which of the following is allowed under this situation?

Alignment and operation of...

a. both Unit 1 RWST purification and Unit 2 RWST purification with flow through the Unit 2 Spent Fuel Pool Demineralizer and Unit 2 Spent Fuel Pool Filter.

b. Unit 1 Spent Fuel Pool purification and Unit 1 RWST purification with flow through the Unit 1 Spent Fuel Pool Demineralizer and Unit 1 Spent Fuel Pool Filter.

c. Unit 2 RWST purification with flow through the Unit 1 Spent Fuel Pool Filter and return to Unit 2 RWST.

d. Unit 2 RWST purification with flow through the Unit 2 Spent Fuel Pool Demineralizer and Unit 2 Spent Fuel Pool Filter.

Answer d **Exam Level** R **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

033 Spent Fuel Pool Cooling System

K1. Knowledge of the physical connections and/or cause-effect relationships between Spent Fuel Pool Cooling System and the following:

K1.05 RWST 2.7 2.8

Explanation of Answer The lineup allows Unit 2 only to be used for Unit 2 RWST cleanup. Only one unit RWST can be aligned at time due to common input path via Refueling Water Purification Pumps. With the cooling loop inservice only, the Unit's RWST may be aligned through the same Unit's, demin and filter train. Simultaneous use of Demin/filter for the same Unit's SFP and RWST is NOT allowed due to concerns of draining RWST.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
STARTUP OF THE PURIFICATION SYSTEM TO PURIFY OR RECIRCULATE THE REFUELING WATER STORAGE TANK	BOP FC-7	E.5, F.10	2, 4-5	7	
Fuel Pool Cooling	FC-1	Schematic		3	
Chp 51, Spent Fuel Pool Cooling and Cleanup	Chp 51	I.B.1	2-4	2	4

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 50 Steam Dump input malfunction

The following conditions exist on Unit 1:

- Reactor power was 65% when the turbine tripped
- An ATWS occurred
- The reactor tripped 15 seconds later when B reactor trip breaker was locally opened
- Reactor trip breaker A is failed closed
- RCS Tave - 559°F
- Pzr pressure - 2255 psig
- Steamline header pressure - 1100 psig
- No controls other than control rods and boration controls have been operated

What is the status of the Steam Dump valves?

Steam Dumps are...

a. modulating open due to steam header pressure.

b. modulating open due to Tave above no-load Tave.

c. closed because Tave is NOT greater than 3°F above Tref.

d. closed because the dumps are NOT armed.

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 3 **SRO Group:** 3

041 Steam Dump System and Turbine Bypass Control

A3. Ability to monitor automatic operations of the Steam Dump System and Turbine Bypass Control including:

3.02 RCS pressure, RCS temperature, and reactor power 3.3 3.4

Explanation of Answer The "A" reactor trip breaker provides the arming signal for dumps on normal reactor trip. Since "A" RTB is still closed, the steam dumps respond to event like load rejection, with C-7 load rejection (10% load decrease in 2 minutes sensed on PT-506) arming the dumps. Since the "B" RTP was opened, the steam dump controller does operate on the plant trip controller (No load Tave compared to Auct Hi Tave).

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Main Steam Dumps	MS-4	Schematic		4	
Chp 24, Steam Dumps	Chp 24	II.A.2.b, c	10-12	1	2, 3.b

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 51 Turbine Control response to Failed Impulse Channel

The following conditions exist on Unit 1:

- Reactor power 28%
- All systems normal
- Turbine EHC Panel settings:
  - Turbine REFERENCE DEMAND - 580 MW
  - Turbine REFERENCE - 330 MW
- The GO pushbutton is LIT

What would be the DEHC System response to a slow failure to ZERO for the turbine impulse pressure channel that feeds into the DEHC?

Turbine load will...

a. decrease until the difference between REFERENCE and impulse pressure exceeds 30%, the operator would then be alerted to select MANUAL control.

b. decrease until the difference between REFERENCE DEMAND and impulse pressure exceeds 30%, then load will stabilize in MANUAL control.

c. increase until the difference between REFERENCE and impulse pressure exceeds 30%, then load will stabilize in MANUAL control.

d. increase until the difference between REFERENCE DEMAND and impulse pressure exceeds 30%, the operator would then be alerted to select MANUAL control.

Answer C **Exam Level** R **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

er: Plant Systems **RO Group:** 3 **SRO Group:** 3

045 Main Turbine Generator System

K1. Knowledge of the physical connections and/or cause-effect relationships between Main Turbine Generator System and the following:

K1.20 Protection system 3.4 3.6

Explanation of Answer When the difference between actual load and turbine impulse pressure (IMP IN) channel exceeds, circuit AUTO transfer impulse feedback to IMP OUT

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
TV / GV Control	EHC-3	Impulse		1	
Chp 37a, Main Turbine Control and Protection	Chp 37a	II.A.1.a, II.A.1.f.4).d)	21-23, 39	1	2.a, 2.b

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 52 S/G Level program - low power

The following conditions exist on Unit 1:

- Reactor power 35%
- All systems normal

What failure would cause an INITIAL decrease in feedwater flow to all S/Gs?

a. Turbine first stage impulse pressure PT-505 fails low.

b. Main steamline pressure PT-507 fails low.

c. Turbine first stage impulse pressure PT-506 fails low.

d. Main feedwater header pressure PT-508 fails low.

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

059 Main Feedwater System

2.1 Conduct Of Operations

2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation. 3.7 4.4

Explanation of Answer PT-507 fails low causes feed pump speed to decrease which reduces FW pressure. This would initially result in a decrease of flow to all S/Gs.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
FW EH Controls	EHC-6	dP Actual schematic		1	
Chp. 27, Steam Generator Water Level Control System	Chp. 27	I.B.2, II.C.3.I.2)	12, 42-44	1	5, 15.c

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 53 Effect of failure of S/G steam pressure channel

The following conditions exist on Unit 1:

- Reactor power 100%
- All systems normal
- FT-512 selected for steam flow input into SGWLC for S/G 1A

What is the effect of the pressure transmitter associated with FT-512 failing low?

1A S/G level will decrease,...

a. feed pump speed will decrease and S/G level will decrease below the LO-2 setpoint.

b. feed pump speed is unaffected, and S/G level will return to normal.

c. feed pump speed will increase and S/G level will return to normal.

d. feed pump speed is unaffected, and S/G level will decrease below LO-2 setpoint.

Answer a **Exam Level** R **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

059 Main Feedwater System

K1. Knowledge of the physical connections and/or cause-effect relationships between Main Feedwater System and the following:

K1.04 S/GS water level control system 3.4 3.4

Explanation of Answer Steam flow is output to summator for FW control system program Delta-P. Delta-P program will decrease causing feed pump speed and FW header pressure to decrease.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
FW EH Controls	EHC-6	dP Reference		1	
SGWLC	FW-2	FT-512 input to FRV schematic(PT-514)		0	
Chp. 27, Steam Generator Water Level Control System	Chp. 27	I.B.2, II.B.2.c, II.C.3.k.2)	12-13, 30, 44(39)	1	5, 15.c

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 54 AFW Startup

The following conditions exist on Unit 1:

- The reactor tripped from an at-power condition
- An undervoltage condition exists on RCP 1C bus
- Power Range NIS channel N42 failed at 100% on the trip
- ESF bus 141 undervoltage occurred
- 1A D/G automatically started and ACB 1413 is closed
- S/G levels lowest readings were - 19% (A); 25% (B); 22% (C); 20% (D)

What is the status of the Auxiliary Feedwater (AF) Pumps on Unit 1 for these conditions at ONE minute following the trip?

a. Both AF pumps are running.

b. The 1A AF pump is running and the 1B AF pump is NOT running.

c. The 1B AF pump is running and the 1A AF pump is NOT running.

d. NO AF start signal is initiated.

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

061 Auxiliary / Emergency Feedwater System

A3. Ability to monitor automatic operations of the Auxiliary / Emergency Feedwater System including:

A3.01 AFW startup and flows 4.1 4.2

Explanation of Answer SG levels are above AF actuation setpoints and the motor driven AF pump starts on the detected undervoltage.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Auxiliary Feedwater System	LO-PSC-12C	II.A.3.g; II.A.4.j	4 & 6	2	5
Chp 26, Auxiliary Feedwater System (AF)	Chp 26	II.A.3.c	12	3	3, 5
Chp 9, Diesel Generators & Aux. Systems	Chp 9	III.D.2	49-51	1	7.a

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 55 AFW flow requirements for cooldown

In accordance with the BEPs, which of the following describes the MINIMUM AFW pump flow and S/G configuration necessary to remove all of the reactor decay heat load following a reactor trip from 102% power to preclude entry into loss of heat sink RED path entry?

a. The 1A AF pump supplying 480 gpm to at least ONE S/G with S/G blowdown manually isolated.

b. The 1B AF pump supplying 245 gpm to each of TWO S/G with S/G blowdown in service.

c. The 1A AF pump supplying 170 gpm flow to each of THREE S/Gs with S/G blowdown manually isolated.

d. The 1B AF pump supplying 130 gpm flow to each of FOUR S/Gs with S/G blowdown in service.

Answer C **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

061 Auxiliary / Emergency Feedwater System

K5. Knowledge of the operational implications of the following concepts as they apply to the Auxiliary / Emergency Feedwater System:

K5.02 Decay heat sources and magnitude 3.2 3.6

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Bases - AF System	Byron 1 & 2 ITS	B.3.7.5 - Background	B 3.7-28	A	
Auxiliary Feedwater System	Chp 26	I.C.1, 5	6	3	1, 11

Material Required for Examination

Question Source: New **Question Modification Method:** Significantly Modified

Question Source Comments: Comanche Peak 11/93 NRC Exam

Comment Type **Comment**

Question 56 DC bus battery charger

The following conditions exist on Unit 1:

- Reactor power - 100%

Investigation has located a ground on the 125 VDC Normal supply to the 1A D/G. What action is required to transfer DC control power to the reserve source?

The Reserve power breaker from...

a. DC 111 will be closed after opening the Normal power breaker and the Reserve power breaker at the D/G control panel.

b. DC 111 will be closed after swapping the no-blow link at the Normal and Reserve power fuse blocks at the D/G control panel.

c. DC 112 will be closed after opening the Normal power breaker and the Reserve power breaker at the D/G control panel.

d. DC 112 will be closed after swapping the no-blow link at the Normal and Reserve power fuse blocks at the D/G control panel.

Answer b **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 1

063 D.C. Electrical Distribution

2.1 Conduct Of Operations

2.1.30 Ability to locate and operate components, including local controls. 3.9 3.4

Explanation of nswer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
125VDC System	DC-1			0	
DC Control Power Transfer From Normal To Reserve Source	BOP DC-6A1	6, 7	1	51	
Chp 8a, 125 VDC	Chp 8a	II.A.4.b, c	12	1	4.d

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 57 Sequencing of ESF pumps - SI & SI w LOP

Unit 1 was being synchronized to the grid when the following occurred:

- Trip of 345 KV breakers resulted in deenergizing the SATs

- A steamline break occurred that resulted in containment pressure reaching 20 psig 20 seconds after the D/Gs output breakers have closed

When would the 1A SX pump re-start?

a. Following start of the 1A CS Pump.

b. Between the start of the 1A CV pump and the 1A RH pump.

c. Between the start of 1A CC pump and the 1A AF pump.

d. Coincident with the starting of the 1A and 1C RCFCs.

Answer C **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

064 Emergency Diesel Generators

A3. Ability to monitor automatic operations of the Emergency Diesel Generators including:

A3.07 Load sequencing 3.6 3.7

Explanation of Answer The SX pump would be started in this case by the SI signal which is overrides the UV condition. The SX pump starts in following sequence: CV (0 sec); SI ((5 sec); RH (10sec); CS (15-18 secs, if actuation signal present);.CC pumps (20 sec); SX pumps (25 sec); AF 1A pump (35 sec); CS pump (40 sec, if actuation signal now present but not present at 18 sec)

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
D/G Relaying	DG-2	Sequencing Order		1	
Chp 9, Diesel Generators & Aux. Systems	Chp 9	III.D.1	47-49	1	7.c
Chp 20, Essential Service Water System	Chp 20	II.C.1.a	60	2	8.a

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 58 RCDT operation - effect of CNMT Isolation

The following conditions exist on Unit 1:

- Unit is in MODE 3
- A cooldown had just been initiated
- Steam Dump Bypass Interlock control switches have just been taken to BYPASS
- No other operator actions have been performed
- The Steam Dump valves fail open and the following parameters are observed:
- RCS temperature - 537°F (A); 539°F (B); 538°F (C); 538°F (D)
- Pzr pressure - 1820 psig
- Pzr level - 10%
- S/G pressure - 850 psig (A); 740 psig (B); 800 psig (C); 750 psig (D)
- S/G flow - 1.0 Mlb/hr (A); 1.5 Mlb/hr (B); 1.1 Mlb/hr (C); 1.6 Mlb/hr (D)
- The level in the RCDT has risen to the alarm setpoint (80%) for REACTOR COOLANT DRAIN TANK UNIT 1 LEVEL HI-LO

Assuming all systems are functioning correctly, what is the status of the RCDT system?

a. BOTH RCDT pumps are running and flow is directed to the Holdup Tanks.

b. BOTH RCDT pumps are running and flow is recirculated back to the RCDT.

c. ONE RCDT pump is running and flow is directed to the Holdup Tanks.

d. NEITHER RCDT pump is running and NO flow exists for the system.

Answer d **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

068 Liquid Radwaste System

A4. Ability to manually operate and/or monitor in the control room:

A4.04 Automatic isolation 3.8 3.7

Explanation of Answer Conditions for low Pzr pressure actuates SI. The coincident CNMT Phase A Isolation signal isolates RCDT valves out. Closure of valve RE9170 causes pumps to stop.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
PRT & RCDT	RY-4	Schematic		2	
Chp 46b Liquid Radwaste System	Chp 46b	II.A.3.a.4), 5), 7), 8)	26	2	6
Chp 61,Engineered Safety Features	Chp 61	II.C.1.a, 2.f	25, 27	2	7.d

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 59 CNMT Sump sources of input during normal operations

During at-power operations with systems in their normal alignment, what is a source of water to the Containment Floor Sump?

a. SI Accumulator valve leakoffs.

b. Leakoff from the #3 RCP seals.

c. Leakoff from the reactor vessel flange.

d. Valve packing leakage from the CVCS letdown isolation valves.

Answer b **Exam Level** R **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

068 Liquid Radwaste System

K1. Knowledge of the physical connections and/or cause-effect relationships between Liquid Radwaste System and the following:

K1.07 Sources of liquid wastes for LRS 2.7 2.9

Explanation of Answer Rx cavity sump output to CNMT Floor sump, #2 seals directed to RCDT, RV flange to RCDT , valve leakoffs directed to PRT

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Chp 46b Liquid Radwaste System	Chp 46b	II.A.3.b	28	2	

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 60 Waste Gas Decay Tank Operations

When aligned for normal operation (BOP GW-1), what is the response to high pressure sensed at the in-service Gas Decay Tank?

An alarm is generated that...

a. alerts the operator to manually place a standby Gas Decay Tank in service.

b. indicates auto swap of in-service Gas Decay Tank to selected standby Gas Decay Tank, and alerts the operator to align another standby Gas Decay Tank.

c. indicates auto swap of in-service Gas Decay Tank to selected standby Gas Decay Tank and auto swap of standby Gas Decay Tank to new standby Gas Decay Tank.

d. shuts down the Waste Gas Compressors and isolates the in-service Gas Decay Tank.

Answer b **Exam Level** R **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

071 Waste Gas Disposal System

A4. Ability to manually operate and/or monitor in the control room:

A4.05 Gas decay tanks, including valves, indicators, and sample line 2.6 2.6

Explanation of Answer Indicates auto swap to standby WGD Tank at 95 psig.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
GASEOUS WASTE SYSTEM STARTUP AND OPERATION	BOP GW-6	E.3	2	5	
GAS DECAY TANK SEL SW REPOS REQ'D	0GW02J-A1	A.2		51	
Chp 46a, Radwaste Systems Gas	Chp 46a	II.C.4.d	27	2	5.c, g

Material Required for Examination

uestion Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 61 Check Source operation

Area Radiation Monitor for Fuel Bldg Fuel Handling Incident (0RE-AR055) is being manually Check Source tested. What is the system response when the monitor's CHECK SOURCE (C/S) pushbutton is depressed at the RM-23 panel?

a. The alarm and automatic action output will be blocked, and the RM-23 amber INTLK LED will be lit.

b. The alarm and automatic action output will be blocked, and the RM-23 green AVAIL LED will be lit.

c. The alarm will actuate when the alert setpoint value is reached, and the RM-23 amber INTLK ED will be lit.

d. The alarm will actuate when the high setpoint value is reached, and the RM-23 red HIGH LED will be lit.

Answer b **Exam Level** R **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

072 Area Radiation Monitoring System

A4. Ability to manually operate and/or monitor in the control room:

A4.03 Check source for operability demonstration 3.1 3.1

Explanation of Answer Depressing the C/S blocks the alarm and auto function of the monitor but the AVAIL light remains lit.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
CONTROL FUNCTIONS CHANNEL CHECK SOURCE ENERGIZED	BOP AR/PR-11A26	B.1	1	1	
RADIATION MONITORS SYSTEM	Chapter 49 (text)	RM-23 Control Room Monitors	49-28	2	
Chp 49, Radiation Monitors	Chp 49	III.A.1.a.4), III.C.2.r	18, 40	3	3.d, 8

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 62 Loss of FHB Overhead Crane rad monitor

The following conditions exist on Unit 2:

- Refueling operations are in progress

While using the Fuel Handling Building Crane to move new fuel into the Spent Fuel Pool, the radiation monitor 0RE-AR039, Fuel Handling Building Crane Monitor, goes into high alarm. What action is affected?

a. Traverse of the Fuel Handing Building Crane bridge and trolley.

b. Both lowering and raising the Fuel Handing Building Crane hoist.

c. Traverse of the Fuel Handing Building Crane trolley and raising the hoist.

d. Raising the Fuel Handing Building Crane hoist.

Answer d **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

072 Area Radiation Monitoring System

K3. Knowledge of the effect that a loss or malfunction of the Area Radiation Monitoring System will have on the following:

K3.02 Fuel handling operations 3.1 3.5

Explanation of Answer Rad monitor prevents raising hoist.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Chp 49, Radiation Monitors	Chp 49	III.C.2.a	34	3	4.a.3)

aterial Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 63 Evaluation of eqpt affected for slow loss

The following conditions exist on Unit 1:

- A unit startup is in progress with reactor power raised above 18%.
- Turbine is at 1800 rpm ready to be synchronized to grid.
- Motor driven feedwater pump is supplying the S/Gs with Feed Reg Bypass valves in AUTO.
- Steam Dump demand in AUTO at 12%.
- Instrument air header pressure begins to slowly drop due to a leak

If the leak CANNOT be isolated and instrument air pressure continues to drop, which of the following would occur?
(Assume NO operator action taken.)

a. AF recirculation flow to the CST would be lost due to AF recirc, 1AF022A, failing closed.

b. Pressurizer level would increase due to charging header flow control valve, 1CV121, failing open.

c. Pressurizer pressure would decrease due to Aux spray isolation, 1CV8145, failing open.

d. Feedwater heater 17A extraction steam would isolate due to emergency drain, 1HD038A, failing closed.

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 3 **SRO Group:** 3

078 Instrument Air System

K3. Knowledge of the effect that a loss or malfunction of the Instrument Air System will have on the following:

K3.02 Systems having pneumatic valves and controls 3.4 3.6

xplanation of nswer Charging flow goes to maximum due to 1CV121 failing open, and letdown isol 1CV459 & 1CV460 fail closed. 'a' is incorrect because both 1A & 1B AF pump recirc valves fail open . 'c' main turbine not directly affected. 'd' not occur because steam dumps fail closed.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Loss of Instrument Air	BOA SEC-4	Table A	5	52	
Chp 53, Service Air and Instrument Air	Chp 53	III.C.2.c	62	1	9

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 64 Effect of loss of DC - CO2 actuation

With the fire protection systems in their normal alignment, what is the affect of a loss of DC power?

Loss of DC control power to the...

a. halon control cabinet will cause halon release in the Upper Cable Spreading Room.

b. battery control panel will cause automatic start of the diesel driven fire pump.

c. fire detection system will cause start of the motor driven fire pump.

d. carbon dioxide system will cause the master EMPC valve to open pressurizing the CO2 header.

Answer d **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

086 Fire Protection System

K4. Knowledge of Fire Protection System design feature(s) and or interlock(s) which provide for the following:

K4.06 CO2 3.0 3.3

Explanation of Answer EMPCs uses DC control power. On loss of power, the master EMPC valves fail open which in turn cause the master discharge/selector valve to open, charging the affected header.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Chp 57, Fire Protection System	Chp 57	II.C.3.a.4)	42	3	8
CO2 Fire Suppression System Reset After Manual Initiation Following Loss Of Power	BOP FP-26	F	3-4	2	

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 65 Evaluate conditions - unwarranted rod withdrawal

The following conditions exist on Unit 1:

- Reactor power is 30%.
- Rod control is in Automatic
- Tref - 564°F
- Tave values - 564°F (A); 565°F (B); 565°F (C); 564°F (D)
- Power Range NI - 31% (N41); 29% (N42), 30% (N43); 30% (N44)
- Control bank D is at 156 steps.

Which condition would result in continuous rod withdrawal?

a. Turbine first stage pressure PT-505 fails to 100%.

b. Power Range channel N41 fails to 20%.

c. Loop A Tcold fails 553°F.

d. Tref signal fails 557°F.

Answer a **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 1

001 Continuous Rod Withdrawal

AA2. Ability to determine and interpret the following as they apply to Continuous Rod Withdrawal:

AA2.05 Uncontrolled rod withdrawal, from available indications 4.4 4.6

Explanation of Answer Input to rod control Tref, auctioneered HIGH Tave & Auctioneered high PRNIs: PT-505 provides input signal for development of Tref. If it fails high Tref goes to maximum value (581°F) and results in rods being withdrawn to match Tave to Tref. PR failure high compares the rate of change of reactor power to the rate of change of turbine power. Initially high rate of change during failure but rapidly the rate of change falls to zero and so rods may initially begin to insert but quickly stop motion with no more rate of change. Auctioneered high Tave is used and Tcold failing low will remove this input (if previously auctioneered high). Tref failing low will cause rods to move inward to match Tave to Tref.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Rod Control Unit	RD-2	If rods stepping...		20	
Chapter 28: Rod Control System	Chapter 28	II.A.2.b, 3.b, II.B.1.a	22, 26-27, 68	1	4, 7
Uncontrolled Rod Motion	BOA ROD-1	II.C.3.d	4-5	02	2, 3

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 66 P/A vs. Group Step Counters

A Control Bank D rod was dropped from 156 steps. The P-A converter was left at 156 steps when it was to be reset to ZERO steps as directed by procedure BOA ROD-3 "Dropped Rod Recovery".

Select the affect of performing the procedure in this manner?

a. While performing the procedure, the C-11 Rod Stop will be received prior to realigning the rod.

b. While performing the procedure, the Rod Insertion Limit Alarm will be received at a lower rod position than required.

c. After the procedure is complete, Bank C control rods will begin insertion at a lower value of Control Bank D.

d. After the procedure is complete, Bank C control rods will begin insertion at a higher value of Control Bank D.

Answer a **Exam Level** B **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 1

003 Dropped Control Rod

AK3. Knowledge of the reasons for the following responses as they apply to Dropped Control Rod:

AK3.10 RIL and PDIL 3.2 4.2

Explanation of Answer The bank overlap units are bypassed when rods are moved with individual bank selector positions. The P to A converter provides step information to rod position indication including the C-11 circuit. As the individual rod was withdrawn to approximately 67 steps the C11 circuit would sense that bank D was at 223 steps and block outward motion.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
RD Data Logging / Rod Stops	RD-5 / RD-1	P/A & C-11 Rod Stop / Auto Rod Withdrawal ZY442CX		0 / 0	
Chapter 28: Rod Control System	Chapter 28	II.A.7.b, II.B.2	46, 70	1	1.c, 10
Dropped or Misaligned Rod	BOA ROD-3	C.14	13	2	4

Material Required for Examination

Question Source: New **Question Modification Method:** Editorially Modified

Question Source Comments: D.C. Cook 6/13/1995

Comment Type **Comment**

Question: 67 Stabilized RCS temperature with failure of Steam Dumps

On Unit 1, a loss of all circulating water pumps has resulted in a reactor trip. All control systems respond as expected. Significant decay heat causes RCS temperature to increase following the trip.

At what RCS temperature should temperature stabilize?

Temperature should stabilize at...

a. 550°F.

b. 557°F.

c. 561°F.

d. 565°F.

Answer C **Exam Level** B **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

007 Reactor Trip

EA1. Ability to operate and / or monitor the following as they apply to Reactor Trip:

EA1.03 RCS pressure and temperature 4.2 4.1

Explanation of Answer The condenser would NOT be available for steam dumps (either on trip controller or load rejection controller). The S/G pressure would stabilize based on the secondary PORV opening setpoint normally set at 1115 psig. The Main Steam safety valve setting is 1175 psig. At 550°F the steam dumps would be blocked (P12).

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Main Steam Dumps	MS-4	C-9, If Main Condenser NOT available		4	
Chp 24, Steam Dumps	Chp 24	II.C.1.b	24	1	4.b
Chp. 23, Main Steam System	Chp. 23	II.A.3.b	18	2	3

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type Comment

Question 68 Reactor Trip requirements

If Unit 2 is operating at full load, which group of conditions will result in an automatic reactor trip either directly or indirectly?

a. RCP bus frequency(Hz):56.9 (Bus 156) 57.1(Bus 157) 56.9 (Bus 158) 57.2 (Bus 159)

b. Power range (%): 107 (N41) 108 (N42) 108 (N43) 109 (N44)

c. PZR pressure (psig): 2375 (PT-455) 2380 (PT-456) 2385 (PT-457) 2380 (PT-458)

d. S/G C NR level (%): 35 (LT-537) 38 (LT-538) 38 (LT-539) 37 (LT-558)

Answer a **Exam Level** R **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

007 Reactor Trip

EK2. Knowledge of the interrelations between Reactor Trip and the following:

EK2.03 Reactor trip status panel 3.5 3.6

Explanation of Answer Trip condition RCP UF - 2/4 RCP buses < 57.0 Hz. Other trip setpoints: Rx power - 2/4 >109%; Pzr pressure 2/4 > 2385 psig

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
ESF Setpoints	EF-1	Reactor Trip		4	
Emergency Procedures - BEP-0, BEP ES-0.0 - 0.4	BEP-0	III.B	1	3	6
Chapter 60b/Reactor Protection System	Chapter 60b	II.B.3.c.3)	Number(s)	2 [illegible]	n 4

Material Required for Examination

Question Source: New **Question Modification Method:** Significantly Modified

Question Source Comments: Comanche Peak 11/94

Comment Type **Comment**

Question 69 Tail-Pipe conditions

With the RCS at normal operating pressure and temperature, what is the condition of the steam entering the PRT at normal conditions, if a PORV opens? (Assume an ideal thermodynamic process).

a. Superheated steam at 651°F.

b. Superheated steam at 250°F.

c. Saturated steam-water mixture at 222°F.

d. Saturated steam water mixture at 163°F.

Answer c **Exam Level** R **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

008 Pressurizer Vapor Space Accident

AK1. Knowledge of the operational implications of the following concepts as they apply to Pressurizer Vapor Space Accident:

AK1.01 Thermodynamics and flow characteristics of open or leaking valves 3.2 3.7

Explanation of Answer Nominal PRT pressure 3 psig; Hg = 1154 BTU/lb. Saturation temperature 221.9°F. At NOP Pzr pressure 2235 psig with Hg = 1117.7 BTU/lb. Therefore PRT conditions are within saturation parameters.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Steam Tables					
Chp 14, Pressurizer (RY)	Chp 14	II.A.7.g	30-32	3	13, 28

Material Required for Examination Steam Tables

Question Source: New **Question Modification Method:** Significantly Modified

Question Source Comments: South Texas 9/95

Comment Type **Comment**

Question 70 Calculation of subcooled margin on Iconics

The following conditions exist on Unit 1:

- Subcooling Margin output from the SPDS Iconics has failed
- 1C RCP and 1D RCP are running

The Unit Supervisor has asked you to determine the subcooling margin using the same valid inputs as used by SPDS.

What are the parameters used to calculate subcooling margin?

a. RCS wide range pressure from loop C hot leg and core exit thermocouple temperatures.

b. Pressurizer pressure and core exit thermocouple temperatures.

c. RCS wide range pressure from loop A and loop C hot leg, and RCS loop A and loop C hot leg temperatures.

d. Pressurizer pressure and RCS loop C and loop D hot leg temperatures.

Answer a **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

009 Small Break LOCA

EA1. Ability to operate and / or monitor the following as they apply to Small Break LOCA:

EA1.10 Safety parameter display system 3.8 3.9

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
SPDS Display	CX-1	Subcooling		1	
Chapter 34b Inadequate Core Cooling Detection	Chapter 34b	II.A.3	22-23	2	6

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 71 RCP trip criteria evaluation

The following conditions exist during performance of BEP-0.

- Train A ECCS pumps failed to start.
- RCS pressure is 1350 psig.
- Containment pressure of 7 psig.
- Bus 142 has an overcurrent trip on the normal feeder breaker.
- SI actuated due to High Containment Pressure.
- The highest critical safety function is Yellow on Heat Sink.
- All other equipment and components operated as expected.

Based on above plant conditions, the RCPs should...

a. remain running because NO SI pumps or Charging Pumps are running.

b. be stopped because RCS pressure is below the RCP trip criteria.

c. remain running until Pressurizer level decreases below 34%.

d. be stopped because CC flowpath to the RCP motor oil coolers is isolated.

Answer a **Exam Level** B **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 1

011 Large Break LOCA

EA1. Ability to operate and / or monitor the following as they apply to Large Break LOCA:

EA1.03 Securing of RCPs 4.0 4.0

Explanation of Answer The trip criteria is < 1425 psig, with NO cooldown in progress, and HHSI flow > 50 gpm or SI flow > 100 gpm.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Operator Action Summary for 1BEP-0	1BEP-F:0	TRIP RCPs		1C WOG-1 B	
Emergency Procedures - BEP-0 REACTOR TRIP OR SAFETY INJECTION. BEP ES-0.0 - 0.4	BEP-0	Rx Coolant Pump Trip Criteria	3	3	5, 7

Material Required for Examination

Question Source: New **Question Modification Method:** Significantly Modified

Question Source Comments: Watts Bar 3/3/1995

Comment Type **Comment**

Question 72 Eval loss of cooling flow

On a loss of seal injection to the RCPs, what criteria is used to determine if the RCPs should be tripped per BOA RCP-2 "Loss Of Seal Cooling"?

a. High temperatures on the RCP lower bearing outlet temperatures.

b. Time elapsed since loss of seal injection.

c. RCP Thermal Barrier Component Cooling Water low flow alarms.

d. High vibration condition on the RCP.

Answer a **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

015 Reactor Coolant Pump Malfunctions

AA2. Ability to determine and interpret the following as they apply to Reactor Coolant Pump Malfunctions:

AA2.10 When to secure RCPs on loss of cooling or seal injection 3.7 3.7

Explanation of Answer Seal & bearing temperatures are monitored for trip setpoint.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Loss of Seal Cooling	BOA RCP-2	step 1 RNO	2	54	
Abnormal Operating Procedures, Loss of Seal Cooling	1BOA RCP-2	II. 1.a	4-5	1	6

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 73 Eval of RCP seal failure

Unit 1 is operating at 100% power when the following alarms are received/reported:

- RCP SEAL LEAKOFF FLOW LOW (1-7-C3)

The NSO investigates and reports the following additional information:

- RCP 1A seal injection flow is 10.7 gpm
- #1 Seal Leakoff Flow on 1A RCP is 0.4 gpm
- RCP 1A Seal Water Outlet Temperature is 140°F and STABLE
- RCP 1A Bearing Outlet Temperature is 145°F and STABLE
- Unit 1 RCDT level indicates 75%

Based on the above information, which of the following events has occurred?

a. RCP 1A #1 Seal has failed closed

b. RCP 1A #1 Seal has failed open.

c. RCP 1A #2 Seal has failed closed.

d. RCP 1A #2 Seal has failed open.

Answer d **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

015 Reactor Coolant Pump Malfunctions

AK2. Knowledge of the interrelations between Reactor Coolant Pump Malfunctions and the following:

AK2.07 RCP seals 2.9 2.9

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.	
Reactor Coolant Pump Seal Failure	1BOA RCP-1	steps 3-5 & 8	4-7	55B		
BOA RCP-1, REACTOR COOLANT PUMP Seal Failure		BOA RCP-1	C. 6	8	2	2, 7

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: Braidwood bank

Comment Type **Comment**

Question 74 VCT level transmitter malfunction

Given the following:

- The plant is at 90% power with ALL controls in AUTO.
- VCT level transmitter, LT-112, fails HIGH causing a letdown diversion.
- At the time of failure VCT level transmitter, LT-185, reads 50%.

What will occur if NO operator action is taken?

VCT level decreases...

a. until Auto makeup starts and maintains VCT level.

b. with NO auto makeup capability and charging suction shifts to RWST.

c. faster than auto makeup input and charging suction shifts to RWST.

d. until charging pumps lose suction and start to cavitate.

Answer d **Exam Level** B **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

022 Loss of Reactor Coolant Makeup

AA1. Ability to operate and / or monitor the following as they apply to Loss of Reactor Coolant Makeup:

AA1.08 VCT level 3.4 3.3

Explanation of Answer LT 112 provides for AUTO makeup to the VCT. If NO operator action taken, then level will continue to fall until NPSH is lost to the CENT CHG pump(s). Transfer will NOT occur to RWST since both channels are required for swap. An alarm will be generated from LT-185 at 20% level.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
CVCS Notes	CV-2	LT-112 Table		3	
Chp 15a - Chemical and Volume Control System	Chp 15a	II.A.1.n.3).g), h), I)	35-37	2	11

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 75 Time/amount E-boration for condition

Given the following after a reactor trip:

- THREE rods remain withdrawn.
- Due to equipment malfunctions boration is only available from the RWST.
- Charging flow rate 132 gpm.
- RCS boron concentration was 1050 prior to the trip.
- 120 gpm letdown in service.

Of the listed times, which would be minimum acceptable total time that boration from the RWST would have to occur?

a. 1 hour.

b. 2 hours.

c. 3 hours.

d. 4 hours.

Answer C **Exam Level** B **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

024 Emergency Boration

AA2. Ability to determine and interpret the following as they apply to Emergency Boration:

AA2.05 Amount of boron to add to achieve required SDM 3.3 3.9

Explanation of Answer 1BEP ES-0.1 requires 5500 gallons boration from RWST for each rod not fully inserted, therefore requiring 16,500 gallons. The net turnover rate in the RCS is 120 gpm, then required time is 16,500/120 = 137.5 minutes (rounded up to 140 min.). Other answers based on counting 2 rods and/or borating from CV-8104 @ 57 gpm with total of 1320 x # rods out gallons.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Emergency Boration	BOA PRI-2	1 RWST 4)	3	55B	
Abnormal Operating Procedures Emergency Boration	BOA PRI-2	3.a	8-9	1	4, 6
Emergency Procedures - BEP-0 REACTOR TRIP OR SAFETY INJECTION. BEP ES-0.0 - 0.4	BEP ES-0.1	5	6	3	7

Material Required for Examination 1BEP ES-0.1, page 6 (step 5)

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 76 Calc of time to saturation/core boiling

The following conditions exist on Unit 1:

- A forced outage is in progress
- The plant was shutdown 8½ days ago to repair a steam generator tube leak.
- Draining of the RCS was initiated to allow access to S/Gs.
- Reactor vessel level is at 397' 1" with Thot at 212°F.
- A loss of RHR pumps due to cavitation has occurred

Which of the following is the smallest amount of flow that meets the minimum makeup flow required to maintain current RCS level?

a. 80 gpm.

b. 72 gpm.

c. 65 gpm.

d. 59 gpm.

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

025 Loss of Residual Heat Removal System

AK1. Knowledge of the operational implications of the following concepts as they apply to Loss of Residual Heat Removal System:

AK1.01 Loss of RHRS during all modes of operation 3.9 4.3

Explanation of Answer 81/2 days is 204 after shutdown. The curve shows minimum flow at approximately 70 gpm.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
OSS OF RH COOLING	1BOA PRI-10	Fig. 1BOA PRI 9 10-1		56	
BOA PRI-10 Loss of RH Cooling	BOA PRI-10	Fig 1BOA PRI 9 10-1		2	5

Material Required for Examination Figure 1BOA PRI 10-1

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 77 Alternate RCS cooling

The following conditions exist on Unit 2:

- MODE 5 operation during normal cooldown
- RCS temperature - 195° F
- RCS pressure - 325 psig
- Train A RH in service, train B RHR tagged out for repairs

What is the preferred method of core cooling if a loss of RH cooling occurs?

Alternate RCS cooling using...

a. the SI accumulators.

b. the S/Gs.

c. normal charging and RHR letdown.

d. SI Pump hot leg injection.

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

025 Loss of Residual Heat Removal System

AK3. Knowledge of the reasons for the following responses as they apply to Loss of Residual Heat Removal System:

AK3.01 Shift to alternate flowpath 3.1 3.4

Explanation of Answer Steaming Intact/non-isolated SGs is the preferred alternate decay heat removal method if the RCS is intact.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
OSS OF RHR SHUTDOWN COOLING	1BOA PRI-10	Table A	8	56	
BOA PRI-10 Loss of RH Cooling	BOA PRI-10	Seven Attachments (Attachment B	8	02	4

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 78 Evaluation of CCW leak

The following conditions exist on Unit 1:

- The reactor is shutdown.
- RHR is in shutdown cooling.
- RCS temperature is 300ºF.
- RCS pressure is 160 psig.
- CCW surge tank level is decreasing

What leak locations will produce these indications?

a. RHR Heat Exchanger.

b. Thermal Barrier Heat Exchanger.

c. Letdown Heat Exchanger.

d. Seal Water Heat Exchanger.

Answer d **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

026 Loss of Component Cooling Water

AA1. Ability to operate and / or monitor the following as they apply to Loss of Component Cooling Water:

AA1.05 The CCWS surge tank, including level control and level alarms, and radiation alarm 3.1 3.1

Explanation of Answer The seal water HX would be the only location where the CC pressure would be lower than the process fluid pressure. RHR HX approx. 165 psig; L/D Hx pressure should be approximately 160 psig; & Thermal barrier pressure should be about 160 psig.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Component Cooling Malfunctions	BOA PRI-6	Attachment B, steps 1, 2	1-2	56	
Abnormal Operating Procedures, Component Cooling Malfunctions	BOA PRI-6	Attach B, step 2	13	1	3.c, 5

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: Zion 7/13/92

Comment Type **Comment**

Question 79 Pressure controller step change

The following conditions exist on Unit 2:

- Reactor power is 100%
- Pressurizer pressure control is in automatic.

What is the immediate response of the pressure control system if the Master Pressure Controller setpoint is inadvertently changed to 2330 psig (step change)?

a. PORV RY455A opens and spray valves open.

b. PORV RY455A opens, spray valves open, and all heaters energize.

c. Spray valves open and proportional heaters go to minimum.

d. Spray valves close and proportional heaters go to maximum.

Answer d **Exam Level** B **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 2

027 Pressurizer Pressure Control Malfunction

AA1. Ability to operate and / or monitor the following as they apply to Pressurizer Pressure Control Malfunction:

AA1.01 PZR heaters, sprays, and PORVs 4.0 3.9

Explanation of Answer Setting the pot setting higher reduces the output from the controller and raises the demanded pressure setpoint. This reduction results in spray valve closure & heaters turning fully on.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Pzr Pressure Control	RY-2	PK-455A in "AUTO"		3	
Chp 14, Pressurizer (RY)	Chp 14	II.C.1.a.2), II.C.1.c.	53	3	21

Material Required for Examination

Question Source: New **Question Modification Method:** Significantly Modified

Question Source Comments: Calvert Cliffs 11/97

Comment Type **Comment**

Question 80 Non-Controlling channel failure

The following conditions exist on Unit 1:

- Reactor power is 100%
- All systems are in automatic
- Pressurizer pressure channels PT-456 and PT-458 reads normal
- Channel I Pressurizer Pressure Channel (PT-455) was declared inoperable and taken out of service with the appropriate bistables placed in the tripped condition .
- Controlling pressurizer pressure channel (PT-457) fails high

Assuming NO operator action, what is the plant response to the channel failure?

a. Both PORVs and both spray valves open resulting in a reactor trip from low pressurizer pressure followed by SI actuation.

b. The reactor will trip on high pressure, and safety injection will actuate on low pressure due to spray valve operation.

c. Pressurizer proportional heaters will de-energize and spray valves will open resulting in an OTdT runback prior to reactor tripping, and SI will actuate due to low pressurizer pressure.

d. Both PORVs and both spray valves remain closed while pressurizer heaters de-energize.

Answer b **Exam Level** B **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 2

027 Pressurizer Pressure Control Malfunction

AA2. Ability to determine and interpret the following as they apply to Pressurizer Pressure Control Malfunction:

AA2.15 Actions to be taken if PZR pressure instrument fails high 3.7 4.0

xplanation of Answer TWO PZR pressure channels will have HIGH PZR PRESSURE bistables actuated resulting in the reactor trip. The sprays will have modulated fully open resulting in actual pressure decreasing (PORV 1RY455A would have also opened on the failure of PT-457, but would close when the PZR pressure fell to 2185 psig PT-458 will actuate the low pressure interlock closing the PORV) until SI occurs at 1829 psig.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Pzr Pressure Control	RY-2	Pzr Press Channels schematic		3	
Chp 14, Pressurizer (RY)	Chp 14	III.C.2.d	81	3	21

Material Required for Examination

Question Source: New **Question Modification Method:** Significantly Modified

Question Source Comments: BV 8/91

Comment Type **Comment**

Question 81 Failed level channel low.

The plant is operating at 100% power with all control systems in AUTO. The following parameters are noted:

- Letdown Hx outlet flow (FI-132) - 75 gpm
- Charging Header flow (FI-121) - 87 gpm
- Total seal injection flow (FI-142 -FI -45) - 33 gpm

What is the effect on total seal injection flow initially if controlling Pzr level channel LT-459 fails LOW?

Total seal injection flow will...

a. decrease to 0 gpm.

b. decrease to approximately 20 gpm.

c. remain approximately 33 gpm.

d. increase to greater than 40 gpm.

Answer d **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 3 **SRO Group:** 3

028 Pressurizer Level Control Malfunction

AK3. Knowledge of the reasons for the following responses as they apply to Pressurizer Level Control Malfunction:

AK3.05 Actions contained in EOP for PZR level malfunction 3.7 4.1

Explanation of Answer The failure of the level instrument low increases charging flow and charging discharge header pressure. Since seal injection flow is normally increased by throttling close on CV182 to increase backpressure, the result is the same and seal injection flow will increase.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.	
CVCS notes	CV-2	CVCS ratings		2		
Operation with a Failed Instrument Channel	1BOA INST-2	Attachment C, step 1	1	56A	3	
Abnormal Operating Procedures, Operation with a Failed Instrument Channel		1BOA INST-2 step 1	Attachment C,	35	1	3

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: Braidwood 1996 NRC exam. Modified premise from failed controller to failed level channel. Changed location of correct answer based on different response (increasing flow instead of decreasing flow).

Comment Type **Comment**

Question 82 AMS conditions

The following conditions exist on Unit 1:

- At t= 0 sec, Turbine load was decreased below 352 MW (30% power)
- At t=240 sec, The running main feedwater pump tripped.
  The reactor did NOT trip due equipment malfunction.
- At t=250 sec, All feedflow indications decrease to 0% flow
- At t=320 sec, All steam generator levels decrease below 15%.

Based on this information, AMS would...

a. initiate at t=320 sec.

b. initiate at t=345 sec.

c. initiate at t=360 sec.

d. NOT initiate because C-20 is cleared.

Answer b **Exam Level** B **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 1

029 Anticipated Transient Without Scram

2.4 Emergency Procedures / Plan

2.4.48 Ability to interpret control room indications to verify the status and operation of system, and understand how operator actions and directives affect plant and system conditions. 3.5 3.8

Explanation of Answer AMS remains armed for 6 minutes(360 sec) following decrease below 30%(C-20). The actuation signal is generated after 3/4 SGs level have fallen 3% below the LO-2 (reactor trip) setpoints of 18% for a period of 25 seconds. C-20 would clear @ t=360sec. AMS actuation occurs at 320 + 25 = 345 sec.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
.MS	PN-3	Logic #1 schematic		2	
Chapter 60c, AMS	Chapter 60c	I.B.1.b, II.B.1.f, 2.c	2, 9	2	2

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 83 Evaluation of SR NIS voltage failure

The following conditions exist on Unit 1:

- Reactor startup in progress
- Intermediate power range indication: 2.5E-5 amp N35 & 2.8E-5 amp N36
- SOURCE RANGE PERMISSIVE P-6 permissive light clear
- Source Range Channel N31 high voltage power supply fails to HALF its normal value

What indication(s) would be available to alert the operator to this failure?

a. None, until power is lowered below the P-6 setpoint, and then the Source Range N31 indication will indicate lower than expected.

b. None, until power is lowered below the P-6 setpoint, and then the Source Range N31 indication will indicate higher than expected.

c. Annunciator SR HIGH VOLTAGE FAILURE (1-10-B1) will remain in alarm when power exceeds P-10.

d. Annunciator SR HIGH VOLTAGE FAILURE (1-10-B1) will re-flash when the voltage source fails.

Answer a **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

032 Loss of Source Range Nuclear Instrumentation

AK1. Knowledge of the operational implications of the following concepts as they apply to Loss of Source Range Nuclear Instrumentation:

AK1.01 Effects of voltage changes on performance 2.5 3.1

Explanation of Answer Based on Gas filled detector curve (Region III), the number of events collected would drop (counts drop). Alarm and voltage input to SR detector is blocked until both IR NIS fall below the P-6 setpoint.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
SR HIGH VOLT FAILURE	BAR 1-10-B1	Setpoint, NOTEs	1	51	
Source Range Detector	NI-4	SR schematic, 1950V DC to Preamp, Gas filled detector six-region curve		4	
Chp 31, Source Range Nuclear Instrumentation	Chp 31	I.B.6.c.3), III.B.2.g, h	10, 64	1	11.b

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 84 Eval of failed IR channel on SU

The following conditions exists on Unit 2:

- Plant shutdown is in progress.
- Power range channels indicate: 9% (N41), 10% (N42), 11% (N43) , 11% (N44)
- Intermediate range channel N-36 fails HIGH.

When this failure occurs, what is the plant response this failure?

a. The reactor will trip on high IR flux, and source range trip will reinstate when N-35 decreases below P-6.

b. The reactor will trip on high IR flux, and source range trip will NOT be automatically reinstated.

c. The reactor will NOT trip immediately, but will trip when the source range trip is reinstated when N-35 decreases below P-6

d. The reactor will NOT trip, and source range trip will NOT be automatically reinstated.

Answer d **Exam Level** B **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

033 Loss of Intermediate Range Nuclear Instrumentation

AA2. Ability to determine and interpret the following as they apply to Loss of Intermediate Range Nuclear Instrumentation:

AA2.04 Satisfactory overlap between source-range, intermediate-range and power-range instrumentation 3.2 3.6

Explanation of Answer Since reactor power is < P-10 setpoint (10% power), the IR trip setpoint at 25% EICA will be exceeded resulting in reactor trip. SR will NOT be reinstated automatically because only one IR channel will fall below P-6 and Two are required to remove P-6.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Intermediate Range	NI-3	IR High Flux schematic		4	
Chp 32, Intermediate Range Nuclear Instrumentation Sys	Chp 32	II.C.1, II.C.2	25-27	1	4.a, 4.c

Material Required for Examination

Question Source: New **Question Modification Method:** Significantly Modified

Question Source Comments: Watts Bar 8/94

Comment Type **Comment**

Question 85 Monitors for S/G Tube leakage

The following conditions exist on Unit 1:

- Reactor power is 75%
- Troubleshooting has commenced due to reduced condenser vacuum with the air ejectors out of service.
- Hogging vacuum pumps are aligned to the main condenser to aid in maintaining vacuum.

What would **NOT** be an indication of a Steam Generator Tube Leak under these conditions?

a. Increasing conductivity levels for the main condenser hotwell.

b. Increasing radiation level on 1RE-PR027, "SJAE/Gland Steam Exhaust Monitor".

c. Decreasing feed flow to ONE S/G.

d. Increasing radiation levels on 1RE-PR08 "S/G Blowdown Monitor".

Answer a **Exam Level** R **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

037 Steam Generator Tube Leak

AA1. Ability to operate and / or monitor the following as they apply to Steam Generator Tube Leak:

AA1.02 Condensate exhaust system 3.1 2.9

Explanation of Answer Hotwell conductivity levels should NOT be affected by SG tube leak. The Hogger discharge is aligned through the Off Gas header which is monitored by 1RE-PR027 detecting offgas activity. Blowdown would be sampled an indicate rising trend. Decrease in feed flow would be indicative of RCS inleakage to S/G (adding volume).

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Steam Generator Tube Leak	BOA SEC-8	step 5	4	54A	
hp 38D, Condenser Air Removal	Chp 38D	II.C.1, III.B.5	14, 20	1	2
Chp 49, Radiation Monitors	Chp 49	II.C.2.q	40	3	4.b.6)

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 86 Loss of subcooling

BEP-3 "Steam Generator Tube Rupture" is being performed in response to a tube rupture on 2C S/G. The cooldown has just been completed but the target temperature value selected by the operators was higher than that stipulated in the procedure.

What condition could result because of this error?

a. Loss of RCS subcooling before RCS and ruptured S/G pressures are equalized.

b. Increase in pressure of the ruptured S/G with resultant lifting of the S/G Safety Valve.

c. Increase in pressure of the non-ruptured S/Gs with resultant lifting of their S/G Safety Valves.

d. Filling the Pressurizer solid during the subsequent depressurization.

Answer a **Exam Level** B **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

038 Steam Generator Tube Rupture

EK3. Knowledge of the reasons for the following responses as they apply to Steam Generator Tube Rupture:

EK3.06 Actions contained in EOP for RCS water inventory balance, S/G tube rupture, and plant shutdown procedures 4.2 4.5

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Emergency Operating Procedures, EP-3,Steam Generator Tube Rupture, ES 3.1-3.3	BEP 3	13,	17	2	1, 8
ERG Basis				6	

aterial Required for Examination

uestion Source: New **Question Modification Method:** Editorially Modified

Question Source Comments: Salem 6/94

Comment Type **Comment**

Question 87 Steamline isolation

The following conditions exist on Unit 1:

- The Unit was in MODE 3 at normal operating temperature and pressure prior to the event.
- A faulted steam generator has occurred.
- RCS hot leg temperatures - 547°F (A), 544°F (B), 545°F (C), 547°F (D)
- RCS cold leg temperatures - 545°F (A), 530°F (B), 543°F (C), 545°F (D)
- S/G pressures - 700 psig (A), 635 psig (B), 690 psig (C), 705 psig (D)
- S/G flow - 0.85 MLB/hr (B)
- Containment pressure (Channel) - 8 psig (1), 7.5 psig (2), 7.5 psig (3), 8 psig (4)

Based on these conditions, a main steam line isolation should...

a. have occurred because of the low pressure in at least ONE S/G.

b. have occurred because the steamline high negative rate occurred in S/G 1B.

c. NOT have occurred because Containment pressure is below the setpoint for the CNMT High-2 pressure signal.

d. NOT have occurred because THREE S/Gs have pressures above the isolation setpoint and do NOT indicate high steam flow.

Answer a **Exam Level** B **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnonmal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

040 Steam Line Rupture

AA1. Ability to operate and / or monitor the following as they apply to Steam Line Rupture:

AA1.01 Manual and automatic ESFAS initiation 4.6 4.6

xplanation of nswer The steamline isolation signal is generated by the low pressure sensed on 2/3 pressure transmitters in any one SG. CNMT pressure is below the MSLI setpoint of 8.2 psig and steamline negative rate is blocked since initial condition has PZR pressure > P-11.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
ESF Setpoints	EF-2	Steamline Isolation Signals		5	
Chp. 23, Main Steam System	Chp. 23	II.C.2.c	52	2	5.d

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 88 Eval of Leak

The following conditions exist on Unit 1 following a trip from 100% power:

- Pressurizer level is 0%
- Pressurizer pressure is 1500 psig
- Containment Pressure is 16 psig.
- Tcold is 420°F for all loops.

Where is the location of the leak?

a. On one loop RCS cold leg.

b. On a Main Steam Line inside containment.

c. In a Steam Generator Tube.

d. On a feedwater line between the Feed Reg Valve and the associated Feed Water Isolation Valve, 1FW009.

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

040 Steam Line Rupture

AK1. Knowledge of the operational implications of the following concepts as they apply to Steam Line Rupture:

AK1.06 High-energy steam line break considerations 3.7 3.8

Explanation of Answer Secondary LOCA is indicated by the cold leg temperatures on all loops dropping to low temperature with the RCPs running. With LOCA expected cold leg temperature on affected loop is higher due to reversal of flow in that loop due to blowdown. Expected temperature drop is NOT as severe. The LOCA is inside CNMT as indicated by CNMT pressure rise. Feedline loop is outside CNMT.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
.mergency Procedures, BEP-0 Reactor Trip Or Safety Injection, BEP ES-0.0 - 0.4	BEP-0	I.B.4.b	12	3	6, 7
Emergency Operating Procedures, BEP-2, Faulted SG Isolation	BEP-2	7	9	2	2

Material Required for Examination

Question Source: New **Question Modification Method:** Editorially Modified

Question Source Comments: St. Lucie 10/13/97

Comment Type **Comment**

Question 89 Eval of conditions

In accordance with BOA SEC-3, "Loss of Condenser Vacuum", which of the following sets of conditions requires the operator to trip the reactor?

a. LOW POWER TRIP BLOCKED P-8 annunciator - LIT
Turbine load - 200 MW
Condenser pressure - 5.2 " HgA

b. LOW POWER TRIP BLOCKED P-8 annunciator - LIT
Turbine load - 300 MW
Condenser pressure - 6.3" HgA

c. LOW POWER TRIP BLOCKED P-8 annunciator - CLEAR
Turbine load - 600 MW
Condenser pressure - 7.2" HgA

d. LOW POWER TRIP BLOCKED P-8 annunciator - CLEAR
Turbine load - 900 MW
Condenser pressure - 7.8" HgA

Answer b **Exam Level** B **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98
Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

051 Loss of Condenser Vacuum

AA2. Ability to determine and interpret the following as they apply to Loss of Condenser Vacuum:

AA2.02 Conditions requiring reactor and/or turbine trip 3.9 4.1

Explanation of Answer P-8 permissive active below 30% power (annunciator lit). At 480 MW and below, the minimum acceptable condenser pressure is 5.5 in HgA. At 600 MW minimum acceptable pressure is 7. 8 in HgA. At 610 MW and greater, minimum acceptable pressure is 8.0 in HG

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Loss of Condenser Vacuum	BOA SEC-3	step 5, Figure 1BOA SEC-3-1	5		
Abnormal Operating Procedures, Loss Of Condenser Vacuum	BOA SEC-3	5, Q4	11	1	5, 6

Material Required for Examination Figure 1BOA SEC 3-1
Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 90 Identification of RCP seal LOCA/cooldown

Select the primary reason for rapidly depressurizing the steam generators during a Loss of All AC.

a. To provide maximum core cooling until power can be restored.

b. To minimize RCS inventory loss from RCP seals.

c. To enhance restoration of S/G level from the diesel driven AF pump.

d. To increase subcooling of the RCS.

Answer b **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

055 Station Blackout

EK3. Knowledge of the reasons for the following responses as they apply to Station Blackout:

EK3.02 Actions contained in EOP for loss of offsite and onsite power 4.3 4.6

Explanation of Answer The rapid cooling allows depressurizing the RCS reducing the leak rate via the RCP seals

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
LOSS OF ALL AC POWER	BCA-0.0	CAUTION 2 step 10	70	1A WOG-1 B	
BCA-0.0, 0.1, 0.2, 0.3 LOSS OF ALL AC POWER	BCA-0.0	CAUTION	70	3	4, 5

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 91 Reset of sequencer

How would the sequencer operate if a Safety Injection (SI) actuation occurs while the sequencer is sequencing loads in response to an ESF bus undervoltage condition?

a. There will be no change in operation; the undervoltage sequence overrides the SI sequence.

b. The undervoltage sequencing stops, the sequencer immediately resets and SI loads NOT already running will sequentially start.

c. The undervoltage sequencing stops, all started loads are shed, and SI loads will sequentially start.

d. The undervoltage sequencing completes its cycle, then resets to SI mode, and SI loads NOT already running will sequentially start.

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 3 **SRO Group:** 3

056 Loss of Off-Site Power

AA1. Ability to operate and / or monitor the following as they apply to Loss of Off-Site Power:

AA1.21 Reset of the ESF load sequencers 3.3 3.3

Explanation of Answer The UV sequence is stopped and the SARA sequencing is initiated from step 1.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
D/G Relaying	DG-2	SARA & SSR schematics, On a SI Signal Coincident with Loss of Off-site Power		1	
hp 9, Diesel Generators & Aux. Systems	Chp 9	III.D.1	47-48	1	7.c

Material Required for Examination

Question Source: New **Question Modification Method:** Significantly Modified

Question Source Comments: Vogtle - 5/91

Comment Type Comment

Question 92 Eval of electric bus status

The following conditions exist on Unit 1:

- Bus 141 is powered from its normal source
- D/G 1A surveillance is being performed with the D/G paralleled to the bus

What would occur if a failure of the undervoltage relay results in a sensed undervoltage condition on Bus 141?

a. SAT feeder breaker ACB 1412 and D/G feeder breaker ACB 1413 remain closed. The Safe Shutdown loads will NOT sequence and CANNOT be manually started from the control room.

b. SAT feeder breaker ACB 1412 and D/G feeder breaker ACB 1413 will open. After a 10-second delay, ACB 1413 will close and the Safe Shutdown loads will sequence.

c. SAT feeder breaker ACB 1412 will open but D/G feeder breaker ACB 1413 will remain closed. The Safe Shutdown loads will sequence normally.

d. SAT feeder breaker ACB 1412 will open but D/G feeder breaker ACB 1413 will remain closed. The Safe Shutdown loads will NOT sequence and CANNOT be manually started from the control room.

Answer d **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 3 **SRO Group:** 3

056 Loss of Off-Site Power

AA2. Ability to determine and interpret the following as they apply to Loss of Off-Site Power:

AA2.46 That the ED/Gs have started automatically and that the bus tie breakers are closed 4.2 4.4

xplanation of Answer On sensed UV, the SAT feeder breaker opens (and alternate feeder breaker would also have opened if closed) and the control switches for the safe shutdown loads will be locked out due to constant undervoltage signal.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Chp 4, A.C. Electrical Power Distribution System	Chp 4	II.A.2.h.7).b)	62, 67	2	10.a, 11

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 93 Eqpt affected on bus loss

On Unit 1 power is lost to 120 VAC Instrument Bus 111

How are the ESF and Safe Shutdown loads affected?

- a. "A" Train ESF loads will NOT load on an SI signal, but Safe Shutdown loads will load on a U/V signal.

  "B" Train loads are NOT affected.
- b. A" Train ESF loads will load on an SI signal, but Safe Shutdown loads will NOT load on a U/V signal.
  "B" Train loads are NOT affected.
- c. "A" Train ESF loads will NOT load on an SI signal, and Safe Shutdown loads will NOT load on a U/V signal.
  "B" Train loads are NOT affected.
- d. "A" Train AND "B" Train ESF loads will NOT load on an SI signal, but Safe Shutdown loads will load on a U/V signal.

Answer C **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

057 Loss of Vital AC Instrument Bus

AA2. Ability to determine and interpret the following as they apply to Loss of Vital AC Instrument Bus:

AA2.19 The plant automatic actions that will occur on the loss of a vital ac electrical instrument bus 4.0 4.3

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Loss of Instrument Bus	1BOA ELEC-2	Table A, 1.a	7	56	
BOA ELEC-2	BOA ELEC-2	Table A.a	7	2	3.c, 7
Chp 60a - SSPS	Chp 60a	II.B.5		1	4, 9.b

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 94 Operations required for transfer

Which of the following sets of indications are available on the Remote Shutdown Panel?

- a. Emergency boration flow, S/G level, and RCS wide range temperature.
- b. Red and green lights for reactor trip breaker position indication, S/G pressure, and pressurizer level.
- c. Main feedwater flow, letdown flow, and charging line pressure.
- d. Containment pressure, charging flow, and auxiliary feedwater flow.

Answer a **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

068 Control Room Evacuation

AA1. Ability to operate and / or monitor the following as they apply to Control Room Evacuation:

AA1.12 Auxiliary shutdown panel controls and indicators 4.4 4.4

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
RSP PL04/5J	PN-1	Diagram		2	
CONTROL ROOM INACCESSIBILITY	1BOA PRI-5	Attachment A, step 1	1 (40)	57B	
Chp 62, Remote Shutdown Panel	Chp 62	II.A.1.a, b	10	3	4

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 95 Major action categories

When inadequate core cooling exists, which of the following sets of actions states the proper sequence of the major action categories to be performed in accordance with BFR-C.1, "RESPONSE TO INADEQUATE CORE COOLING", for removing decay heat from the core?

a. Rapid secondary depressurization; reinitiation of safety injection; RCP restart.

b. Reinitiation of safety injection; rapid secondary depressurization; RCP restart.

c. Rapid secondary depressurization; RCP restart ; reinitiation of safety injection.

d. RCP restart; rapid secondary depressurization; reinitiation of safety injection.

Answer b **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

074 Inadequate Core Cooling

EK1. Knowledge of the operational implications of the following concepts as they apply to Inadequate Core Cooling:

EK1.03 Processes for removing decay heat from the core 4.5 4.9

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Function Restoration Procedures BFR-C.1, C.2, C.3	BFR-C.1	C	1	3	2, 3

Material Required for Examination

Question Source: New **Question Modification Method:** Editorially Modified

Question Source Comments: VC Summer 5/94

Comment Type **Comment**

Question 96 Actions for reducing activity

High coolant activity has been detected and chemistry has determined that it is due to corrosion product activation.

Identify the effect of placing the cation demineralizer in service.

The cation demineralizer...

a. will remove lithium so it should NOT be used in this condition.

b. will cause the activity level to decrease as soon as it is placed in service.

c. is NOT effective in removing corrosion product activity.

d. is less effective than the mixed bed demineralizer so it is placed in service ONLY if decontamination factor is less than 10.

Answer b or A **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

076 High Reactor Coolant Activity

AA2. Ability to determine and interpret the following as they apply to High Reactor Coolant Activity:

AA2.02 Corrective actions required for high fission product activity in RCS 2.8 3.4

Explanation of Answer The cation demin is highly effective in removing corrosion products from the coolant.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Abnormal Operating Procedures BOA PRI-4, Abnormal Primary Chemistry	1BOA PRI-4	B	1	1	1
Chp 15a - Chemical and Volume Control System	Chp 15a	II.A.1.k.3)	27	02	4

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 97 Interlocks affecting reestablishment of feed

The following conditions exist on Unit 2:

- Reactor power was 8% prior to the event below.
- A failure in the feedwater control system caused ONE S/G level to rise to 83%.
- The main turbine tripped.
- S/G levels have returned to their normal level range
- The Startup FW Pump is running

What are all the conditions that would have to be met to feed the S/Gs using the FW034's Feedwater Tempering Flow Control valves?

a. The FW Isolation Aux Relays would have to be reset and FW035 Feedwater Tempering Isol valves opened.

b. The reactor trip breakers would have to be cycled, the FW Isolation Aux Relays would have to be reset and FW035 Feedwater Tempering Isol valves opened.

c. The FW Isolation Main Relays and Aux Relays would have to be reset and FW035 Feedwater Tempering Isol valves opened.

d. The reactor trip breakers would have to be cycled and FW Isolation Main Relays and Aux Relays reset and FW035 Feedwater Tempering Isol valves opened.

Answer a **Exam Level** B **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

E05 Loss of Secondary Heat Sink

EK2. Knowledge of the interrelations between Loss of Secondary Heat Sink and the following:

EK2.1 Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. 3.7 3.9

Explanation of Answer The P-14 signal, once clear, only maintains FWI signal via the FW Isol Aux relays if NO reactor trip signal is present. So resetting the FW Isolation Aux relay allows opening of FW035s (normal feed path at low power) and throttling of FW034s

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
ESF Setpoints	EF-2	To reset a FW Isolation after a P-14		5	
Feedwater Simple / SGWLC	FW-1 / FW-2	FWI signals, Pump interlocks, Flowpaths during startup		0	
Chp 61,Engineered Safety Features	Chp 61	II.C.6.a		2	7.c

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 98 Identification of heat removal process

The following conditions exist on Unit 1:

- A leak developed on the RCS loop C flow instrument piping.
- Coincident with the RCS leak, on the reactor trip a S/G PORV failed open and was later isolated.
- FR-P.1 was entered to due to an ORANGE PATH condition.
- SI actuated and has been reset.
- All RCPs are stopped.
- Conditions required to support an RCP start are met.

Under the current conditions starting the RCP will...

a. cause excessive thermal stresses in the stagnant loops.

b. cause a pressure surge that will aggravate the PTS condition.

c. provide mixing of the ECCS injection flow thereby decreasing the likelihood of PTS.

d. increase the RCS cooldown rate thereby increase the likelihood of PTS.

Answer C **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

E08 Pressurized Thermal Shock

EK2. Knowledge of the interrelations between Pressurized Thermal Shock and the following:

EK2.2 Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility. 3.6 4.0

Explanation of nswer

Reference Title	Facility Reference Number	Section	Page Number(s)	Revision	L. O.
Function Restoration Procedures BFR-P.1, P.2	BFR-P.1	5	19	0	3, 5
Status Trees	ST-1	Integrity		1	

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 99 Natural Circ conditions and limits

The following conditions exist on Unit 1:

- A natural circulation is in progress per BEP ES-0.2 "Natural Circulation Cooldown"
- Pressurizer pressure is being controlled using Aux. Spray and Pzr heaters
- As pressure is being lowered through 1300 psig, a rapid increase is noted in Pzr level
- Charging and letdown are in manual and are balanced

What actions are required to be taken by the operators?

a. Repressurize the RCS.

b. Isolate the SI Accumulators.

c. Increase the RCS cooldown rate.

d. Place excess letdown in service.

Answer a **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

E09 Natural Circulation Operations

EK3. Knowledge of the reasons for the following responses as they apply to Natural Circulation Operations:

EK3.1 Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics. 3.3 3.

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Emergency Procedures BEP-0 Reactor Trip or Safety Injection BEP ES-0.0 - 0.4	BEP ES-0.2 - 0.4	5, 12	3, 8	3	3, 4
Natural Circulation Cooldown	BEP ES-0.2	Step 14	10	1, WOG-1 B	

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 100 Reason for rapid S/G depressurization

Why are the S/Gs depressurized to less than 675 psig according to BCA-1.1, "Loss of Emergency Coolant Recirculation"?

a. To allow maximum AFW flow to the S/Gs.

b. To ensure adequate subcooling for restart of the RCPs.

c. To set up conditions for controlled injection to the RCS from the accumulators.

d. To decrease RCS temperature and pressure which reduces break flow in a LOCA condition.

Answer C **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

E11 Loss of Emergency Coolant Recirculation

EA1. Ability to operate and / or monitor the following as they apply to Loss of Emergency Coolant Recirculation:

EA1.1 Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. 3.9 4.0

Explanation of Answer The concern is maximizing cooling volumes that supply water to RCS. By cooling RCS, depressurization of RCS can be initiated (while maintaining subcooling) to the point where the SI accumulators inject their volumes into the RCS.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Loss of Emer Coolant Recirc	BCA 1.1	28	24	1B WOG 1B	
BCA-Contingency Action 1.1 and 1.2	BCA-1.1	28	24	2	4

Material Required for Examination

Question Source: New **Question Modification Method:** Editorially Modified

Question Source Comments: South Texas 9/92

Comment Type **Comment**

GENERIC FUNDAMENTALS EXAMINATION
EQUATIONS AND CONVERSIONS HANDOUT SHEET

EQUATIONS

$\dot{Q} = \dot{m}C_p\Delta T$

$\dot{Q} = \dot{m}\Delta h$

$\dot{Q} = UA\Delta T$

$\dot{Q} \propto \dot{m}^3_{\text{Nat Circ}}$

$\Delta T \propto \dot{m}^2_{\text{Nat Circ}}$

$K_{eff} = 1/(1 - \rho)$

$\rho = (K_{eff} - 1)/K_{eff}$

$SUR = 26.06/\tau$

$$\tau = \frac{\overline{\beta} - \rho}{\lambda_{eff}\,\rho}$$

$$\rho = \frac{\ell^*}{\tau} + \frac{\overline{\beta}}{1 + \lambda_{eff}\tau}$$

$\ell^* = 1 \times 10^{-4}$ seconds

$\lambda_{eff} = 0.1$ seconds^{-1}

$P = P_o 10^{SUR(t)}$

$P = P_o e^{(t/\tau)}$

$A = A_o e^{-\lambda t}$

$CR_{S/D} = S/(1 - K_{eff})$

$CR_1(1 - K_{eff1}) = CR_2(1 - K_{eff2})$

$1/M = CR_1/CR_X$

$DRW \propto \phi^2_{tip}/\phi^2_{avg}$

$F = PA$

$\dot{m} = \rho A\hat{v}$

$\dot{W}_{Pump} = \dot{m}\Delta P \upsilon$

$E = IR$

Eff. = Net Work Out/Energy In

$$\upsilon(P_2 - P_1) + \frac{(\bar{v}_2^2 - \bar{v}_1^2)}{2g_c} + \frac{g(z_2 - z_1)}{g_c} = 0$$

$g_c = 32.2$ lbm-ft/lbf-sec^2

CONVERSIONS

1 Mw = 3.41×10^6 Btu/hr

1 hp = 2.54×10^3 Btu/hr

1 Btu = 778 ft-lbf

°C = (5/9)(°F - 32)

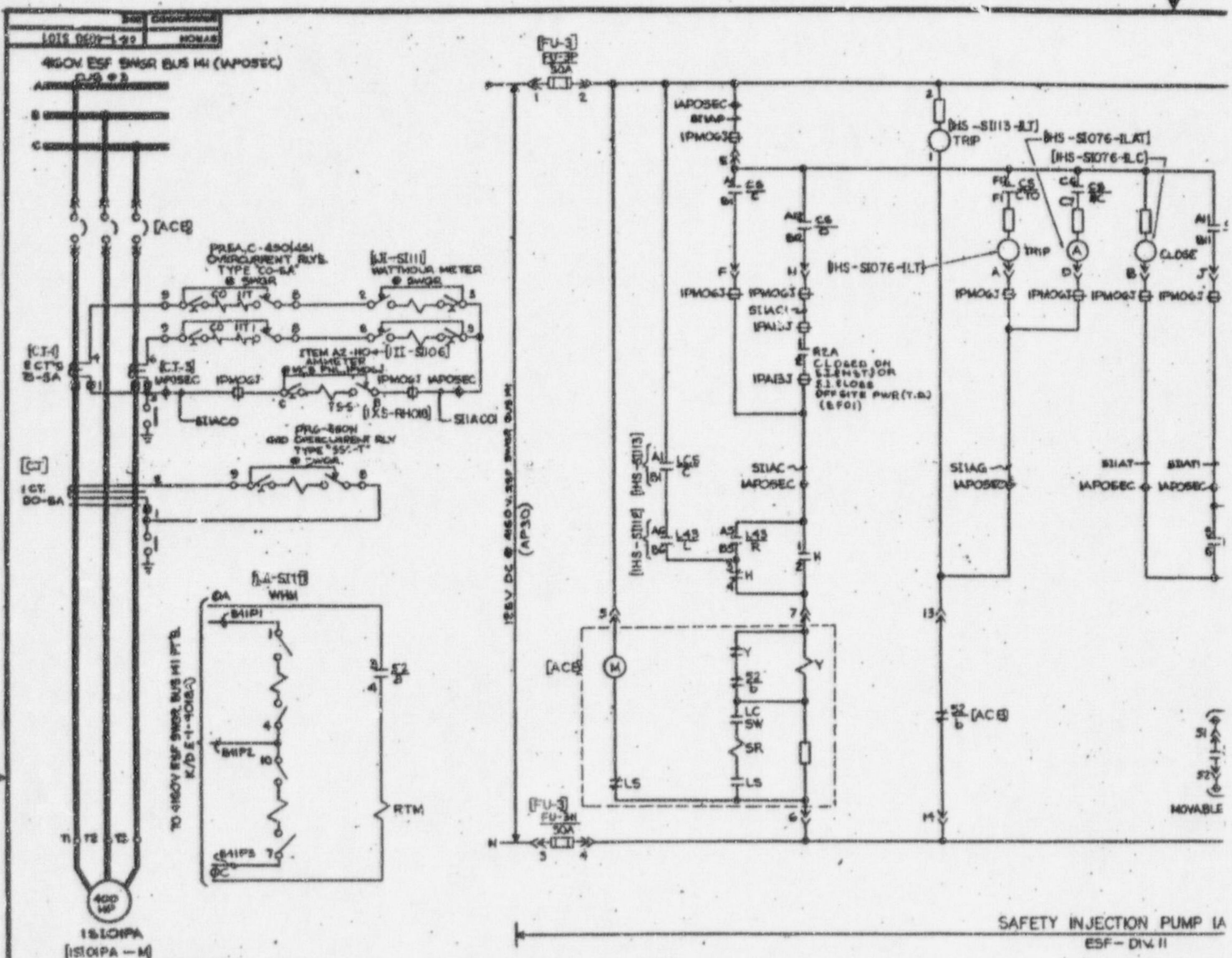
°F = (9/5)(°C) + 32

1 Curie = 3.7×10^{10} dps

1 kg = 2.21 lbm

1 $\text{gal}_{\text{water}}$ = 8.35 lbm

1 $\text{ft}^3_{\text{water}}$ = 7.48 gal

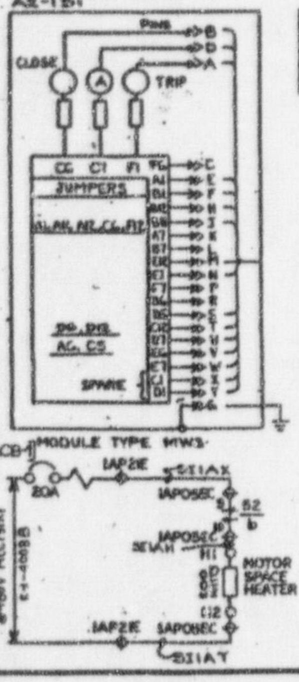


SAFETY INJECTION PUMP IA

ESF- DIV. II

ITEM 1 [1HS-SI076]

A2-1B1



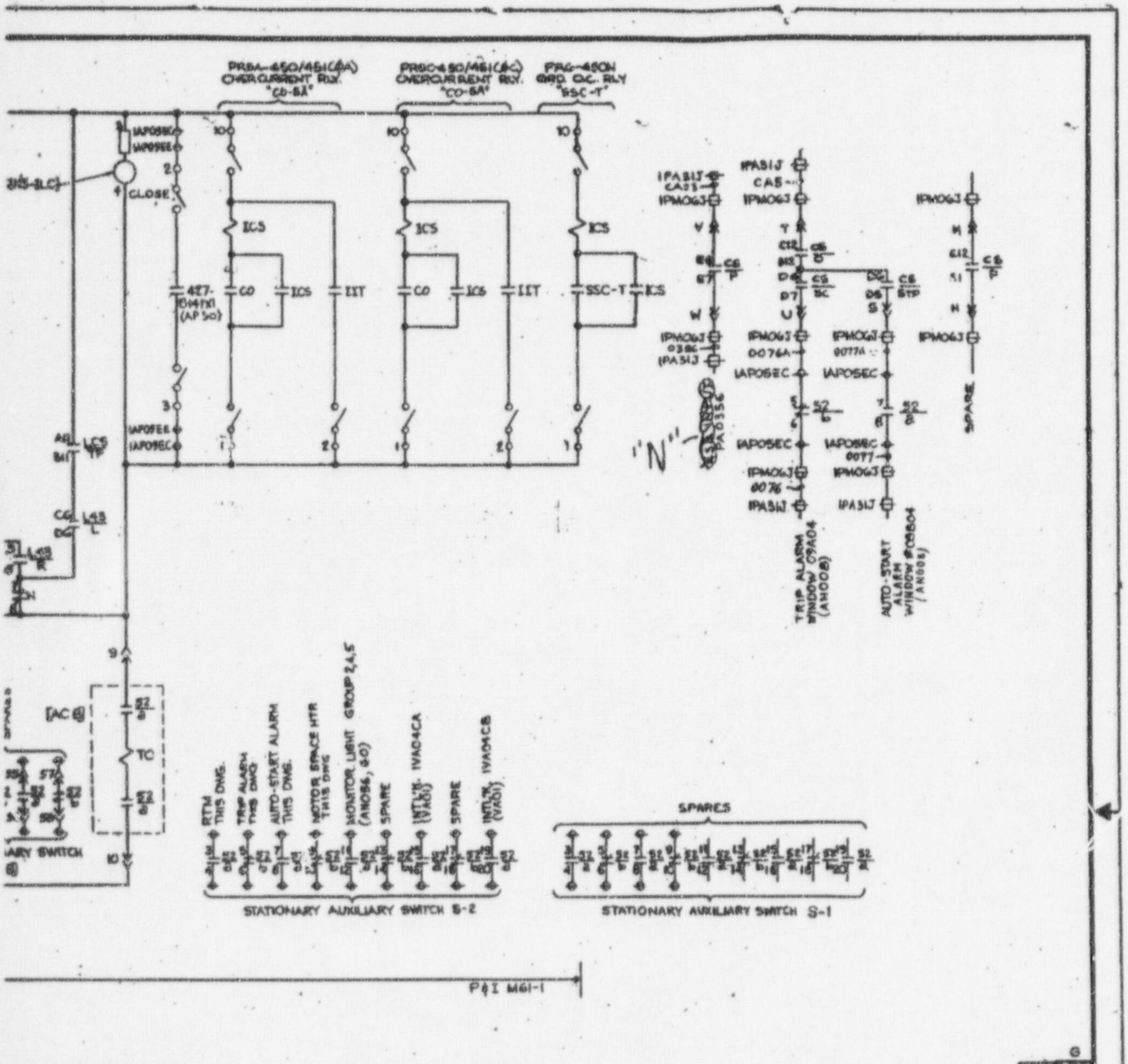
(P.V.)

CONTACT	SYM.	POSITION					
		PULL OUT	TRIP	OFF		CLOSE	
				AFT TRIP	AFT CLOSE		
A11 - B11	TP	X	X				*
A12 - B12	O			X	X		*
A1 - B1	C					X	*
A5 - B5	TP	X	X				
A6 - B6	O			X	X		
A7 - B7	C					X	
C11 - D11	TP	X	X				
C12 - D12	O			X	X		*
C1 - D1	C					X	
C5 - C6	STP	X	X	X			
C6 - C7	SC				X	X	*
D8 - D6	STP	X	X	X			*
D6 - D7	SC				X	X	*
E12 - E1	P	X					
F12 - F1	CTO		X	X	X	X	*
E6 - E7	P	X					*
F6 - F7	CTO		X	X	X	X	

WESTINGHOUSE TYPE W-2 CONTROL SWITCH, 3 POSITIONS, SPRING RETURN TO CENTER, BASIC SWITCH S#50BA864G02, PISTOL GRIP FIXED HANDLE

X - DENOTES CONTACT CLOSED
* - DENOTES SWITCH CONTACT USED

CONTROL SWITCH DEVELOPMENT 'CS'
@ MCB PNL 1PM06J



MPD OK | MPD PT

G

REFERENCE DWGS.	
DWG. NO.	DESCRIPTION
E+4054 SERIES	INT/EXT W/D MCB ENG. SAFETY FEATURES 1PM06J
E+4122 SERIES	INT/EXT W/D ESF SEQUENCING AND ACTUATION CAB-TRAIN A 1PA13J
E+4155 SERIES	INT/EXT W/D ANNUNCIATOR INPUT CAB. (ESF II) 1PA51J
E+4611 SERIES	INT/EXT W/D 4160V ESF SWGR. BUS 141 1AP05E
E+4030 SERIES	S/D NOTES, LEGENDS, REFERENCE DRAWINGS
E+4661 SERIES	INT/EXT W/D 480V MCC 131X1 (1AP21E)
E+4665 SERIES	INT/EXT W/D 480V MCC 131X3 (1AP22E)

DRAWING RELEASE RECORD

REV.	DATE	DESCRIPTION	ENGR. APPROVAL
M	[illegible] 87	ADDED COMPONENT TAGS REL. FOR RECORD ONLY.	
N	9/1/87	[illegible]	
L	1-8-87	ADDED COMPONENT TAGS REL. FOR RECORD ONLY	

DRAWN
CHECKED
ENGR. APPROVAL

NUCLEAR SAFETY RELATED
EQUIPMENT IS SHOWN ON THIS DRAWING.

REF. W.E.CO. DWG. 271C620 SHT. 26

SCHEMATIC DIAGRAM
SAFETY INJECTION PUMP 1A -
1SI01PA

BYRON/[redacted] STATION —UNIT 1
COMMONWEALTH EDISON CO.
CHICAGO, ILLINOIS

SCALE NONE
DRAWN 8/30/74
CHECKED 3-15-75
ENGINEER 3-15-75
APPROVED
JOB NO. 4391

SARGENT & LUNDY
ENGINEERS
CHICAGO

DRAWING NO.
BYRON 6E-1-4030 SI01
[redacted] DWG.

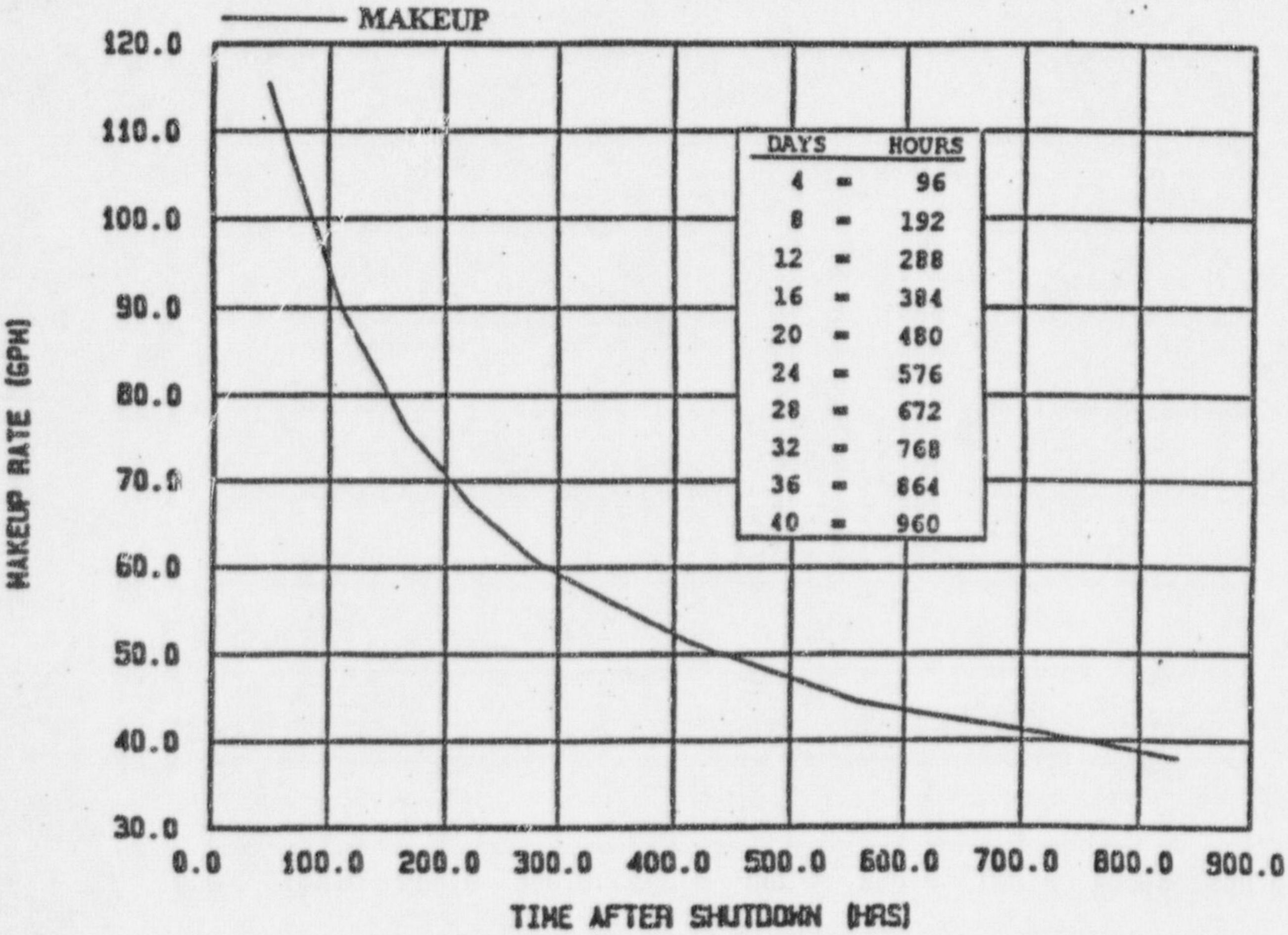
17830

BYH-5236(K), BBHE-10588(H) [illegible](J) BYH-5568(L)
BYH-5810(M) BBHE-8147(G)
BBHE-7745(F)

07

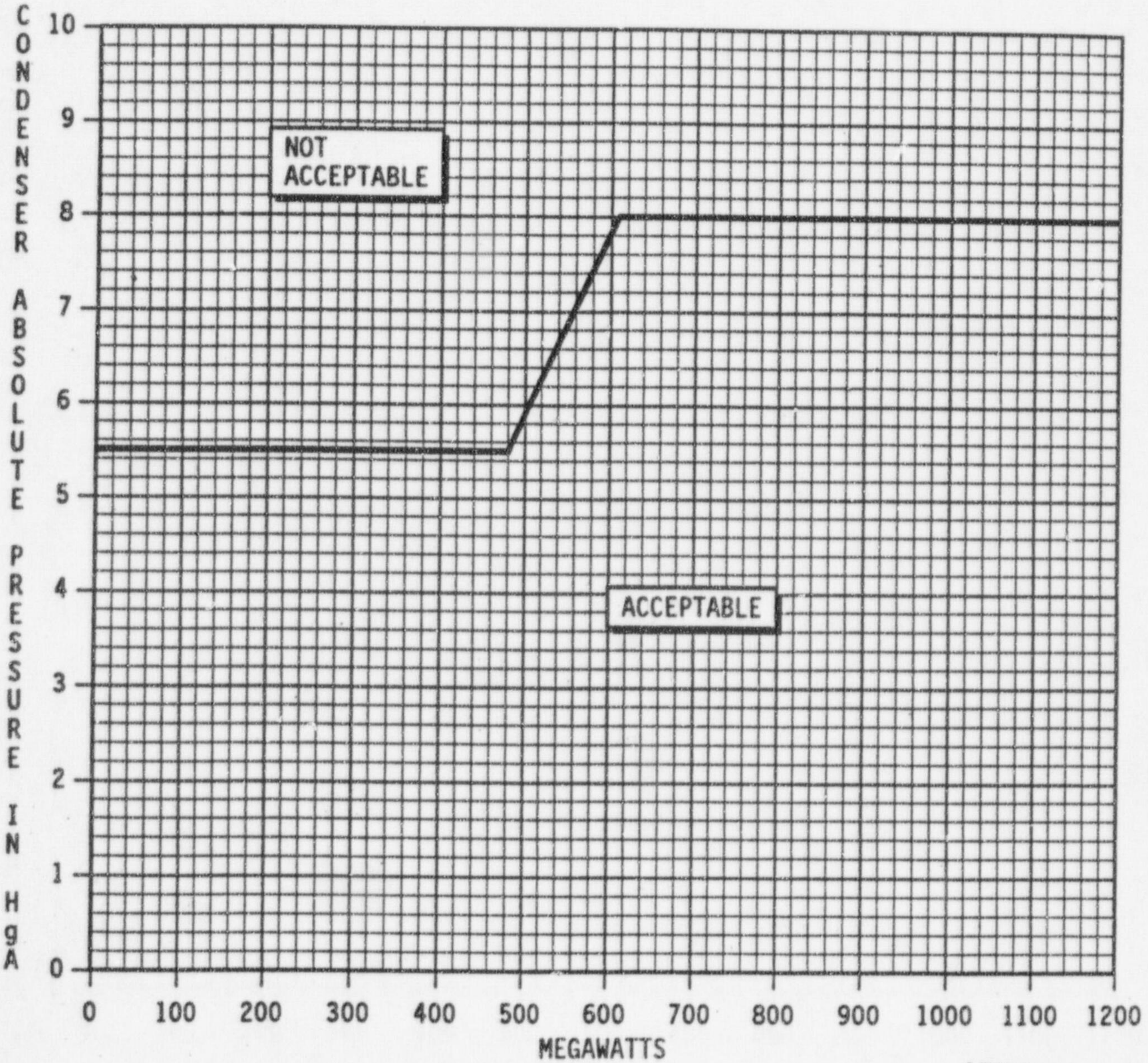
A

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5	VERIFY ALL CONTROL RODS FULLY INSERTED: a. All rod bottom lights - LIT	Perform the following: a. IF two or more rods are NOT fully inserted, THEN borate 1320 GAL (5500 GAL FROM RWST) for each rod NOT fully inserted per 1BOA PRI-2, EMERGENCY BORATION. b. Within 1 HOUR calculate Shutdown Margin per 1BOS 1.1.1.1.e-1, SHUTDOWN MARGIN SURVEILLANCE (ITS 1BOSR 3.1.1.1).



APPROVED
OCT 2 4 1997
B.O.S.R.

FIGURE 1BOA PRI 10-1
MINIMUM MAKEUP FLOW REQUIRED TO MATCH BOIL-OFF



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FEB 0 3 1994
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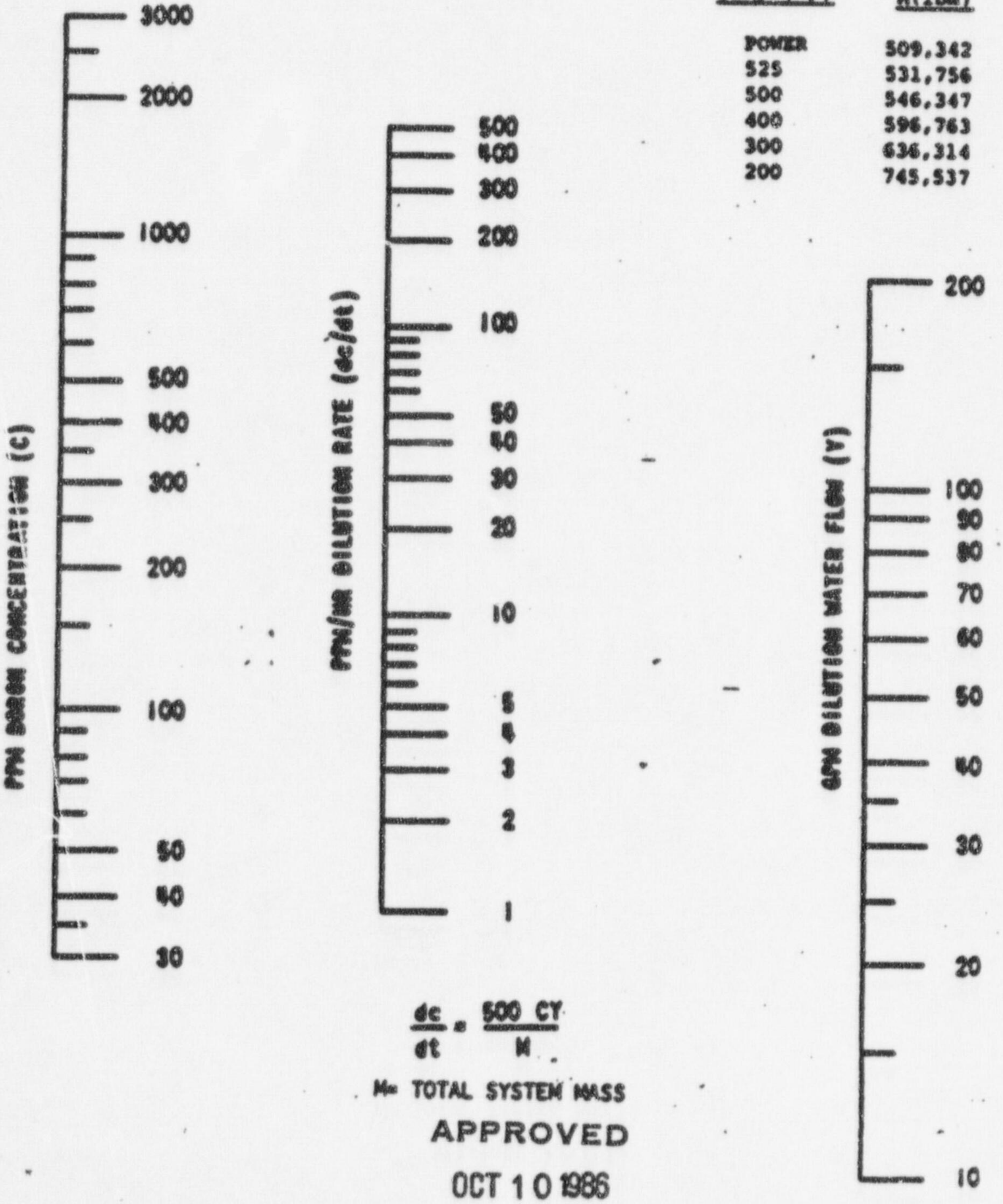
FIGURE 1BOA SEC-3-1
TURBINE LOAD -vs- CONDENSER PRESSURE

BORON DILUTION RATE NOMOGRAPH

TEMP (°F)	M(lbm)
POWER	509,342
525	531,756
500	546,347
400	596,763
300	636,314
200	745,537



$$\frac{dc}{dt} = \frac{500\ CY}{M}$$

M= TOTAL SYSTEM MASS

APPROVED
OCT 10 1986
B.O.S.R.

U.S. Nuclear Regulatory Commission
Site-Specific
Written Examination

Applicant Information

Name: MASTER EXAMINATION	Region: III
Date: SEPTEMBER 14, 1998	Facility/Unit: Byron Units 1 and 2
License Level: SRO	Reactor Type: W
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. The passing grade requires a final grade of at least 80.00 percent. Examination papers will be collected four hours after the examination starts.

Applicant Certification

All work done on this examination is my own. I have neither given nor received aid.

____
Applicant's Signature

Results

Examination Value 99. ~~100~~ Points

Applicant's Score __________ Points

Applicant's Grade __________ Percent

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DATE		HOUR	

TEST RECORD

PART 1	
PART 2	
TOTAL	

SUBJECTIVE SCORE INSTRUCTOR USE ONLY

100	90	80	70	60
50	40	30	20	10
9	8	7	6	5
4	3	2	1	0

IMPORTANT

- MAKE DARK MARKS
- ERASE COMPLETELY TO CHANGE
- EXAMPLE

TO USE SUBJECTIVE SCORE FEATURE

- Mark total possible subjective points
- Only one mark per line on key
- 150 points maximum

EXAMPLE OF STUDENT SCORE

PART 1

	(T)	(F)			KEY	
	%	●	3		●	
1	A	B	●	D	E	
2	A	B	●	D	E	
~~3~~	~~A~~	~~B~~	~~●~~	~~D~~	~~E~~	Delete ✓ HP
4	A	B	C	●	E	
5	●	B	C	D	E	
6	A	B	●	D	E	
7	A	B	●	D	E	
8	●	B	C	D	E	
9	A	B	C	●	E	
10	~~●~~	●	C	D	E	B ✓ HP
11	A	●	C	D	E	
12	●	B	C	D	E	
13	A	●	C	D	E	
14	A	B	C	●	E	
15	A	●	C	D	E	
16	●	B	C	D	E	
17	●	B	●	D	E	(C or A) ✓ HP
18	●	B	C	D	E	
19	A	B	●	D	E	
20	A	B	●	D	E	
21	A	B	C	●	E	
22	A	B	C	●	E	
23	A	B	●	D	E	
24	A	B	●	D	E	
25	A	B	C	●	E	
26	A	B	C	●	E	
27	●	B	C	D	E	
28	A	B	C	●	E	
29	A	●	C	D	E	
30	A	B	C	●	E	
31	●	B	C	D	E	
32	A	B	●	D	E	
33	A	●	C	D	E	
34	A	B	C	●	E	
35	A	B	●	D	E	
36	A	●	C	D	E	
37	A	●	C	D	E	
38	●	B	C	D	E	
39	A	B	C	●	E	
40	A	B	C	●	E	
41	●	B	C	D	E	
42	A	●	C	D	E	
43	A	B	●	D	E	
44	A	●	C	D	E	
45	A	●	C	D	E	
46	A	●	C	D	E	
47	A	●	C	D	E	
48	A	●	C	D	E	
49	A	B	●	D	E	
50	●	B	C	D	E	

FEED THIS DIRECTION

385-P 9999 12 11 10 9 8 7 6 5 3 2 1

BYRON

1998 INITIAL LICENSE TRAINING

Description: NRC SENIOR REACTOR OPERATOR FINAL EXAM

Total Points: 100.0 Points Received: _______

NRC APPROVED

Name/Date: ANSWER KEY HP ___/___/___

Instructions

1. Use black ink only on all portions of the exam package EXCEPT for the scantron answer selections.
2. Print your name and date in the space provided above.
3. If you have any questions or need clarification during the examination, notify the proctor.
4. Conversation during the examination is prohibited except when asking for question clarification from the proctor.
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_______________________ signature

___/___/___ date

SUBJECTIVE SCORE INSTRUCTOR USE ONLY

100	90	80	70	60
50	40	30	20	10
9	8	7	6	5
4	3	2	1	0

IMPORTANT

- MAKE DARK MARKS
- ERASE COMPLETELY TO CHANGE
- EXAMPLE

TO USE SUBJECTIVE SCORE FEATURE

- Mark total possible subjective points
- Only one mark per line on key
- 150 points maximum

EXAMPLE OF STUDENT SCORE

PART 2

NAME

SUBJECT **TEST NO.**

DATE **HOUR**

TEST REC

PART 1

PART 2

TOTAL

FEED THIS DIRECTION

	(T)	(F)			KEY
	%	●	3	●	●
51	A	●	C	D	E
52	A	B	●	D	E
53	A	B	C	●	E
54	A	B	C	●	E
55	A	B	C	●	E
56	A	●	C	D	E
57	A	B	C	●	E
58	●	B	C	D	E
59	☒	B	C	●	E
60	●	B	C	D	E
61	A	●	C	D	E
62	A	B	●	D	E
63	●	B	C	D	E
64	A	B	●	D	E
65	●	B	C	D	E
66	A	B	●	D	E
67	●	B	C	D	E
68	●	B	C	D	E
69	A	B	●	D	E
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78	A	B	C	●	E
79	A	●	C	D	E
80	A	B	C	●	E
81	A	●	C	D	E
82	●	B	C	D	E
83	A	B	C	●	E
84	A	B	●	D	E
85	●	B	C	D	E
86	●	B	C	D	E
87	A	●	C	D	E
88	A	●	C	D	E
89	A	B	●	D	E
90	A	●	C	D	E
91	A	●	C	D	E
92	A	B	C	●	E
93	A	B	●	D	E
94	●	B	C	D	E
95	A	●	C	D	E
96	●	●	C	D	E
97	●	B	C	D	E
98	A	B	●	D	E
99	●	B	C	D	E
100	A	B	●	D	E

D ✓

50

A is also correct (A or B) ✓

Senior Reactor Operator Answer Key

1 .c
2 .c
~~3 .c~~ delete
4 .d
5 .a
6 .c
7 .c
8 .a
9 .d
10 ~~.a~~ b
11 .b
12 .a
13 .b
14 .d
15 .b
16 .a
17 .c or A
18 .a
19 .c
20 .c
21 .d
22 .d
23 .c
24 .c
25 .d

26 .d
27 .a
28 .d
29 .b
30 .d
31 .a
32 .c
33 .b
34 .d
35 .c
36 .b
37 .b
38 .a
39 .d
40 .d
41 .a
42 .b
43 .c
44 .b
45 .b
46 .b
47 .b
48 .b
49 .c
50 .a

Senior Reactor Operator Answer Key

51 .b
52 .c
53 .d
54 .d
55 .d
56 .b
57 .d
58 .a
59 .~~a~~ d
60 .a
61 .b
62 .c
63 .a
64 .c
65 .a
66 .c
67 .a
68 .a
69 .c
70 .a
71 .d
72 .d
73 .c
74 .b
75 .b

76 .d
77 .b
78 .d
79 .b
80 .d
81 .b
82 .a
83 .d
84 .c
85 .a
86 .a
87 .b
88 .b
89 .c
90 .b
91 .b
92 .d
93 .c
94 .a
95 .b
96 .b or A
97 .a
98 .c
99 .a
100 .c

NOTE: DISREGARD THE PAGE NUMBERS AT THE BOTTOM ON EACH PAGE.

THE QUESTION NUMBER ORDER IS AT THE TOP LEFT ON EACH PAGE.

Question 1 Evaluation of requirement for "active" license

An "Active" licensed NSO (original license obtained in 1996) worked the following schedule at Byron:

- 9/4 - 0700 to 1500 as Unit 1 NSO
- 9/7 - 0700 to 1500 as Unit 2 NSO
- 9/8 - 0700 to 1500 as Unit 2 NSO
- 9/9 - 0700 to 1200 as Unit 2 NSO and 1200 to 1500 as WEC NSO
- 9/10 - 0700 to 1500 as WEC NSO
- 9/11 - 0700 to 1500 as Unit 1 NSO
- 9/14 - 1500 to 2200 as Unit 2 NSO
- 9/12 - 1500 to 2200 as Unit 2 NSO

The NSO...

a. meets the requirements for maintaining his/her license active for the next quarter.

b. needs to work an additional FOUR hour shift to maintain his/her license active for the next quarter.

c. needs to work an additional EIGHT hour shift to maintain his/her license active for the next quarter.

d. needs to work TWO additional EIGHT hour shifts to maintain his/her license active for the next quarter.

Answer C **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.1 Conduct of Operations

2.1.1 Knowledge of conduct of operations requirements. 3.7 3.8

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Operators' "License" Information	Policy 400-12	I.A.1	2	71	
OPERATING SHIFT TURNOVER AND RELIEF	BAP 335-1	C.1.e	3	21	
Administrative Procedures - BAP 335-1	335-1r4	I.C.1.e	6	4	1

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 2 Direction of NLO personnel

The following conditions on Unit 1:

- Reactor power 45%
- 1A and 1C Feedwater pumps are operating
- FW PUMP TURB BRNG OIL LEVEL HIGH LOW annunciator (1-16-D3) alarms and the SER monitor indicates a low level.
- An EA is dispatched and confirms a low level exists.

In performing actions to correct the condition (per BOP TO-08 "Filling a Turbine Feed Pump Oil Reservoir"), what is the normal relationship between the US, the NSO and the EA?

a. The US will direct the EA's activities, but will inform the Unit NSO before the job commences.

b. The US will direct the EA's activities, and need NOT inform the Unit NSO unless unit controls are affected.

c. The Unit NSO will direct the EA's activities, but will inform the US before the job commences.

d. The Unit NSO will direct the EA's activities, and need NOT inform the US unless unit load is affected.

Answer C **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.1 Conduct of Operations

2.1.1 Knowledge of conduct of operations requirements. 3.7 3.8

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Conduct Of Operations	BAP 300-01	C.2.b.4) & C.4.a.2)	14 & 20	14	
ADMINISTRATIVE PROCEDURES LESSON 300-1 & 340-1	BAP 300-1 and 340-1	I.C.2.b).4)	34-35	1	7

Material Required for Examination **Number(s)** n

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 3 Operating Daily Orders

How is a procedure change, which significantly changes normal processes, procedurally conveyed to members of the operating crew?

a. The SM places the applicable information in the Daily Order Book, and issues an additional memo to all crew personnel that is initialed.

b. The SM is informed by memo of the addition to the Daily Order Book, and makes an announcement of the addition during the shift briefing.

c. The SOS places the applicable information in the Daily Order Book, and the individual operator is responsible for reviewing and initialing the Daily Order.

d. The SOS places the applicable information in the Daily Order Book, and makes an announcement of the addition during the shift briefing.

deleted

Answer c **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.1 Conduct of Operations

2.1.2 Knowledge of operator responsibilities during all modes of plant operation. 3.0 4.0

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
DAILY ORDER BOOK	BAP 350-2	C.3.h	1	7	
ADMINISTRATIVE PROCEDURES LESSON Selected Administrative Procedures	Selected Administrative Procedures	VII.B.1.h, 4	54	3	25

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type Comment

Question 4 US responsibility on CNMT entry

The following conditions exist on Unit 1:

- Reactor power 75%
- Incore neutron detectors are located at the bottom of the reactor and controls are deenergized
- An electrical penetration connections inside containment is to be checked due to abnormal readings taken on cables from outside containment
- The Personnel Hatch inner door seal is leaking by
- FIVE individuals have been selected for containment entry

What is the proper applicability for making a containment entry to check the penetration and repair the door under the above conditions?

Containment entry...

a. CANNOT be made in MODE 1.

b. CANNOT be made above 40% power.

c. can be made ONLY if the incore detectors are taken to their storage location.

d. can be made but ONLY to those areas outside the missile barrier.

Answer d **Exam Level** S **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.1 Conduct of Operations

2.1.13 Knowledge of facility requirements for controlling vital / controlled access. 2.0 2.9

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
ACCESS TO CONTAINMENT	BAP 1450-1	C. NOTE & C.2.b.4)	1 & 7	18	
ADMINISTRATIVE PROCEDURES LESSON Selected Administrative Procedures	Selected Administrative Procedures	XXI.B.1 & 8	86-88	3	50

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 5 Procedure required usage

An example of a licensed operator evolution that can be performed without having a procedure in hand is...

a. Adjusting rod position following a boration for delta-I control.

b. Starting the 1A Heater Drain Pump.

c. Placing excess letdown in service.

d. Latching and rolling up the main turbine following surveillance trip test.

Answer a **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.1 Conduct of Operations

2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation. 3.9 4.0

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Use Of Procedures For Operating Department	BAP 340-1	C.1.a.1), C.1.d	3, 6-7	9	
ADMINISTRATIVE PROCEDURES LESSON BAP 300-1 and 340-1	BAP 300-1 and 340-1	II.C.1.e	62	4	3

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 6 Use of electrical prints

Assuming an auto-close signal is continuously present in the circuit for the 1A SI pump, which contact will be maintained open in order to prevent the starting relay (SR) from attempting repeated breaker closures onto a faulted bus?

(E 1-4030-SI01 is provided for use.)

a. LC SW

b. 52/b

c. Y

d. LS

Answer c **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.1 Conduct of Operations

2.1.24 Ability to obtain and interpret station electrical and mechanical drawings. 2.8 3.1

Explanation of Answer "Y" is an antipump relay that when prevented from energizing interrupts the circuit that energizes the START relay in the AUTO start circuit

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Schematic Diagram - Safety Injection Pump 1A 1SI01P	6E 1-4030-SI01				
Print Reading	Chap 3		34		2.c

Material Required for Examination E 1-4030-SI01

Question Source: Facility Exam Bank **Question Modification Method:** Editorially Modified

uestion Source Comments: Braidwood requal bank

Comment Type **Comment**

Question 7 Equipment operability during surveillance

The following conditions exist on Unit 1:

- RCS temperature - 225°F
- RCS pressure - 500 psig
- RCPs operating - 1C & 1D

Procedure 1BOS PL- R1, REMOTE SHUTDOWN PANEL CONTROL POWER CHECK, is being performed with operators stationed at the control room (NSO), Remote Shutdown Panel (RSP) (NSO), and the MCR HVAC Train B RSP (NSO). Charging Pump 1A control switch on the main control board has been placed in PULL TO LOCK. The switch at the RSP is in REMOTE.

What is the status of the Charging Pump?

Charging Pump 1A is...

a. OPERABLE and NO ACTION per any LCO is required.

b. OPERABLE, with the operator stationed at the Remote Shutdown transfer switch in communication with the control room.

c. INOPERABLE and the ACTION associated with the LCO must be implemented.

d. INOPERABLE, but the Charging Pump is NOT required per any LCO for current plant conditions.

Answer C **Exam Level** S **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

.2 Equipment Control

2.2.12 Knowledge of surveillance procedures. 3.0 3.4

Explanation of Answer With control still active in the control room (REMOTE), placing the control switch in PTL prevents the auto actions for the pump.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
CONDUCT OF OPERATIONS	BAP 300-1	C.1.g.5)	11	14	
ADMINISTRATIVE PROCEDURES LESSON BAP 300-1 and 340-1	BAP 300-1 and 340-1	I.C.1.g.5)	28	4	12

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 8 MOV tagout

An operator is preparing an OOS that designates 1CC685, RCP Thermal Barrier CC Return CNMT Isolation valve, as an isolation point.

Vhat is the acceptability of using this isolation point?

The OOS is...

a. acceptable if the MOV is tagged at its control switch, power supply and valve handwheel.

b. acceptable if the MOV is tagged at its control switch, power supply and a blocking device is placed on the valve.

c. NOT acceptable because the MOV fails to meet isolation requirements.

d. NOT acceptable because the valve fails open on a loss of power.

Answer a **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.2 Equipment Control

2.2.13 Knowledge of tagging and clearance procedures. 3.6 3.8

Explanation of Answer Valve is MOV and requirements include tagging control switch, electrical power supply and local handwheel if accessible.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
OUT OF SERVICE PROCESS	BAP 330-1	C.4.c NOTE	27	28	
Selected Administrative Procedures - BAP 330-1	Selected Administrative Procedures	I.E.8	26	3	2

aterial Required for Examination

Question Source: New **Question Modificaticn Method:**

Question Source Comments:

Comment Type **Comment**

Question 9 Technical Specification 3.0.3 application

The following conditions exist on Unit 1:

- RCS Tave - 557°F
- RCS pressure -2230 psig
- MSIVs - closed
- MSIV Bypass valves open - 100%
- The breaker for MOV-CS007B, PP 1B HDR ISOL, tripped on overload
- RCFC SX inlet isolation valve 1SX016B is closed and tagged on Out of Service
- The crew is directed to perform a required Test that places the 1A RCFC Low Speed handswitch in PULL-TO-LOCK

What condition/action is required when the RCFC Low Speed handswitch is placed in PULL-TO-LOCK?

a. ONLY the ACTIONs for the inoperable CS train and the inoperable B train of RCFCs are applicable, and each must be restored within 7 days of its outage time.

b. The individual ACTION for each component is applicable, so the CS train must be restored within 7 days of its outage time and each train of RCFCs must be restored within 7days of their outage times.

c. The CONDITION results in combination of THREE inoperable trains, so the applicable ACTION is to be in MODE 5 within 84 hours.

d. The CONDITION results in combination of THREE inoperable trains, so a license event report (LER) is required due to voluntary entry into LCO 3.0.3 condition.

nswer d **Exam Level** S **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.2 Equipment Control

2.2.22 Knowledge of limiting conditions for operations and safety limits. 3.4 4.1

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Limiting Condition For Operation (LCO) Applicability & Containment Systems - Containment Spray (CS) and Reactor Containment Fan Cooler (RCFC)	ITS	3.0.3 & 3.6.6	3.0-1 & 3.6-17	A	
Intro to Technical Specifications	Chp 3	II.C.5.c	17	2	4, 5
ADMINISTRATIVE PROCEDURES LESSON BAP 300-1 and 340-1	BAP 300-1 and 340-1	I.C.1.g.5)	2813	4	32

Material Required for Examination

Question Source: New **Question Modification Method:** -

Question Source Comments:

Comment Type **Comment**

Question 10 Timing for Tech Spec required Shutdown

Unit 2 is at 100% reactor power. Diesel Generator 2B failed its surveillance test when one of the cylinder heads failed at the fuel injector connector. The following timeline tracks the outage time for D/G 2B:

ug 25 1030 - D/G started for one-hour run
1100 - D/G tripped due to failure. LCOAR time initiated.
Aug 27 1450 - Spare fuel injector found
Aug 28 0200 - Post maintenance test of D/G failed when time to rated frequency and voltage was 15 sec. Inspection revealed a crack in the governor casing. No replacement is available for 24 hours.

Per Operations Policy NOD-OP.19, when should the Unit 2 shutdown commence due to the required?

a. August 28 0200.

b. August 28 0300.

c. August 28 0630.

d. August 28 1100.

Answer ~~a~~ b **Exam Level** S **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98
Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC
2.2 Equipment Control
2.2.23 Ability to track limiting conditions for operations. 2.6 3.8

Explanation of Answer Operations Policy directs that if it is determined that repairs to a required OPERABLE piece of eqpt CANNOT be completed in the REQUIRED ACTION timeframe that the Unit shutdown be initiated as soon as this is discovered.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Electrical Power Sources - AC Sources - Operating	ITS	3.8.1 ACTIONS B.4	3.8-2	A	
Operation's Policy - Operations Management Of Technical Specifications Requiring Unit Shutdowns	NOP-OD.19	2.4	3 **Number(s)**	3	n
ADMINISTRATIVE PROCEDURES LESSON BAP 2010-2, Reactivity Management	BAP 2010-2, Reactivity Management	II.F.3.b	38	1	3

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 11 RCS level discrepancy during refueling

The following conditions exist for Unit 1 in preparation for head removal:

- Unit shutdown and cooldown initiated 120 hours ago
- Lowering of RCS level to the reactor vessel flange is underway
- RCS temperature - 95°
- RCS level Control Room indicators: 1LI-RY046 - 401' 0"
  1LI-RY049 - 402' 1"
- RH loop 1A in operation with "normal" indications

What is the appropriate action for these conditions?

a. The lowering of RCS level can continue after verifying appropriate amount of water removed.

b. The level change must be stopped until the cause for the level discrepancy is determined.

c. The running RHR pump shall be immediately stopped to prevent cavitation.

d. The available SI Pump is immediately aligned for hot leg injection and shall be started.

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.2 Equipment Control

2.2.26 Knowledge of refueling administrative requirements. 2.5 3.7

Explanation of Answer With any level discrepancy, the reason for the discrepancy must be determined before further draining can continue.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
REACTOR COOLANT SYSTEM DRAIN	BOP RC-4a	Step 20 CAUTION	11	14	
Residual Heat Removal System	Chp 18	III.C.1.c & III.C.2.C	44 & 48	2	9.c

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: Zion exam bank

Comment Type **Comment**

NRC Significant Industry Event -

Question 12 RO duties in Control Room during refueling

What is a responsibility of the NSO during refueling operations in the main control room?

a. Checking source range counts while a fuel assembly is being placed in the core.

b. Verifying direct phone communication with the Fuel Handling Supervisor once per day during fuel movement.

c. Maintaining a 1/M plot while reloading fuel during a core shuffle.

d. Updating the Control Room tag board per the Nuclear Component Transfer List on an hourly basis.

Answer a **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.2 Equipment Control

2.2.32 Knowledge of RO duties in the control room during fuel handling such as alarms from fuel handling area, communication with fuel storage facility, systems operated from the control room in support of fueling operations, and supporting instrumentation. 3.5 3.3

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
ADMINISTRATIVE CONTROL DURING REFUELING	BAP 370-3	C.1.c	4	18	
ADMINISTRATIVE PROCEDURES LESSON BAP 2010-2	BAP 2010-2, Reactivity Management	II.F.2	36	1	1.d
ADMINISTRATIVE PROCEDURES LESSON BAP 370-3	Administrative Control During Refueling, BAP 370-3	C.1.c.2	8	2	13

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 13 Radiation exposure determination

An operator has the following exposure history this year until today:

Deep Dose Equivalent (DDE)	-	210 mrem
Committed Effective Dose Equivalent (CEDE)	-	45 mrem
Shallow Dose Equivalent (SDE)	-	33 mrem
Committed Dose Equivalent (CDE)	-	28 mrem

Today the operator was required to make two entries into containment:

Entry 1: Gamma dose - 52 mrem; Neutron dose - 24 mrem
Entry 2: Gamma dose - 124 mrem

How much radiation exposure is available to the operator if he has to make additional entries?

His available margin based on the routine Administrative Exposure Control Levels is...

a. 100 mrem for that day; 2484 mrem for the year.

b. 100 mrem for that day; 2545 mrem for the year.

c. 124 mrem for that day; 2569 mrem for the year.

d. 124 mrem for that day; 2614 mrem for the year.

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.3 Radiation Control

2.3.1 Knowledge of 10 CFR: 20 and related facility radiation control requirements. 2.6 3.0

Explanation of Answer Limits are 300 mrem routine DDE/Day and 3000 mrem routine cumulative TEDE/year. C. Neutron rad not counted for daily & yearly; A. All counted for yearly; d. previous DDE+CEDE only counted for year.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
EXPOSURE REVIEW AND AUTHORIZATION	BRP 5300-2	F.1.a & F.5.a	4 & 9	6	
Radiation Protection	Chapter 3			1	4

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 14 Performance of Status Trees/Function Restoration

The following conditions exist on Unit 1:

- A reactor trip has occurred and both reactor trip breakers are verified open
- The turbine has tripped
- BEP-0 "Reactor Trip OR Safety Injection" has been entered.
- BUS 141 ALIVE light is NOT lit with bus voltage at ZERO volts
- BUS 142 ALIVE light is lit with bus voltage at 4149 volts.

Which of the following describes the action(s) the operators is/are required to take?

a. Check SI status.

b. Turn on the synchroscope and manually close ACB 1412, SAT 142-1 feed breaker.

c. Manually start 1A D/G and verify ACB 1413, D/G output breaker, closes.

d. Initiate actions of BOA ELEC-3 and then check SI status.

Answer d **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.4 Emergency Procedures / Plan

2.4.16 Knowledge of EOP implementation hierarchy and coordination with other support procedures. 3.0 4.0

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Reactor Trip Or Safety Injection	BEP-0	Step 3 RNO(Step 1-4 IMMEDIATE ACTION)	4	29	
Use Of Procedures For Operating Department	BAP 340-1	C.4.a	12	9	
ADMINISTRATIVE PROCEDURES LESSON BAP 300-1 and 340-1	BAP 300-1 and 340-1	II.C.4.a, b & f	78, 80, 84	4	3

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 15 Applicability of EOP Foldout Page

Following transition to BEP-1 "Loss of Reactor Or Secondary Coolant", the US refers to the Operator Action Summary, and directs the operator to Cold Leg Recirculation Switchover Criterion. Which of the following describes the complete set of procedures for which the Transfer to Cold Leg Recirculation equirements are applicable?

(NOTE: The following procedures are in the E-1 or CA-1 series:
BEP-1 "Loss Of Reactor Or Secondary Coolant"
BEP ES-1.1 "SI Termination"
BEP ES-1.2 "Post-LOCA Cooldown And Depressurization"
BEP ES-1.3 "Transfer To Cold Leg Recirculation"
BEP ES-1.4 "Transfer To Hot Leg Recirculation"
BCA-1.1 "Loss Of Emergency Coolant Recirculation"
BCA-1.2 "LOCA Outside Containment)

a. BEP-1, BCA-1.1 and BCA-1.2 procedures.

b. BEP-1, BEP ES-1.1 and ES-1.2 procedures.

c. BEP-1 and BEP ES-1.2 procedures.

d. BEP-1 procedure.

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.4 Emergency Procedures / Plan

2.4.20 Knowledge of operational implications of EOP warnings, cautions, and notes. 3.3 4.0

Explanation of nswer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Loss Of Reactor Or Secondary Coolant	BEP-1	OAS	3	19	
USE OF EMERGENCY OPERATING PROCEDURES	BAP 340-1	G.4	9	5	
ADMINISTRATIVE PROCEDURES LESSON BAP 300-1 and 340-1	BAP 300-1 and 340-1	II.C.3.d	78	4	7

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 16 Hazmat Spill Response

The following conditions exist on Unit 1:

- Reactor power is 4%
- Condensate/Feedwater oxygen levels are elevated
- Chemistry is in process of adding Hydrazine to the condensate system

A report is made to the Control Room by the Turbine Building NLO reporting a chemist had tripped in the turbine building, spilling 20 ounces of Hydrazine. The chemist was not seriously injured and did not come in contact with the liquid. The spill is isolated to the building. (Both trackway doors are closed.)

What are the reporting requirements for this spill?

a. Offsite reporting is NOT required because the spill was contained within the Turbine Building.

b. Offsite reporting is NOT required because the amount spilled is below Reportable Quantity.

c. Reporting is required to Environmental Services, the National Response Center, and IESDA because the amount spilled exceeds the Reportable Quantity.

d. Reporting is required to Environmental Services, the National Response Center, IESDA, and the NRC because the plant is in an at-power condition.

Answer a **Exam Level** S **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.4 Emergency Procedures / Plan

.4.29 Knowledge of the emergency plan. 2.6 4.0

Explanation of Answer Spill quantity is > that required to be reported. However, reporting is NOT required because the spill was NOT outside to the ground (fully contained within the building enclosure).

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
REPORTING HAZARDOUS MATERIAL INCIDENTS	BAP 3000-16	F.1.a	3	3	
ADMINISTRATIVE PROCEDURES LESSON Administrative Procedure, BAP 1250-6	Administrative Procedure, BAP 1250-6	II.A.6 & III.A.4	12 & 16	3	1

Material Required for Examination BAP 3000-16

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 17 Identification of inoperable CR annunciators

The following conditions exist on Unit 1:

- Reactor trip breakers status - OPEN
- RCS Tave - 557°F
- Pzr pressure - 2235 psig

Annunciator RCFC VIBRATION HI (1-3-C5) has been in alarm for the past hour due to vibration condition while maintenance troubleshoots the vibration probe on RCFC 1C.

Which of the following actions is appropriate for this alarm window?

a. The alarm should be acknowledged for each actuation and the SER monitored for valid alarm inputs.

b. The alarm should be acknowledged for each actuation and operators stationed locally at each RCFC to monitor vibration.

c. The alarm should have been silenced without acknowledgement with US permission and the SER monitored for valid alarm inputs.

d. The alarm should have been silenced without acknowledgement with US permission and operators stationed locally at each RCFC to monitor vibration.

Answer C or A **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.4 Emergency Procedures / Plan

2.4.31 Knowledge of annunciators alarms and indications, and use of the response instructions. 3.3 3.4

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
RCFC VIBRATION HI	BAR 1-3-C5	E.	1	51	
HANDLING OF MAIN CONTROL BOARD AND RADWASTE PANEL ANNUNCIATOR ALARMS	BAP 380-2	C.5	3	3	
Selected Administrative Procedures - BAP 300-2	Selected Administrative Procedures	XII.C.5	64	3	32

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 18 Effect of Xenon Transient & compensation

A feed pump trip occurred resulting in a rapid power reduction on Unit 1. Power was reduced from 100% steady-state conditions using a combination of rods and boration.

The following conditions exist for Unit 1 following stabilization:

- Reactor Power - 60%
- Delta-I target value - +2.0
- Control Bank D position - 160 steps withdrawn
- Tave - 572°F
- Delta-I - -10.5%
- Core Age - MOL

What actions will be required to maintain the current power level and maintain Delta-I within its normal operating band over the next FIVE hours?

a. Boration and control rod withdrawal, followed by dilution.

b. Boration and control rod insertion, followed by dilution.

c. Dilution and control rod withdrawal, followed by boration

d. Dilution and control rod insertion, followed by boration.

Answer a **Exam Level** B **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

001 Control Rod Drive System

A2. Ability to (a) predict the impacts of the following on the Control Rod Drive System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:

A2.06 Effects of transient xenon on reactivity 3.4 3.7

Explanation of Answer With delta-I near the negative limit of the band, boration would be initiated to allow rod withdrawal and hence shifting of power production toward positive delta-I (power shift toward top of core). Later as Xenon (neutron poison) builds in, dilution will be initiated to maintain power level

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
DELTA I CONSIDERATIONS AND GENERIC COASTDOWN GUIDELINES	BGP 100-8	E.4 & 5; F.3	2; 5-6	5	
BGP 100-8, I Considerations	BGP 100-8, I Considerations	II.B.5 & III.A.3	6 & 12-13	1	1
Curve Book - BYRON UNIT ONE CYCLE NINE Xenon Worth vs Time After Shutdown	BCB-1	Figure 8.c		19	

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type Comment

Question 19 Application of DC Hold

A problem with the rod control system requires checking several rod bank circuits. The affected power cabinet repairs are to be made by supplying power from the DC [illegible] supply cabinet.

Which statement describes the proper operation for DC Hold and the associated response in the event of a reactor trip?

a. ONE control rod bank group can be placed on DC HOLD, and these rods will drop if the controls are taken to OFF at the DC Hold cabinet.

b. ONE control rod bank group and ONE shutdown bank group can be placed on DC HOLD, and these rods will drop if the controls are taken to OFF at the DC Hold cabinet.

c. ONE control rod bank group can be placed on DC HOLD, and these rods will automatically drop.

d. ONE control rod bank group and ONE shutdown bank group can be placed on DC HOLD, and these rods will automatically drop.

Answer c **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

001 Control Rod Drive System

K1. Knowledge of the physical connections and/or cause-effect relationships between Control Rod Drive System and the following:

K1.03 CRDM 3.4 3.6

Explanation of Answer Only one GROUP of control rods can be placed on HOLD at a time in order to ensure the rods are held without falling. Opening the reactor trip breakers interrupts the power to the DC Hold cabinet.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
ROD DRIVE PLACEMENT IN D.C. HOLD	BOP RD-6	NOTE step F.4	1	1	
Rod Control System	Chapter 28	I.C.4 & II.A.6.a	8 & 44	1	9

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 20 Relationship of levels during refueling operations

The following conditions exist for Unit 2:

- Mode 5
- RCS is draining to Pzr level of 40%
- IM calibrations have been completed for LT-048, Refuel Cavity level, in preparation for further draining
- LI-462 indicates 40%

What is the relationship of Pzr level instrument LI-459 as compared to LI-048?

a. LI-459 and LI-048 will be offscale high.

b. LI-048 will be just onscale and LI-459 will be offscale low.

c. LI-459 will read higher than LI-462 and LI-048 will just be onscale

d. LI-048 will be offscale high and LI-459 will read lower than LI-462.

Answer C **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

002 Reactor Coolant System

A1. Ability to predict and/or monitor changes in parameters associated with operating the Reactor Coolant System controls including:

A1.11 Relative level indications in the RWST, the refueling cavity, the PZR and the reactor vessel during preparation for refueling 2.7 3.2

Explanation of Answer LI-462 is the cold calibrated Pzr level instrument and will read lower (but more accurately) than the hot calibrated level instruments (LI-459/460/461) at lower RCS temperatures. The refueling cavity level instrument just comes onscale at 40% Pzr level.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
REACTOR COOLANT SYSTEM DRAIN OR MAINTENANCE	BOP RC-4a	E.4	4	14	
Chp 14, Pressurizer (RY)	Chp 14	II.B.1.a.5).b).(2) & III.B.4.b, c	37, 74-76	3	19, 25
BGP 100-6, Refueling Outage	BGP 100-6	II.A.7	24-26	2	5

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 21 Conditions for loops operable/in operation

The following Unit 1 conditions exist:

- RCS temperature (Average CETC) - 340°F
- All S/G pressures - 100 psig
- RCS pressure - 435 psig
- RCP B - running
- RCP D - breaker tripped on overcurrent
- RCP A & C - Supply breakers tagged OOS
- RHR loops A and B aligned for ECCS

What is/are the required action(s) under these conditions?

a. The Unit must be placed in MODE 5 with either RHR Train in operation within 6 hours.

b. Both RHR Trains must be aligned for RCS cooldown and placed in service.

c. RCP B may be stopped for up to ONE hour to investigate the cause of the other RCP trip provided RCS temperature does NOT exceed 445°F.

d. Either the A or C RCP must be returned to service and made available for starting immediately.

Answer d **Exam Level** S **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

002 Reactor Coolant System

2.1 Conduct of Operations

2.1.10 Knowledge of conditions and limitations in the facility license. 2.7 3.9

Explanation of Answer Other answers plausible but NOT correct: 'a.' - only if ONE required loop operable is RHR & time frame incorrect; 'b.' - RHR cannot be placed in service with RCS pressure > 425 psig; 'c.' Action is allowed IAW NOTE to ITS 3.4.6 with limitation, but TWO loops required OPERABLE & MODE change to MODE 3 NOT allowed.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O. Number(s)
RCS Loops --MODE 4	ITS 3.4.6	ACTION B.1	3.4-11 & 12	A	
Chp 12, Reactor Coolant System	Chp 12	III.A.1	78	2	11.m

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 22 RCP and Pzr spray operations

The following Unit 1 conditions exist:

- RCS temperature (Average CETC) - 140°F
- RCS pressure - 365 psig
- A bubble has just been drawn in the Pressurizer
- All loops are filled and vented
- Preparations are in progress to start the first RCP for continuous run
- 1C RCP is started

What is the effect on RCS pressure control?

RCS pressure will increase and...

a. both Pzr Sprays will function normally for Pzr pressure control.

b. manual cycling of the Pzr heaters will be required for Pzr pressure control.

c. PORV RY456 will open on high pressure from high pressure bistable PB456E.

d. Pzr spray will deliver minimal spray flow for Pzr pressure control.

Answer d **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

003 Reactor Coolant Pump System

A1. Ability to predict and/or monitor changes in parameters associated with operating the Reactor Coolant Pump System controls including:

A1.06 PZR spray flow 2.9 3.1

Explanation of
nswer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
PLANT SHUTDOWN AND COOLDOWN	1BGP 100-5	F.55	31	27	
Pressurizer	RY-1	Schematic		2	
Chp 14, Pressurizer (RY)	Chp 14	II.A.4.c.1)		3	8.c, d

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 23 S/G temperature effect upon start of RCP

The following conditions exist on Unit 1:

- MODE 4 during plant heatup
- RCS temperature - 300°F
- RCS pressure - 400 psig
- Pzr level - 33%
- Preparations are underway for start of the first RCP, RCP 1D

What is the applicability of the temperature difference between S/G temperature and RCS loop temperature?

The requirement of having less than 50°F difference between S/G temperature and the associated RCS loop temperature...

a. is NOT applicable since this is the first RCP to be started.

b. ensures RCP seal parameters within normal operating range.

c. prevents overpressure event for RCS.

d. provides net positive suction head at suction of RCP.

Answer C **Exam Level** S **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

003 Reactor Coolant Pump System

K1. Knowledge of the physical connections and/or cause-effect relationships between Reactor Coolant Pump System and the following:

K1.10 RCS 3.0 3.2

xplanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
STARTUP OF A REACTOR COOLANT PUMP	BOP RC-1	Precaution 6	7	15	
Bases: RCS Loops-MODE 4	B 3.4.6(ITS Bases)	LCO	B 3.4-32	A	
Chp 13, Reactor Coolant Pump	Chp 13	III.A.2 & B.3	48 & 56	2	10, 11

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 24 Calculation of dilution

The following conditions exist on Unit 2:

- Unit is in MODE 5
- Unit burnup is 5700 EFPH in Cycle 7
- SDM - 1.3% DeltaK/K
- RCS pressure - 400 psig
- RCS average temperature - 195°F
- RCS boron concentration - 1006 ppm
- Differential boron worth - -10.75 pcm/ppm
- PZR level - 32.3%
- SR NIS countrate - 10 cps , BOTH channels are stable at "background levels"
- An inadvertent dilution at 70 gpm begins at 1300 hours

Assuming NO operator action is taken and PZR level remains constant over the time period, when would the HIGH FLUX AT SHUTDOWN alarm actuate?

a. NO action, because BDPS will actuate prior to receiving the annunciator.

b. 1430 hours.

c. 1505 hours.

d. 1734 hours.

Answer C **Exam Level** B **Cognitive Level** Application **Facility:** Byron **Exam** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

004 Chemical and Volume Control System

A4. Ability to manually operate and/or monitor in the control room:

A4.07 Boration/dilution 3.9 3.7

Explanation of Answer Dilution rate dc/dt = (500)(C)(Y)/M where M is the RCS mass at the given temperature (200°F). M = 745,537 lbm; C = 1006 ppm (given); Y=70 gpm (given). The dil rate = 47.2 ppm/hr. HIGH FLUX AT SHUTDOWN alarms at 5 x background = 50 cps. With K1= 0.987 dK/K (p1=-0.01317), calculate K2 = 0.9974 DKr/K (p2=-0.00261). Delta-P = 1056 pcm. 1056/-10.75=-98.2 ppm change required. Therefore the time required for the 98 ppm dilution is 98.2/47.2 = 2 hours 5 min. Difference in time based on use of Nomograph for RCS at normal pressure & temperature conditions. 'd' would only occur if count rate doubled in any 10 minute period. Assuming count rate increase is linear, for given dilution rate counts would change by 3 every 10 minutes.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Boron Dilution Rate Nomograph	Byron Curve Book, Unit 2	BCB-2 Figure 12		1	
Chp 15b, Reactor Makeup Control System	Chp 15b	III.B.3.d & e	57-59	2	7.c
Chp 31, Source Range Nuclear Instrumentation	Chp 31	II.B.2	42	1	10.a, 11.a

Material Required for Examination CURVE BOOK CBC-2 Figure 12. And GFES Equation Sheet

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 25 Requirements/Operation of PORVs at low RCS pressures

The following conditions exist on Unit 1:

- Unit is in MODE 5 during cooldown per 1BGP 100-5
- RCS has just been filled to solid plant condition
- RHR pump 1A operating in Shutdown Cooling mode
- RCS temperature - 150°F equilibrium
- RCS pressure - 335 psig

A failure of the letdown pressure control valve PCV-131 causes RCS pressure to rise to 454 psig, with RHR pump 1A deltaP measured at 120 psig.

Which of the following occurs to provide warning of or mitigate the consequences of this pressure rise?

a. The RHR loop suction relief valve will open, and the RHR suction valves from the RCS, 1RH8701A and B, will close.

b. Both Pzr PORVs will open, and the RHR loop suction relief valve and the RHR loop 1A discharge relief will open.

c. Pzr PORV 1RY456 will open, and the RHR suction valves from the RCS, 1RH8701 and 1RH8702, will close.

d. Pzr PORV 1RY455A will open, and the RHR loop suction relief valve will open.

Answer d **Exam Level** S **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 3 **SRO Group:** 3

)5 Residual Heat Removal System

A2. Ability to (a) predict the impacts of the following on the Residual Heat Removal System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:

A2.02 Pressure transient protection during cold shutdown 3.5 3.7

Explanation of Answer PORV 1RY455A setpoint for opening in LTOPS is 446 psig for given temperature. RHR suction relief valves are set to open at 450 psig. PORV 1RY456 setpoint is 462 psig. The RHR discharge reliefs are set to open at 600 psig.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Unit 1 (LTOPS) Low Temperature Overpressure Protection System		1BCB-1		Figure 29	3
Chp 14, Pressurizer (RY)	Chp 14	II.C.2	58	3	24
Chp 18, Residual Heat Removal System	Chp 18	II.A.5 & 9	24 & 32	2	4.e & f, 10

Material Required for Examination Curve Book BCB-1 Figure 29

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 26 **Topic** Recirc interties to SI Pumps & CV Pumps

The following conditions exist on Unit 1:

- A LOCA has occurred
- Actions of 1BEP ES-1.3, 'Transfer To Cold Leg Recirculation, have been completed.
- During alignment, 1CV8804A, RH HX to CENT CHG Pumps Isolation Valve, failed to open and could NOT be manually opened.

What is the status of the ECCS system?

a. The RHR discharge headers are cross-tied with only RHR Pump 1B running and supplying suction to the SI pumps and Centrifugal Charging pumps from the B train connection.

b. The RHR discharge headers are cross-tied with both RHR pumps running and supplying suction to the SI pumps only from the B train connection. The Centrifugal Charging pumps are stopped.

c. RHR Pump 1B is discharging through the B Train cold leg injection headers and supplying suction to the SI Pumps. RHR Pump 1A and the Centrifugal Charging pumps are stopped.

d. RHR Pump 1B is discharging through the B Train cold leg injection headers and supplying suction to the SI pumps and Centrifugal Charging pumps. RHR Pump 1A is discharging through the A Train cold leg injection headers.

Answer d **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 3 **SRO Group:** 3

005 Residual Heat Removal System

K1. Knowledge of the physical connections and/or cause-effect relationships between Residual Heat Removal System and the following:

1.12 Safeguard pumps 3.1 3.4

Explanation of Answer CL recirc lineup has any ONE running RHR pump aligned to provide suction path to all other ECCS pumps (SI & CENT CHG). The discharge headers between RH trains are required to be separate so that the ONE running RH pump does not operate in runout condition.

Reference Title	Facility Reference Number	Section	Page Number(s)	Revision	L. O.
Transfer To Cold Leg Recirculation	1BEP ES-1.3	steps 3, 5, 6	3-5	1A WOG-1 B	
Abnormal Operating Procedures - Loss of Reactor Or Secondary Coolant	BEP 1, BEP ES-1.1 - 1.4	BEP ES-1.3, 3-5	3-5	2	10
Chp 58, Emergency Core Cooling System	Chp 58	II.C.2, III.D.1.b	48-50, 74	2	8

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question ~~26~~ 27 Systems response to SI

The following conditions exist on Unit 1:

- A plant heatup is underway
- MODE 3 has just been entered
- RCS pressure 450 psig

SI Accumulator 1C was drained below required level during the outage for repair work. System configuration has NOT allowed refilling the Accumulator until now. The SI Accumulator line is being flushed in accordance with BOP SI-14 "SI Accumulator Fill Line Flush" (Valve lineup includes: 1SI-8964, SI Test Lines to Radwaste Isolation Valve, and SI-8888, SI Pps to Accumulator Fill Valve, are open. 1SI 8821A, SI Pump to Cold Leg Isolation Valve, and 1SI 8802A, SI to Hot Leg 1A & 1D Isol valve are closed). SI pump 1A running. During the flushing, an inadvertent SI signal is generated.

What is the status of the ECCS based on the current alignment without operator action?

a. 1B SI pump injection flow is directed to the RCS cold legs and 1A SI pump flow is directed to the Accumulator 1C fill line flush.

b. 1A SI pump flow is directed to the 1C Accumulator fill line flush and 1B SI Pumps is in PULL-TO-LOCK.

c. BOTH SI pump flows are directed to the RCS cold legs and to the Accumulator 1C fill line flush.

d. BOTH SI pump flows are directed to the RCS cold legs ONLY.

Answer a **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98
Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

06 Emergency Core Cooling System

A2. Ability to (a) predict the impacts of the following on the Emergency Core Cooling System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:

A2.13 Inadvertent SIS actuation 3.9 4.2

Explanation of Answer SI pumps are operable; SI8821A remains closed; SI8888 and SI8964 remain open.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
PLANT HEATUP	1BGP 100-1	F.49	38	29	
SI ACCUMULATOR FILL LINE FLUSH	BOP SI-14	F	2-3	4	
Chp 58, Emergency Core Cooling System	Chp 58	III.C.8	68	2	6.d & 9.b

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 28 10CFR50.46 Design Criteria

To meet the 10CFR50.46 criteria, the ECCS System is designed such that under accident conditions it will maintain...

a. total hydrogen production from zirconium-water reaction below maximum value of 5%.

b. maximum fuel temperature at the inside surface of the cladding NOT to exceed 2000°F.

c. the core at least 5% dK/K shutdown to prevent an inadvertent return to criticality.

d. fuel clad oxidation less than 17% of total clad thickness anywhere within the core.

Answer d **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/96

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

006 Emergency Core Cooling System

K3. Knowledge of the effect that a loss or malfunction of the Emergency Core Cooling System will have on the following:

K3.02 Fuel 4.3 4.4

Explanation of Answer Third selection addresses design criteria for reactivity control per ITS.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
	10CFR50	47			
Chp 61,Engineered Safety Features	Chp 61	I.C.3	10	2	3
Chp 58, Emergency Core Cooling System	Chp 58	I.D.1	10	2	2

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments:

Comment Type **Comment**

Question 29 Evaluation of flow ECCS pumps

The following conditions exist on Unit 1:

- A LOCA has occurred
- 1B SI pump trips and cannot be restarted
- Transfer to Cold Leg recirculation is required
- RCS pressure is approximately 50 psig

What is the approximate total SI pump flow indicated on the main control board and how will this value change following transfer of BOTH trains of ECCS to cold leg recirculation?

	Total Flow	Flow Change
a.	400 gpm	Decrease
b.	650 gpm	Increase
c.	800 gpm	Decrease
d.	1300 gpm	Increase

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

006 Emergency Core Cooling System

K6. Knowledge of the of the effect of a loss or malfunction on the following will have on the Emergency Core Cooling System:

K6.03 Safety Injection Pumps 3.6 3.9

Explanation of Answer SI pump design values provide for 400 gpm flow per pump @ 1200 psig and 650 gpm @ 800 psig (or less). The flow from the pumps increases since the RH pumps are now providing a suction pressure of approximately 250 psig to the pumps instead of the lower pressure (30 psig or less) provided by the head associated with RWST level.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Chp 58, Emergency Core Cooling System	Chp 58	II.A.3.c & 5.c & d	22, 31	3	3, 8.a

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 30 Spray using Normal and Aux Spray

What are the parameters and values used by the operator to ensure the temperature difference between the PZR and the spray fluid are within the specified limit(s) in the PRESSURE AND TEMPERATURE LIMIT REPORT when initiating PZR spray?

a. For normal spray, the difference between RCS hot leg loop temperature and PZR vapor space temperature limit is 50°F, and for aux spray, the difference between Regenerative Hx charging inlet temperature and PZR vapor space limit is 320°F.

b. For normal spray, the difference between RCS cold leg loop temperature and PZR vapor space temperature limit is 50°F, and for aux spray, the difference between Regenerative Hx charging outlet temperature and PZR vapor space limit is 320°F.

c. For normal spray, the difference between RCS hot leg loop temperature and PZR vapor space temperature limit is 320°F, and for aux spray, the difference between Regenerative Hx charging inlet temperature and PZR vapor space limit is 320°F.

d. For normal spray, the difference between RCS cold leg loop temperature and PZR vapor space temperature limit is 320°F, and for aux spray, the difference between Regenerative Hx charging outlet temperature and PZR vapor space limit is 320°F.

Answer d **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

010 Pressurizer Pressure Control System

A1. Ability to predict and/or monitor changes in parameters associated with operating the Pressurizer Pressure Control System controls including:

A1.08 Spray nozzle DT 3.2 3.3

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
PLANT HEATUP	1BGP 100-1	E.3.d	11	29	
PRESSURIZER TEMPERATURE LIMIT SURVEILLANCE PRESSURIZER SPRAY WATER TEMPERATURE DIFFERENTIAL LIMIT SURVEILLANCE	1BOS 4.9.2-1 / 1BOS 4.9.2-2	7-10, 2-5	3, 2	3 / 1	
Chp 14, Pressurizer (RY)	Chp 14	III.A.6	70	3	26.b

Material Required for Examination

Question Source: New **Question Modification Method:** Significantly Modified

Question Source Comments: Kewaunee 2/94 NRC Exam

Comment Type **Comment**

Question 31 DNB Limits

For Unit 1, a power increase is underway at the maximum allowed continuous rate within the fuel preconditioning limits following a refueling outage. The current conditions:

- Reactor power - 40%
- RCS Tave - 567°F
- Pzr pressure - 2175 psig
- PZR level - 33%
- CVCS letdown isolated
- Excess letdown in service

How are RCS DNB limits to be addressed under these conditions?

a. Pzr pressure must be raised to at least 2219 psig within 2 hours.

b. Pzr level must be restored to within 5% of programmed level within the next 2 hours.

c. NO action is required since the Pzr pressure limit is NOT applicable with the current power increase underway.

d. NO action is required since RCS temperature limits are NOT exceeded.

Answer a **Exam Level** S **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

010 Pressurizer Pressure Control System

2.1 Conduct Of Operations

2.1.12 Ability to apply technical specifications for a system. 2.9 4.0

Explanation of Answer Pzr level is NOT DNB parameter, but is out of normal band. Pzr pressure limit is NOT applicable only if it resulted from conditions involving a ramp of >5%/minute. Precondition limits power rise above 25% RTP to 3%/hour max.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits	Byron Unit 1&2 ITS	3.4.1	3.4-1	A	
POWER ASCENSION	1BGP 100-3	E.1.e.2)	8	25	
Chp 12, Reactor Coolant System	Chp 12	III.A.1	78	2	11.h

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 32 Evaluation of Pzr conditions

The following conditions exist on Unit 1:

- A load reject from 100% power has occurred
- Reactor power - 80%
- Pzr level - 56%
- Pzr vapor temperature - 655°F
- Pzr liquid temperature - 653°F
- RCS Tave - 578°F

What is the current status of the Pressurizer based on given conditions?

a. Backup and proportional heaters are fully on.

b. Proportional heaters are modulated on.

c. Pzr spray valves have modulated open.

d. Pzr spray valves and Pzr PORVs are open.

Answer C **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

010 Pressurizer Pressure Control System

K5. Knowledge of the operational implications of the following concepts as they apply to the Pressurizer Pressure Control System:

K5.01 Determination of condition of fluid in PZR, using steam tables 3.5 4.0

Explanation of Answer At 655°F, saturation pressure is 2272 psig. At this pressure, with current PZR level deviation <5% of program level(53%), the sprays are the only component "on".

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
zr Pressure Control	RY-2	Pzr Pressure Setpoints		3	
Chp 14, Pressurizer (RY)	Chp 14	II.C.1.c.4)	56	3	7 & 8
Steam tables		Saturation Table			

Material Required for Examination Steam Tables

Question Source: Facility Exam Bank **Question Modification Method:** Concept Used

Question Source Comments: Braidwood 1997 NRC exam

Comment Type **Comment**

Question 33 Pzr Level Reactor Trip

The following conditions exist on Unit 1 with all controls in normal lineup:

- Reactor power - 30% stable
- RCS Tave - 564.5°F
- Pzr pressure - 2230 psig
- Pzr level - 36% (LI-459), 37% (LI-460), 36% (LI-461)
- Pzr LVL CONT CH SELECT is in 459/460 position

The pressurizer level controller 1LK-459 output fails low. What automatic actions will occur as a result of this failure assuming NO operator action taken?

a. Pzr level will NOT change due to LT-460 being the controlling channel

b. The reactor will trip on high Pzr level due to letdown isolation.

c. Pzr level will control at 25% due to low output from the controller.

d. Pzr level will control at 60% due to low output from the controller.

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

011 Pressurizer Level Control System

K1. Knowledge of the physical connections and/or cause-effect relationships between Pressurizer Level Control System and the following:

K1.04 RPS 3.8 3.9

Explanation of Answer NOTE that this failure is like the failure of the controlling level channel high in that charging flow falls to minimum. At 17% level, letdown isolates charging continues at minimum (52 gpm) and Pzr level rises to high level trip setpoint.).

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Pzr Level Control	RY-3	Schematic, Pressurizer Level Setpoints		2	
Chp 14, Pressurizer (RY)	Chp 14	III.C.3.g	86-88	3	21

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments:

Comment Type **Comment**

Question 34 Input that can be bypass & condition

The following conditions exist on Unit 2:

- Unit shutdown is in progress
- Reactor power - 20%
- RCS Tave - 562°F
- Pzr pressure - 2235 psig
- Pzr level - 32%
- First stage turbine pressure channel PT-506 fails high

What affect does this failure have on operations as unit shutdown is continued, if NO action is taken for the channel failure?

a. At 10% power, the reactor will trip if the SR MAN BLOCK switches are taken to RESET.

b. At 9% power, the reactor will trip if an RCP trips.

c. At 7% power, the reactor will trip if the TURBINE TRIP pushbuttons are depressed.

d. At 5% power, the reactor will be manually tripped as required during a normal shutdown.

Answer d **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

012 Reactor Protection System

A4. Ability to manually operate and/or monitor in the control room:

A4.03 Channel blocks and bypasses 3.6 3.6

Explanation of Answer PT-506 failure results in P13 interlock NOT clearing when turbine power falls below 10%. This also feeds into P7 "AT POWER TRIPS" interlock also remains active. Trips affected: 1) 2 loop loss of flow, 2) Pzr low press, 3) Pzr high level, 4) RCP brkr open, 5) RCP UV, 6) RCP UF. At 10% power, the SR NIS should still be auto blocked by P-10 (active). The turbine is normally tripped from ~65 Mwe at 5% power per BGP.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
POWER DESCENSION	1BGP 100-4	NOTE at step F.27	15	16	
ESF Setpoints	EF-1	Permissive, Reactor Trip		4	
Chapter 60b/Reactor Protection System	Chapter 60b	II.C.3.d, II.C.8	36, 42	2	4

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 35 Basis for OTdT with input

The following Unit 1 conditions existed at the time of a reactor trip:

- LOOP	1	2	3	4
- Indicated NI power-	102%	107%	108%	110%
- Indicated OT Delta T-	107%	110%	107%	105%
- Indicated OP Delta T-	106%	109%	109%	108%
- Indicated loop Delta T-	105%	106%	107%	108%
- PZR pressure -	1885 psig	2500 psig	1920 psig	1910 psig
- PZR level -	10%	12%	8%	10%
- S/G levels -	34%	32%	25%	19%

- 1B Main Feedwater pump tripped just prior to the reactor trip
- No penalties for any trip setpoints existed at the time of the trip

What is the reason for the Reactor Trip System function that initiated this automatic reactor trip?

a. Protects the integrity of the RCS against overpressurization.

b. Maintains centerline temperature of fuel pellet below melting point.

c. Ensures that the design limit DNBR is met.

d. Provides for protection from a loss of heat sink.

Answer C **Exam Level** S **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

012 Reactor Protection System

4. Knowledge of Reactor Protection System design feature(s) and or interlock(s) which provide for the following:

4.02 Automatic reactor trip when RPS setpoints are exceeded for each RPS function; basis for each 3.9 4.3

Explanation of Answer TWO channels of OTdT are below their required trip setpoint. OTdT trips are provided for DNB protection.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
ESF Setpoints	EF-1	Reactor Trip - OTdT		4	
RTS Instrumentation	Byron Unit 1 & 2 ITS	BASES B.3.3.1, Applicable Safety Analysis, LCO, & Applicability, 6	B 3.3-21	A	
Chapter 60b/Reactor Protection System	Chapter 60b	I.B.3.a.4), c, II.B.3.b	4, 8,	2	3.c, 5.c

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 36 CNMT Spray/Phase B

A heatup is in progress on Unit 1.

At 0700, the following conditions are noted:

- RCS pressure - 1750 psig
- RCS temperature - 480°F
- S/G pressures - 565 psig

At 0730, the following conditions are noted:

- RCS pressure - 1850 psig
- RCS temperature - 485°F
- S/G pressures - 593 psig

If the current trend continues, the FIRST event that the operators should expect to see is the...

a. Pzr PORVs open

b. MSIVs close

c. Pzr sprays open.

d. S/G PORVs open

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

013 Engineered Safety Features Actuation System

4. Knowledge of Engineered Safety Features Actuation System design feature(s) and or interlock(s) which provide for the following:

K4.03 Main Steam Isolation System 3.9 4.4

Explanation of Answer RCS (Pzr) pressure rises above the P-11setpoint (1930 psig), which provides permissive for SI/Main Steam Line Isolation on low S/G pressure, and S/G pressure is less than MSL isolations.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
ESF Setpoints	EF-1	Permissive		3	
ESF Setpoints CS MCB Indications	EF-2	Steamline Isolation Signals		5	
Chp 61,Engineered Safety Features	Chp 61	II.C.17	41	2	7

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type Comment

Question 37 Conditions for MODE change - ESFAS function inop

The following conditions exist on Unit 2:

- Mode 5 with plant heatup in progress
- ONE channel of Pressurizer Pressure Low Safety Injection input logic is inoperable

In accordance with Improved Technical Specifications (ITS), what is the limitation on the plant heatup with this channel Out Of Service?

Entry can be made into...

a. MODE 4 but entry into MODE 3 is NOT allowed.

b. MODE 4 and MODE 3, but conditions must be maintained below the P-11 interlock.

c. MODE 4 and MODE 3, and the channel must be returned to service within 6 hours after entry into MODE 3.

d. MODE 4 and MODE 3 without limitations.

Answer b **Exam Level** S **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

013 Engineered Safety Features Actuation System

K6. Knowledge of the of the effect of a loss or malfunction on the following will have on the Engineered Safety Features Actuation System:

K6.03 Breakers, relays, and disconnects 2.4 2.9

Explanation of Answer Requires differentiation between SI requirements: SI function reqd MODES 1-4; Spec 3.0.4 does NOT allow MODE entry based on actions and P-11 limits for listed input.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
ESFAS Instrumentation	Byron Unit 1 & 2 ITS	3.3.2; Table 3.3.2-1 Function 1.d	3.3-32	A	
Chp 61,Engineered Safety Features	Chp 61	II.C.1.a, III.A.1	25, 52	2	5, 7.a

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 38 DRPI vs. Demand Position

A failure has occurred affecting the rod position indication that normally provides the more precise indication. Which of the following describes the action required for the failure of this indication?

a. Demand position requires verification of DRPI operability every 8 hours.

b. Demand position requires verification of DRPI operability within 24 hours.

c. DRPI requires the affected rod position(s) must be verified by use of incore detectors every 8 hours.

d. DRPI requires affected rod position(s) must be verified by use of incore detectors within 24 hours.

Answer a **Exam Level** S **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 1

014 Rod Position Indication System

K5. Knowledge of the operational implications of the following concepts as they apply to the Rod Position Indication System:

K5.02 RPIS independent of demand position 2.8 3.3

Explanation of Answer Demand position is more precise under normal conditions since it indicates rod position within ± 1 step. DRPI position provides a precision (or accuracy) of only ± 4 steps under normal conditions.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Rod Position Indication	Byron Units 1 & 2 ITS	3.1.7 CONDITION C	3.1-17	A	
Rod Position Indication Bases	Byron Units 1 & 2 ITS	B 3.1.7-Background	B3.1-52 - 53	A	
Chp 29, Rod Position Indication System	Chp 29	II.A.1.c, 4, III.A.1		2	2, 7

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 39 SR NIS discriminator failure

How would the failure of the pulse height discriminator to a low value affect the indication of the affected Source Range channel?

'he output would increase due to...

a. electronic filtering which narrows the pulse height window.

b. failure in removing the higher amplitude neutron generated pulses.

c. increased gamma interaction inside the detector.

d. counting of the gamma generated pulses and decay-alpha generated pulses.

Answer d **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

015 Nuclear Instrumentation System

A2. Ability to (a) predict the impacts of the following on the Nuclear Instrumentation System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:

A2.02 Faulty or erratic operation of detectors or compensating components 3.1 3.5

Explanation of Answer Pulse height discriminator used to set window to detect those pulses with energy level high enough to be from event associated with neutron detection. Gamma and other interactions such as the alpha decay of fission product daughters is of lower height (energy) and discriminator normally electronically removes.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Source Range Detector	NI-4	Schematic - Pulse Amp & Discriminator		4	
Chp 31, Source Range Nuclear Instrumentation	Chp 31	II.A.2.b	16-17	1	3

.laterial Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 40 SR NIS - loss of control power

The following conditions exist on Unit 1:

- RCS at NOT NOP
- Reactor trip breakers - closed
- Source Range readings:
  N31 - 18 cps
  N32 - 22 cps

What indication would the operator observe if Control Power was lost to the N31 Drawer?

The N31 meter would read...

a. downscale, the associated drawer bistable lamps NOT lit, and reactor trip breakers closed.

b. downscale, the associated drawer bistable lamps lit, and reactor trip breakers open.

c. 18 cps, the associated drawer bistable lamps NOT lit, and reactor trip breakers closed.

d. 18 cps, the associated drawer bistable lamps lit, and reactor trip breakers open.

Answer d **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

015 Nuclear Instrumentation System

K2. Knowledge of electrical power supplies to the following:

K2.01 NIS channels, components, and interconnections 3.3 3.7

Explanation of Answer Control power loss affects bistables which trip but NOT drawer instrument indication which is from Instrument Power source.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Source Range Detector	NI-4	Loss Of Control Power		4	
Chp 31, Source Range Nuclear Instrumentation	Chp 31	II.A.2.g.5), III.C.2	18, 70	1	8.b

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 41 Work on PR NIS affect on SR NIS

The following conditions exist on Unit 1:

- Mode 3, with reactor trip breakers closed
- Power Range Channel N44 OOS for calibration
- Source Range readings:
  N31 - 10 cps
  N32 - 14 cps

118 VAC power to the Channel II rack of NIS is lost when the supply breaker trips.

What action is required for this situation?

a. Verify the reactor trip breakers are open due to loss of all Source Range trip functions.

b. Suspend any positive reactivity additions due to the loss of Source Range N32.

c. Initiate an emergency boration due to the loss of BOTH Source Range Channels.

d. Restore at least ONE Source Range Channel to OPERABLE status and in ONE hour close boron dilution isolation valves.

Answer a **Exam Level** S **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

015 Nuclear Instrumentation System

K4. Knowledge of Nuclear Instrumentation System design feature(s) and or interlock(s) which provide for the following:

K4.05 Reactor trip 4.3 4.5

Explanation of Answer A loss of power affects N31(SR), N36 (IR) and N42 (PR). When N42 lost, P-10 interlock affected which deenergizes the SR instruments.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
RTS Instrumentation	Byron Units 1&2 ITS	3.3.1, Table 3.3.1-1 Function 5, CONDITION G	3.3-14, 3.3-4	A	
Chp 31, Source Range Nuclear Instrumentation	Chp 31	III.A.2	62	1	9.a
Chp 33, Power Range Nuclear Instrument	Chp 33	II.C.4.c, III.D.1	44, 60-61	1	5.e, 8

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type Comment

Question 42 NR RTD Failure effects

The following conditions exist on Unit 1:

- Reactor power - 50%
- RCS Tave - 570°F (A); 569°F (B); 569°F (C); 570°F (D)
- RCS Thot - 585°F (A); 584°F (B); 583°F (C); 585°F (D)
- RCS Tcold - 555°F (A) 554°F (B); 555°F (C); 555°F (D)
- Pzr pressure - 2235 psig
- Pzr level - 43 %

If loop B Thot output channel fails LOW, what is the response of Pzr level ?

Pressurizer level will...

a. increases to 60%.

b. remains the same.

c. decreases to 25%.

d. decreases to the letdown isolation setpoint.

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

016 Non-Nuclear Instrumentation System

K3. Knowledge of the effect that a loss or malfunction of the Non-Nuclear Instrumentation System will have on the following:

K3.02 PZR LCS 3.4 3.5

Explanation of Answer Thot fails to 510°F. With loop Tcold of 537°F, loop Tave is now 524°F. Auctioneered HIGH Tave is used for Pzr level program.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.	
Pzr Level Control	RY-3	Schematic - Level Program Controller		2		
Abnormal Operating Procedures, Operation with a Failed Instrument Channel		1BOA INST-2	I.B.2 - Th fails low	15	1	1.a, 4
Chp 12, Reactor Coolant System	Chp 12	II.B.2.f.3), c.3)	23-24, 29	2	6.a	

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Concept Used

Question Source Comments: Zion 2/92 NRC Exam (along with several others). Change includes failure of Thot loop, failure low and conditions instead of dual condition.

Comment Type **Comment**

Question 43 Sequence for securing CNMT Spray

The following conditions exist on Unit 1:

- A LOCA has occurred
- Transition has been made to BEP ES-1.3 "Transfer To Cold Leg Recirculation"
- Containment Spray actuated due to high containment pressure
- All systems and components operating as expected

What conditions allow for termination of Containment Spray?

a. ONE pump is stopped when containment pressure is less than 15 psig. The other pump is stopped when RWST LO-3 level is reached.

b. ONE pump is stopped when containment pressure is less than 20 psig. The other pump is stopped after it has operated for a period of at least TWO hours

c. BOTH pumps are stopped when containment pressure is less than 15 psig and have operated for a period of at least TWO hours.

d. BOTH pumps are stopped when containment pressure is less than 20 psig and RWST LO-3 level is reached.

Answer C **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 1

026 Containment Spray System

A2. Ability to (a) predict the impacts of the following on the Containment Spray System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:

A2.08 Safe securing of containment spray when it can be done) 3.2 3.7

Explanation of nswer

Reference Title	**Facility Reference Number**	**Section**	**Page**	**Revisio**	**L. O.**
Containment Spray	CS-1	CS Termination		3	
Loss Of Reactor Or Secondary Coolant	1BEP-1	Step 7.d	9	1A WOG-1 B	
Chp 59, Containment Spray System	Chp 59	III.E.4	54	2	12

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 94 CNMT Spray/Spray Additive operability requirements - Basis

What is the safety analysis basis for the minimum OPERABILITY requirements for the Spray Additive System?

The Design Basis Accident analyses assume that...

a. ONE train is OPERABLE and the volume of NaOH added to the spray in the time the RWST reaches the LO-2 setpoint is adequate to ensure a minimum 7.0 pH in the containment recirculation sump to reduce stress corrosion of mechanical components.

b. ONE train is OPERABLE and the volume of NaOH added to the spray in the time the RWST reaches the LO-3 setpoint is adequate to remove iodine from containment atmosphere and maintain it in solution in the recirculation sump.

c. TWO trains are OPERABLE and the volume of NaOH added to the spray in the time the RWST reaches the LO-3 setpoint is adequate to ensure a minimum 7.0 pH in the containment recirculation sump to reduce stress corrosion of mechanical components.

d. TWO trains are OPERABLE and the volume of NaOH added to the spray in the time the RWST reaches the LO-2 setpoint is adequate to remove iodine from containment atmosphere and maintain it in solution in the recirculation sump.

Answer b **Exam Level** S **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 1

026 Containment Spray System

2.1 Conduct Of Operations

2.1.10 Knowledge of conditions and limitations in the facility license. 2.7 3.9

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Containment Systems - Spray Additive System - Applicable Safety Analysis	Byron Unit 1 & 2 ITS Bases	B 3.6.7	B 3.6-44	A	
Chp 59, Containment Spray System	Chp 59	I.C.2 ,3, III.A	8, 48	2	3, 4

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 45 Safety Analysis on dilution during refueling

The following conditions exist:

- SFP level - 424 feet, 4 inches
- SFP boron concentration - 2000 ppm
- SFP temperature - 102°F

Normal unborated makeup is established to the SFP.

If makeup is NOT isolated, what is the effect on SHUTDOWN MARGIN (SDM) for the Spent Fuel Pit?

a. If the boron concentration were to drop to 500 ppm, the SDM would remain at least 2%.

b. If the boron concentration were to drop to 500 ppm, the SDM would remain at least 5%.

c. With a maximum dilution flow rate of 175 gpm, the operator has at least 4 hours before SDM is lost.

d. With a maximum dilution flow rate of 175 gpm, the operator has at least 50 minutes before SDM is lost.

Answer b **Exam Level** S **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

033 Spent Fuel Pool Cooling System

A2. Ability to (a) predict the impacts of the following on the Spent Fuel Pool Cooling System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:

A2.01 Inadequate SDM 3.0 3.5

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Bases - Plant Systems - Spent Fuel Assembly Storage - Background	Byron Unit 1 & 2 ITS	3.7.16	B 3.7-100 -101	A	
Chp 51, Spent Fuel Pool Cooling and Cleanup	Chp 51	I.C.3	4	3	7

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 46 Steam Dump input malfunction

The following conditions exist on Unit 1:

- Reactor power was 65% when the turbine tripped
- An ATWS occurred
- The reactor tripped 15 seconds later when B reactor trip breaker was locally opened
- Reactor trip breaker A is failed closed
- RCS Tave - 559°F
- Pzr pressure - 2255 psig
- Steamline header pressure - 1100 psig
- No controls other than control rods and boration controls have been operated

What is the status of the Steam Dump valves?

Steam Dumps are...

a. modulating open due to steam header pressure.

b. modulating open due to Tave above no-load Tave.

c. closed because Tave is NOT greater than 3°F above Tref.

d. closed because the dumps are NOT armed.

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 3 **SRO Group:** 3

041 Steam Dump System and Turbine Bypass Control

A3. Ability to monitor automatic operations of the Steam Dump System and Turbine Bypass Control including:

A3.02 RCS pressure, RCS temperature, and reactor power 3.3 3.4

Explanation of Answer The "A" reactor trip breaker provides the arming signal for dumps on normal reactor trip. Since "A" RTB is still closed, the steam dumps respond to event like load rejection, with C-7 load rejection (10% load decrease in 2 minutes sensed on PT-506) arming the dumps. Since the "B" RTP was opened, the steam dump controller does operate on the plant trip controller (No load Tave compared to Auct Hi Tave).

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Main Steam Dumps	MS-4	Schematic		4	
Chp 24, Steam Dumps	Chp 24	II.A.2.b, c	10-12	1	2, 3.b

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 47 S/G Level program - low power

The following conditions exist on Unit 1:

- Reactor power 35%
- All systems normal

What failure would cause an INITIAL decrease in feedwater flow to all S/Gs?

a. Turbine first stage impulse pressure PT-505 fails low.

b. Main steamline pressure PT-507 fails low.

c. Turbine first stage impulse pressure PT-506 fails low.

d. Main feedwater header pressure PT-508 fails low.

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

059 Main Feedwater System

2.1 Conduct Of Operations

2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation. 3.7 4.4

Explanation of Answer PT-507 fails low causes feed pump speed to decrease which reduces FW pressure. This would initially result in a decrease of flow to all S/Gs.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
FW EH Controls	EHC-6	dP Actual schematic		1	
Chp. 27, Steam Generator Water Level Control System	Chp. 27	I.B.2, II.C.3.I.2)	12, 42-44	1	5, 15.c

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 48 AFW Startup

The following conditions exist on Unit 1:

- The reactor tripped from an at-power condition
- An undervoltage condition exists on RCP 1C bus
- Power Range NIS channel N42 failed at 100% on the trip
- ESF bus 141 undervoltage occurred
- 1A D/G automatically started and ACB 1413 is closed
- S/G levels lowest readings were - 19% (A); 25% (B); 22% (C); 20% (D)

What is the status of the Auxiliary Feedwater (AF) Pumps on Unit 1 for these conditions at ONE minute following the trip?

a. Both AF pumps are running.

b. The 1A AF pump is running and the 1B AF pump is NOT running.

c. The 1B AF pump is running and the 1A AF pump is NOT running.

d. NO AF start signal is initiated.

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

061 Auxiliary / Emergency Feedwater System

A3. Ability to monitor automatic operations of the Auxiliary / Emergency Feedwater System including:

A3.01 AFW startup and flows 4.1 4.2

Explanation of Answer SG levels are above AF actuation setpoints and the motor driven AF pump starts on the detected undervoltage.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Auxiliary Feedwater System	LO-PSC-12C	II.A.3.g; II.A.4.j	4 & 6	2	5
Chp 26, Auxiliary Feedwater System (AF)	Chp 26	II.A.3.c	12	3	3, 5
Chp 9, Diesel Generators & Aux. Systems	Chp 9	III.D.2	49-51	1	7.a

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 49 AFW flow requirements for cooldown

In accordance with the BEPs, which of the following describes the MINIMUM AFW pump flow and S/G configuration necessary to remove all of the reactor decay heat load following a reactor trip from 102% power to preclude entry into loss of heat sink RED path entry?

a. The 1A AF pump supplying 480 gpm to at least ONE S/G with S/G blowdown manually isolated.

b. The 1B AF pump supplying 245 gpm to each of TWO S/G with S/G blowdown in service.

c. The 1A AF pump supplying 170 gpm flow to each of THREE S/Gs with S/G blowdown manually isolated.

d. The 1B AF pump supplying 130 gpm flow to each of FOUR S/Gs with S/G blowdown in service.

Answer C **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

061 Auxiliary / Emergency Feedwater System

K5. Knowledge of the operational implications of the following concepts as they apply to the Auxiliary / Emergency Feedwater System:

K5.02 Decay heat sources and magnitude 3.2 3.6

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Bases - AF System	Byron 1 & 2 ITS	B.3.7.5 - Background	B 3.7-28	A	
Auxiliary Feedwater System	Chp 26	I.C.1, 5	6	3	1, 11

Material Required for Examination

Question Source: New **Question Modification Method:** Significantly Modified

Question Source Comments: Comanche Peak 11/93 NRC Exam

Comment Type **Comment**

Question 50 Evaluation of Electrical Supplies

The following conditions exist:
Unit 1, plant heatup in progress:

- RCS temperature - 190°F
- RCS pressure - 300 psig

Unit 2, stabilized following a reactor trip

- RCS Tave - 557°F
- RCS pressure - 2235 psig

Electrical Lineup:

- SAT 142-1 deenergized
- Bus 141 energized with ACB 1414 and ACB 2414 closed
- Bus 142 energized with ACB 1422 closed and ACB 1423 closed with D/G running
- Bus 241 energized with ACB 2412 closed
- Bus 242 energized with ACB 2422 closed

What is the status of the RH pumps fed from their respective bus under these conditions in relation to Technical Specifications (ITS)?

a. Both Trains of Unit 1 and Unit 2 RH pumps are operable from operable power sources.

b. Both Trains of Unit 2 RH pumps are operable from operable power sources, only B Train of Unit 1 RH pumps is operable from an operable power source.

c. Both Trains of Unit 1 RH pumps are operable from operable power sources, only B Train of Unit 2 RH pumps is operable from an operable power source.

d. Only B Train of Unit 1and Unit 2 RH pumps are operable from operable power sources.

Answer a **Exam Level** S **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

062 A.C. Electrical Distribution

K2. Knowledge of electrical power supplies to the following:

K2.01 Major system loads 3.3 3.4

Explanation of Answer BOP AP-84, Section E.2 identifies that when the busses on each Unit are crosstied, entry into ITS ACTION ITS 3.8.1 & 3.8.3 are applicable. The AC busses are operable and the required diversity of power is available form offsite/on-site D/Gs. ACTION statement is entered on inoperable SAT, but equipment fed from ESF Busses is still operable under given conditions.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
AC Sources - Operating	ITS	3.8.1	3.8-1 - 3.8-5	Amend. 178/165	
1A Diesel Generator Loading Test	PT-11-DG1A	1.3	2	9	
AC Electrical Power Distribution Systems	LO-PSC-01	II.A & II.B.2.d	47; 48	2	19; 24; 25

Topic Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Question 57 DC bus battery charger

The following conditions exist on Unit 1:

- Reactor power - 100%

Investigation has located a ground on the 125 VDC Normal supply to the 1A D/G. What action is required to transfer DC control power to the reserve source?

The Reserve power breaker from...

a. DC 111 will be closed after opening the Normal power breaker and the Reserve power breaker at the D/G control panel.

b. DC 111 will be closed after swapping the no-blow link at the Normal and Reserve power fuse blocks at the D/G control panel.

c. DC 112 will be closed after opening the Normal power breaker and the Reserve power breaker at the D/G control panel.

d. DC 112 will be closed after swapping the no-blow link at the Normal and Reserve power fuse blocks at the D/G control panel.

Answer b **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 1

063 D.C. Electrical Distribution

2.1 Conduct Of Operations

2.1.30 Ability to locate and operate components, including local controls. 3.9 3.4

Explanation of nswer

Reference Title	**Facility Reference Number**	**Section**	**Page**	**Revisio**	**L. O.**
125VDC System	DC-1			0	
DC Control Power Transfer From Normal To Reserve Source	BOP DC-6A1	6, 7	1	51	
Chp 8a, 125 VDC	Chp 8a	II.A.4.b, c	12	1	4.d

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 52 Sequencing of ESF pumps - SI & SI w LOP

Unit 1 was being synchronized to the grid when the following occurred:

- Trip of 345 KV breakers resulted in deenergizing the SATs
- A steamline break occurred that resulted in containment pressure reaching 20 psig 20 seconds after the D/Gs output breakers have closed

When would the 1A SX pump re-start?

a. Following start of the 1A CS Pump.

b. Between the start of the 1A CV pump and the 1A RH pump.

c. Between the start of 1A CC pump and the 1A AF pump.

d. Coincident with the starting of the 1A and 1C RCFCs.

Answer C **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

064 Emergency Diesel Generators

A3. Ability to monitor automatic operations of the Emergency Diesel Generators including:

A3.07 Load sequencing 3.6 3.7

Explanation of Answer The SX pump would be started in this case by the SI signal which is overrides the UV condition. The SX pump starts in following sequence: CV (0 sec); SI ((5 sec); RH (10sec); CS (15-18 secs, if actuation signal present); CC pumps (20 sec); SX pumps (25 sec); AF 1A pump (35 sec); CS pump (40 sec, if actuation signal now present but not present at 18 sec)

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
D/G Relaying	DG-2	Sequencing Order		1	
Chp 9, Diesel Generators & Aux. Systems	Chp 9	III.D.1	47-49	1	7.c
Chp 20, Essential Service Water System	Chp 20	II.C.1.a	60	2	8.a

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 53 RCDT operation - effect of CNMT Isolation

The following conditions exist on Unit 1:

- Unit is in MODE 3
- A cooldown had just been initiated
- Steam Dump Bypass Interlock control switches have just been taken to BYPASS
- No other operator actions have been performed
- The Steam Dump valves fail open and the following parameters are observed:
- RCS temperature - 537°F (A); 539°F (B); 538°F (C); 538°F (D)
- Pzr pressure - 1820 psig
- Pzr level - 10%
- S/G pressure - 850 psig (A); 740 psig (B); 800 psig (C); 750 psig (D)
- S/G flow - 1.0 Mlb/hr (A); 1.5 Mlb/hr (B); 1.1 Mlb/hr (C); 1.6 Mlb/hr (D)
- The level in the RCDT has risen to the alarm setpoint (80%) for REACTOR COOLANT DRAIN TANK UNIT 1 LEVEL HI-LO

Assuming all systems are functioning correctly, what is the status of the RCDT system?

a. BOTH RCDT pumps are running and flow is directed to the Holdup Tanks.

b. BOTH RCDT pumps are running and flow is recirculated back to the RCDT.

c. ONE RCDT pump is running and flow is directed to the Holdup Tanks.

d. NEITHER RCDT pump is running and NO flow exists for the system.

Answer d **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

068 Liquid Radwaste System

A4. Ability to manually operate and/or monitor in the control room:

A4.04 Automatic isolation 3.8 3.7

Explanation of Answer Conditions for low Pzr pressure actuates SI. The coincident CNMT Phase A Isolation signal isolates RCDT valves out. Closure of valve RE9170 causes pumps to stop.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
PRT & RCDT	RY-4	Schematic		2	
Chp 46b Liquid Radwaste System	Chp 46b	II.A.3.a.4), 5), 7), 8)	26	2	6
Chp 61,Engineered Safety Features	Chp 61	II.C.1.a, 2.f	25, 27	2	7.d

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 54 Loss of FHB Overhead Crane rad monitor

The following conditions exist on Unit 2:

- Refueling operations are in progress

While using the Fuel Handling Building Crane to move new fuel into the Spent Fuel Pool, the radiation monitor 0RE-AR039, Fuel Handling Building Crane Monitor, goes into high alarm. What action is affected?

a. Traverse of the Fuel Handing Building Crane bridge and trolley.

b. Both lowering and raising the Fuel Handing Building Crane hoist.

c. Traverse of the Fuel Handing Building Crane trolley and raising the hoist.

d. Raising the Fuel Handing Building Crane hoist.

Answer d **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

072 Area Radiation Monitoring System

K3. Knowledge of the effect that a loss or malfunction of the Area Radiation Monitoring System will have on the following:

K3.02 Fuel handling operations 3.1 3.5

Explanation of Answer Rad monitor prevents raising hoist.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Chp 49, Radiation Monitors	Chp 49	III.C.2.a	34	3	4.a.3)

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 55 Containment Ventilation (Purge)

The following conditions exist on Unit 1:

- Refueling operations are in progress with fuel movement in containment
- Containment purge supply and exhaust valves are open to control Containment pressure
- Containment Fuel Handling Incident Monitor 1RE-AR011 fails LOW

Which of the following describes the required operator response?

a. With Fuel Handling Incident Monitor 1RE-AR012 operable, the required function is met so the valves can remain open indefinitely.

b. With both Containment Atmosphere 1RE-PR011A,B channels operable, the required function is met so that the valves can remain open indefinitely.

c. Movement of irradiated fuel within containment must be suspended immediately.

d. The containment purge supply and exhaust valves must be closed within FOUR hours.

Answer d **Exam Level** S **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

073 Process Radiation Monitoring System

A2. Ability to (a) predict the impacts of the following on the Process Radiation Monitoring System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:

A2.02 Detector failure 2.7 3.2

Explanation of Answer Must isolate the purge.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Containment Ventilation Isolation Instrumentation	Byron 1 & 2 ITS	3.3.6, Table 3.3.6-1 Function 1, REQUIRED ACTIONS A & C	3.3-48	A	
Containment Ventilation Isolation Instrumentation	Byron 1 & 2 ITS Bases	B 3.3.6 Background	B 3.3-157	A	
Chp 49, Radiation Monitors	Chp 49	III.A		3	15.b

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 56 Evaluation of eqpt affected for slow loss

The following conditions exist on Unit 1:

- A unit startup is in progress with reactor power raised above 18%.
- Turbine is at 1800 rpm ready to be synchronized to grid.
- Motor driven feedwater pump is supplying the S/Gs with Feed Reg Bypass valves in AUTO.
- Steam Dump demand in AUTO at 12%.
- Instrument air header pressure begins to slowly drop due to a leak

If the leak CANNOT be isolated and instrument air pressure continues to drop, which of the following would occur?
(Assume NO operator action taken.)

a. AF recirculation flow to the CST would be lost due to AF recirc, 1AF022A, failing closed.

b. Pressurizer level would increase due to charging header flow control valve, 1CV121, failing open.

c. Pressurizer pressure would decrease due to Aux spray isolation, 1CV8145, failing open.

d. Feedwater heater 17A extraction steam would isolate due to emergency drain, 1HD038A, failing closed.

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 3 **SRO Group:** 3

078 Instrument Air System

K3. Knowledge of the effect that a loss or malfunction of the Instrument Air System will have on the following:

K3.02 Systems having pneumatic valves and controls 3.4 3.6

xplanation of nswer Charging flow goes to maximum due to 1CV121 failing open, and letdown isol 1CV459 & 1CV460 fail closed. 'a' is incorrect because both 1A & 1B AF pump recirc valves fail open . 'c' main turbine not directly affected. 'd' not occur because steam dumps fail closed.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Loss of Instrument Air	BOA SEC-4	Table A	5	52	
Chp 53, Service Air and Instrument Air	Chp 53	III.C.2.c	62	1	9

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 57 Effect of loss of DC - CO2 actuation

With the fire protection systems in their normal alignment, what is the affect of a loss of DC power?

Loss of DC control power to the...

a. halon control cabinet will cause halon release in the Upper Cable Spreading Room.

b. battery control panel will cause automatic start of the diesel driven fire pump.

c. fire detection system will cause start of the motor driven fire pump.

d. carbon dioxide system will cause the master EMPC valve to open pressurizing the CO2 header.

Answer d **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

086 Fire Protection System

K4. Knowledge of Fire Protection System design feature(s) and or interlock(s) which provide for the following:

K4.06 CO2 3.0 3.3

Explanation of Answer EMPCs uses DC control power. On loss of power, the master EMPC valves fail open which in turn cause the master discharge/selector valve to open, charging the affected header.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Chp 57, Fire Protection System	Chp 57	II.C.3.a.4)	42	3	8
CO2 Fire Suppression System Reset After Manual Initiation Following Loss Of Power	BOP FP-26	F	3-4	2	

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 58 Evaluate conditions - unwarranted rod withdrawal

The following conditions exist on Unit 1:

- Reactor power is 30%.
- Rod control is in Automatic
- Tref - 564°F
- Tave values - 564°F (A); 565°F (B); 565°F (C); 564°F (D)
- Power Range NI - 31% (N41); 29% (N42), 30% (N43); 30% (N44)
- Control bank D is at 156 steps.

Which condition would result in continuous rod withdrawal?

a. Turbine first stage pressure PT-505 fails to 100%.

b. Power Range channel N41 fails to 20%.

c. Loop A Tcold fails 553°F.

d. Tref signal fails 557°F.

Answer a **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 1

001 Continuous Rod Withdrawal

AA2. Ability to determine and interpret the following as they apply to Continuous Rod Withdrawal:

AA2.05 Uncontrolled rod withdrawal, from available indications 4.4 4.6

Explanation of Answer Input to rod control Tref, auctioneered HIGH Tave & Auctioneered high PRNIs: PT-505 provides input signal for development of Tref. If it fails high Tref goes to maximum value (581°F) and results in rods being withdrawn to match Tave to Tref. PR failure high compares the rate of change of reactor power to the rate of change of turbine powe . Initially high rate of change during failure but rapidly the rate of change falls to zero and so rods may initially begin to insert but quickly stop motion with no more rate of change. Auctioneered high Tave is used and Tcold failing low will remove this input (if previously auctioneered high). Tref failing low will cause rods to move inward to match Tave to Tref.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Rod Control Unit	RD-2	If rods stepping...		20	
Chapter 28: Rod Control System	Chapter 28	II.A.2.b, 3.b, II.B.1.a	22, 26-27, 68	1	4, 7
Uncontrolled Rod Motion	BOA ROD-1	II.C.3.d	4-5	02	2, 3

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 59 Reactivity effect with positive MTC

Which of the following conditions would result in the largest reactivity addition as a result of a 20 step rod withdrawal?

a. Beginning of life with reactor power at 4% at initiation of withdrawal.

b. Beginning of life with reactor power at 80% at initiation of withdrawal.

c. End of life with reactor power at 4% at initiation of withdrawal.

d. End of life with reactor power at 80% at initiation of withdrawal.

Answer ~~a~~ d AP **Exam Level** S **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 1

001 Continuous Rod Withdrawal

AK1. Knowledge of the operational implications of the following concepts as they apply to Continuous Rod Withdrawal:

AK1.03 Relationship of reactivity and reactor power to rod movement 3.9 4.0

Explanation of Answer BOL at low power, MTC may be positive (or is its most positive) therefore an equivalent withdrawal, the amount of reactivity added will be higher.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
B 3.1.3 - MTC	Byron 1 & 2 ITS Bases	LCO	B 3.1-19	A	
Chapter 28: Rod Control System	Chapter 28	III.C.1	84	1	16
Uncontrolled Rod Motion	BOA ROD-1	II.C.4	5	02	2

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 60 P/A vs. Group Step Counters

A Control Bank D rod was dropped from 156 steps. The P-A converter was left at 156 steps when it was to be reset to ZERO steps as directed by procedure BOA ROD-3 "Dropped Rod Recovery".

Select the affect of performing the procedure in this manner?

a. While performing the procedure, the C-11 Rod Stop will be received prior to realigning the rod.

b. While performing the procedure, the Rod Insertion Limit Alarm will be received at a lower rod position than required.

c. After the procedure is complete, Bank C control rods will begin insertion at a lower value of Control Bank D.

d. After the procedure is complete, Bank C control rods will begin insertion at a higher value of Control Bank D.

Answer a **Exam Level** B **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 1

003 Dropped Control Rod

AK3. Knowledge of the reasons for the following responses as they apply to Dropped Control Rod:

AK3.10 RIL and PDIL 3.2 4.2

Explanation of Answer The bank overlap units are bypassed when rods are moved with individual bank selector positions. The P to A converter provides step information to rod position indication including the C-11 circuit. As the individual rod was withdrawn to approximately 67 steps the C11 circuit would sense that bank D was at 223 steps and block outward motion.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
RD Data Logging / Rod Stops	RD-5 / RD-1	P/A & C-11 Rod Stop / Auto Rod Withdrawal ZY442CX		0 / 0	
Chapter 28: Rod Control System	Chapter 28	II.A.7.b, II.B.2	46, 70	1	1.c, 10
Dropped or Misaligned Rod	BOA ROD-3	C.14	13	2	4

Material Required for Examination

Question Source: New **Question Modification Method:** Editorially Modified

Question Source Comments: D.C. Cook 6/13/1995

Comment Type Comment

Question 61 Reason for power reduction

The following conditions exist on Unit 1:

- Reactor power is 82%.
- Bank D rods have just been withdrawn to 168 steps.
- One bank D rod is at 155 steps.

Reactor power is to be reduced prior to rod recovery...

a. to ensure the power increase, when the rod is withdrawn, will NOT cause core design criteria to be exceeded.

b. to ensure the local linear heat rate increase, due to a misaligned rod, will NOT cause the core design criteria to be exceeded.

c. to allow the remaining rods in the bank to be inserted without reaching the RIL.

d. because Nuclear Instrumentation may NOT indicate actual power with a control rod misaligned.

Answer b **Exam Level** S **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

005 Inoperable/Stuck Control Rod

AK1. Knowledge of the operational implications of the following concepts as they apply to Inoperable/Stuck Control Rod:

AK1.06 Bases for power limit, for rod misalignment 2.9 3.8

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
B 3.1.4 - Rod Group Alignment Limits	Byron 1 & 2 ITS Bases	ACTIONS	B 3.1-32	A	
Dropped or Misaligned Rod	BOA ROD-3	I.C.9	8	02	4

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 62 Stabilized RCS temperature with failure of Steam Dumps

On Unit 1, a loss of all circulating water pumps has resulted in a reactor trip. All control systems respond as expected. Significant decay heat causes RCS temperature to increase following the trip.

At what RCS temperature should temperature stabilize?

Temperature should stabilize at...

a. 550°F.

b. 557°F.

c. 561°F.

d. 565°F.

Answer C **Exam Level** B **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

007 Reactor Trip

EA1. Ability to operate and / or monitor the following as they apply to Reactor Trip:

EA1.03 RCS pressure and temperature 4.2 4.1

Explanation of Answer The condenser would NOT be available for steam dumps (either on trip controller or load rejection controller). The S/G pressure would stabilize based on the secondary PORV opening setpoint normally set at 1115 psig. The Main Steam safety valve setting is 1175 psig. At 550°F the steam dumps would be blocked (P12).

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Main Steam Dumps	MS-4	C-9, If Main Condenser NOT available		4	
Chp 24, Steam Dumps	Chp 24	II.C.1.b	24	1	4.b
Chp. 23, Main Steam System	Chp. 23	II.A.3.b	18	2	3

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 63 BEP-0 indications

The following conditions exist on Unit 1:

- Reactor trip occurred 3 minutes ago
- Main feedwater failed to isolate
- IR "A" SUR - -.25 dpm
- IR "B" SUR - -.25 dpm

What action should be taken?

a. Continue in BEP ES-0.1," Reactor Trip Response."

b. Transition to BFR-S.1, "Response to Nuclear Power Generation/ ATWS."

c. Enter applicable LCO due to loss of IR instruments and transition to BFR-P.1, " Response to Imminent Pressurized Thermal Shock Condition."

d. Transition to BEP ES-0.0, "Rediagnosis."

Answer a **Exam Level** S **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

007 Reactor Trip

EA2. Ability to determine and interpret the following as they apply to Reactor Trip:

EA2.01 Decreasing power level, from available indications 4.1 4.3

Explanation of Answer With SUR negative, verification of reactor trip is satisfied and operators should remain in BEP-0. IR instruments do not indicate abnormally under given conditions.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Emergency Procedures - BEP-0, BEP ES-0.0 - .4 lesson Plan	BEP-0	step 1 (RNO)	3	3	7

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 64 Evaluation of PORV leak - Tech Spec Limits

The following conditions exist on Unit 1:

- The reactor is operating at 100% power
- Previous RCS leakrate surveillance information:
  - Identified leakage - 2.7 gpm
  - Unidentified leakage - 0.3 gpm
  - S/G Leakage - 0.0 gpm
- A pressurizer PORV is leaking to the PRT at a rate of 1.5 gpm as measured over the past hour
- The Block valve associated with the PORV is open

Which of the following describes the Technical Specifications requirement for this situation?

a. Declare the PORV inoperable, close and deenergize the associated Block Valve.

b. Declare the PORV inoperable and initiate shutdown.

c. IDENTIFIED leakage is within allowed limits.

d. UNIDENTIFIED leakage requires shutdown.

Answer C **Exam Level** S **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

008 Pressurizer Vapor Space Accident

2.2 Equipment Control

2.2.22 Knowledge of limiting conditions for operations and safety limits. 3.4 4.1

Explanation of Answer The leakage for the PORV is considered IDENTIFIED LEAKAGE (Leakage that is captured and conducted to collection system or a sump or collecting tank. Leakage past PORV is directed to PRT.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
RCS Operational Leakage	Byron 1 & 2 ITS	3.4.13	LCO 3.4.13	A	
BASES - Pressurizer PORVs / RCS Operational Leakage	Byron 1 & 2 ITS BASES	LCO / LCO C	B 3.4-59 / B 3.4-86	A	
Chp 12, Reactor Coolant System	Chp 12	III.A.1		2	11.q

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 65 Calculation of subcooled margin on Iconics

The following conditions exist on Unit 1:

- Subcooling Margin output from the SPDS Iconics has failed
- 1C RCP and 1D RCP are running

The Unit Supervisor has asked you to determine the subcooling margin using the same valid inputs as used by SPDS.

What are the parameters used to calculate subcooling margin?

a. RCS wide range pressure from loop C hot leg and core exit thermocouple temperatures.

b. Pressurizer pressure and core exit thermocouple temperatures.

c. RCS wide range pressure from loop A and loop C hot leg, and RCS loop A and loop C hot leg temperatures.

d. Pressurizer pressure and RCS loop C and loop D hot leg temperatures.

Answer a **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

009 Small Break LOCA

EA1. Ability to operate and / or monitor the following as they apply to Small Break LOCA:

EA1.10 Safety parameter display system 3.8 3.9

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.	
SPDS Display	CX-1	Subcooling		1		
Chapter 34b Inadequate Core Cooling Detection		Chapter 34b	II.A.3	22-23	2	6

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 66 Effects of Adverse containment

High containment pressure is used to determine adverse containment values...

a. to expedite EOP recovery actions when SI has actuated.

b. because it provides a backup indication of containment radiation.

c. because it corresponds to containment saturation temperature.

d. to provide a redundant indication whenever any SI actuation signal is received.

Answer C **Exam Level** S **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

009 Small Break LOCA

EK3. Knowledge of the reasons for the following responses as they apply to Small Break LOCA:

EK3.16 Containment temperature, pressure, humidity and level limits 3.8 4.1

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
LOSS OF REACTOR OR SECONDARY COOLANT	1BEP-1	ATTACHMENT 25 A		1A WOG-1 B	
LOSS OF REACTOR OR SECONDARY COOLANT, BEP ES-1.1, ES-1.2, ES-1.3 AND ES-1.4	1BEP 1			2	2.d

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments: Kewaunee 2/94

Comment Type **Comment**

Question 67 RCP trip criteria evaluation

The following conditions exist during performance of BEP-0.

- Train A ECCS pumps failed to start.
- RCS pressure is 1350 psig.
- Containment pressure of 7 psig.
- Bus 142 has an overcurrent trip on the normal feeder breaker.
- SI actuated due to High Containment Pressure.
- The highest critical safety function is Yellow on Heat Sink.
- All other equipment and components operated as expected.

Based on above plant conditions, the RCPs should...

a. remain running because NO SI pumps or Charging Pumps are running.

b. be stopped because RCS pressure is below the RCP trip criteria.

c. remain running until Pressurizer level decreases below 34%.

d. be stopped because CC flowpath to the RCP motor oil coolers is isolated.

Answer a **Exam Level** B **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 1

011 Large Break LOCA

EA1. Ability to operate and / or monitor the following as they apply to Large Break LOCA:

EA1.03 Securing of RCPs 4.0 4.0

Explanation of Answer The trip criteria is < 1425 psig, with NO cooldown in progress, and HHSI flow > 50 gpm or SI flow > 100 gpm.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Operator Action Summary for 1BEP-0	1BEP-F:0	TRIP RCPs		1C WOG-1 B	
Emergency Procedures - BEP-0 REACTOR TRIP OR SAFETY INJECTION. BEP ES-0.0 - 0.4	BEP-0	Rx Coolant Pump Trip Criteria	3	3	5, 7

Material Required for Examination

Question Source: New **Question Modification Method:** Significantly Modified

Question Source Comments: Watts Bar 3/3/1995

Comment Type **Comment**

Question 68 **Pzr level requirements**

The following conditions exist on Unit 1:

- A small break LOCA has occurred
- RCS pressure is 2200 psig
- CETC temperature 550°F on Iconics

The actions of BEP-0 and BEP-1 have been completed and the crew has transitioned to BEP ES-1.1, SI Termination.

Which of the following conditions would REQUIRE the operator to manually start ECCS pumps and realigning SI? (Adverse containment conditions do NOT exist)

a. Establishing normal charging during SI termination results in pressurizer level decreasing to 3%.

b. A pressurizer PORV fails open causing RCS pressure to decrease to 1500 psig prior to PORV isolation.

c. A steam generator atmospheric relief valve fails open causing a faulted steam generator.

d. Radiation readings on one main steam line are observed increasing.

Answer a **Exam Level** S **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 1

011 Large Break LOCA

EA2. Ability to determine and interpret the following as they apply to Large Break LOCA:

EA2.04 Significance of PZR readings 3.7 3.9

Explanation of Answer Conditions in BEP ES-1.1: 1) RCS subcooling NOT acceptable; OR PZR level CANNOT be maintained > 4%

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Operator Action Summary for 1BEP-1 Series Procedure	1BEP-F.1	SI Reinitiation		1 WOG-1 B	
LOSS OF REACTOR OR SECONDARY COOLANT, BEP ES-1.1, ES-1.2, ES-1.3 AND ES-1.4	1BEP 1	II.C.	9	2	1.f

Material Required for Examination Figure 1BEP 1-1

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 69 Use of Adverse containment

What are the guidelines for use of adverse containment values during a LOCA?

a. If containment pressure or radiation exceeds the stated value on the OAS page, adverse containment values must be used for the duration of the event regardless if either value decreases to its normal value.

b. If adverse conditions were entered due to both pressure and radiation, a return to normal values in the procedures is allowed any time when pressure and radiation readings decrease to less than the ATTACHMENT A values.

c. Once in adverse conditions, a return to normal procedure values can be made immediately if containment pressure was the ONLY reason adverse conditions had been declared and has returned to less than the ATTACHMENT A value.

d. Once in adverse conditions, a return to normal procedure values can be made immediately if containment radiation was the ONLY reason adverse conditions had been declared and has returned to less than the ATTACHMENT A value.

Answer C **Exam Level** S **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 1

011 Large Break LOCA

EK3. Knowledge of the reasons for the following responses as they apply to Large Break LOCA:

EK3.12 Actions contained in EOP for emergency LOCA (large break) 4.4 4.6

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Reactor Trip or Safety Injection	1BEP-0	NOTE step 1, ATTACHMENT A, A.1)	3, 38	1C WOG-1 B	
Emergency Procedures - BEP-0 REACTOR TRIP OR SAFETY INJECTION. BEP ES-0.0 - 0.4	BEP-0	NOTE 2, step 1	3	3	4

Material Required for Examination

Question Source: New **Question Modification Method:** Editorially Modified

Question Source Comments: Turkey Point 4/92

Comment Type **Comment**

Question 70 Eval loss of cooling flow

On a loss of seal injection to the RCPs, what criteria is used to determine if the RCPs should be tripped per BOA RCP-2 "Loss Of Seal Cooling"?

a. High temperatures on the RCP lower bearing outlet temperatures.

b. Time elapsed since loss of seal injection.

c. RCP Thermal Barrier Component Cooling Water low flow alarms.

d. High vibration condition on the RCP.

Answer a **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

015 Reactor Coolant Pump Malfunctions

AA2. Ability to determine and interpret the following as they apply to Reactor Coolant Pump Malfunctions:

AA2.10 When to secure RCPs on loss of cooling or seal injection 3.7 3.7

Explanation of Answer Seal & bearing temperatures are monitored for trip setpoint.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Loss of Seal Cooling	BOA RCP-2	step 1 RNO	2	54	
Abnormal Operating Procedures, Loss of Seal Cooling	1BOA RCP-2	II. 1.a	4-5	1	6

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 71 Eval of RCP seal failure

Unit 1 is operating at 100% power when the following alarms are received/reported:

- RCP SEAL LEAKOFF FLOW LOW (1-7-C3)

The NSO investigates and reports the following additional information:

- RCP 1A seal injection flow is 10.7 gpm
- #1 Seal Leakoff Flow on 1A RCP is 0.4 gpm
- RCP 1A Seal Water Outlet Temperature is 140°F and STABLE
- RCP 1A Bearing Outlet Temperature is 145°F and STABLE
- Unit 1 RCDT level indicates 75%

Based on the above information, which of the following events has occurred?

a. RCP 1A #1 Seal has failed closed

b. RCP 1A #1 Seal has failed open.

c. RCP 1A #2 Seal has failed closed.

d. RCP 1A #2 Seal has failed open.

Answer d **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

015 Reactor Coolant Pump Malfunctions

AK2. Knowledge of the interrelations between Reactor Coolant Pump Malfunctions and the following:

AK2.07 RCP seals 2.9 2.9

xplanation of nswer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.	
Reactor Coolant Pump Seal Failure	1BOA RCP-1	steps 3-5 & 8	4-7	55B		
BOA RCP-1, REACTOR COOLANT PUMP Seal Failure		BOA RCP-1	C. 6	8	2	2, 7

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: Braidwood bank

Comment Type **Comment**

Question 72 VCT level transmitter malfunction

Given the following:

- The plant is at 90% power with ALL controls in AUTO.
- VCT level transmitter, LT-112, fails HIGH causing a letdown diversion.
- At the time of failure VCT level transmitter, LT-185, reads 50%.

What will occur if NO operator action is taken?

VCT level decreases...

a. until Auto makeup starts and maintains VCT level.

b. with NO auto makeup capability and charging suction shifts to RWST.

c. faster than auto makeup input and charging suction shifts to RWST.

d. until charging pumps lose suction and start to cavitate.

Answer d **Exam Level** B **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

022 Loss of Reactor Coolant Makeup

AA1. Ability to operate and / or monitor the following as they apply to Loss of Reactor Coolant Makeup:

AA1.08 VCT level 3.4 3.3

Explanation of Answer LT 112 provides for AUTO makeup to the VCT. If NO operator action taken, then level will continue to fall until NPSH is lost to the CENT CHG pump(s). Transfer will NOT occur to RWST since both channels are required for swap. An alarm will be generated from LT-185 at 20% level.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
CVCS Notes	CV-2	LT-112 Table		3	
Chp 15a - Chemical and Volume Control System	Chp 15a	II.A.1.n.3).g), h), I)	35-37	2	11

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 73 Time/amount E-boration for condition

Given the following after a reactor trip:

- THREE rods remain withdrawn.
- Due to equipment malfunctions boration is only available from the RWST.
- Charging flow rate 132 gpm.
- RCS boron concentration was 1050 prior to the trip.
- 120 gpm letdown in service.

Of the listed times, which would be minimum acceptable total time that boration from the RWST would have to occur?

a. 1 hour.

b. 2 hours.

c. 3 hours.

d. 4 hours.

Answer c **Exam Level** B **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

024 Emergency Boration

AA2. Ability to determine and interpret the following as they apply to Emergency Boration:

AA2.05 Amount of boron to add to achieve required SDM 3.3 3.9

Explanation of Answer 1BEP ES-0.1 requires 5500 gallons boration from RWST for each rod not fully inserted, therefore requiring 16,500 gallons. The net turnover rate in the RCS is 120 gpm, then required time is 16,500/120 = 137.5 minutes (rounded up to 140 min.). Other answers based on counting 2 rods and/or borating from CV-8104 @ 57 gpm with total of 1320 x # rods out gallons.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Emergency Boration	BOA PRI-2	1 RWST 4)	3	55B	
Abnormal Operating Procedures Emergency Boration	BOA PRI-2	3.a	8-9	1	4, 6
Emergency Procedures - BEP-0 REACTOR TRIP OR SAFETY INJECTION. BEP ES-0.0 - 0.4	BEP ES-0.1	5	6	3	7

Material Required for Examination 1BEP ES-0.1, page 6 (step 5)

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 74 Calc of time to saturation/core boiling

The following conditions exist on Unit 1:

- A forced outage is in progress
- The plant was shutdown 8½ days ago to repair a steam generator tube leak.
- Draining of the RCS was initiated to allow access to S/Gs.
- Reactor vessel level is at 397' 1" with Thot at 212°F.
- A loss of RHR pumps due to cavitation has occurred

Which of the following is the smallest amount of flow that meets the minimum makeup flow required to maintain current RCS level?

a. 80 gpm.

b. 72 gpm.

c. 65 gpm.

d. 59 gpm.

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

025 Loss of Residual Heat Removal System

AK1. Knowledge of the operational implications of the following concepts as they apply to Loss of Residual Heat Removal System:

AK1.01 Loss of RHRS during all modes of operation 3.9 4.3

Explanation of Answer 81/2 days is 204 after shutdown. The curve shows minimum flow at approximately 70 gpm.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
OSS OF RH COOLING	1BOA PRI-10	Fig. 1BOA PRI 10-1	9	56	
BOA PRI-10 Loss of RH Cooling	BOA PRI-10	Fig 1BOA PRI 10-1	9	2	5

Material Required for Examination Figure 1BOA PRI 10-1

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 75 Alternate RCS cooling

The following conditions exist on Unit 2:

- MODE 5 operation during normal cooldown
- RCS temperature - 195° F
- RCS pressure - 325 psig
- Train A RH in service, train B RHR tagged out for repairs

What is the preferred method of core cooling if a loss of RH cooling occurs?

Alternate RCS cooling using...

a. the SI accumulators.

b. the S/Gs.

c. normal charging and RHR letdown.

d. SI Pump hot leg injection.

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

025 Loss of Residual Heat Removal System

AK3. Knowledge of the reasons for the following responses as they apply to Loss of Residual Heat Removal System:

AK3.01 Shift to alternate flowpath 3.1 3.4

Explanation of Answer Steaming intact/non-isolated SGs is the preferred alternate decay heat removal method if the RCS is intact.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
OSS OF RHR SHUTDOWN COOLING	1BOA PRI-10	Table A	8	56	
3OA PRI-10 Loss of RH Cooling	BOA PRI-10	Seven Attachments (Attachment B	8	02	4

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 76 Evaluation of CCW leak

The following conditions exist on Unit 1:

- The reactor is shutdown.
- RHR is in shutdown cooling.
- RCS temperature is 300ºF.
- RCS pressure is 160 psig.
- CCW surge tank level is decreasing

What leak locations will produce these indications?

a. RHR Heat Exchanger.

b. Thermal Barrier Heat Exchanger.

c. Letdown Heat Exchanger.

d. Seal Water Heat Exchanger.

Answer d **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

026 Loss of Component Cooling Water

AA1. Ability to operate and / or monitor the following as they apply to Loss of Component Cooling Water:

AA1.05 The CCWS surge tank, including level control and level alarms, and radiation alarm 3.1 3.1

Explanation of Answer The seal water HX would be the only location where the CC pressure would be lower than the process fluid pressure. RHR HX approx. 165 psig; L/D Hx pressure should be approximately 160 psig; & Thermal barrier pressure should be about 160 psig.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Component Cooling Malfunctions	BOA PRI-6	Attachment B, steps 1, 2	1-2	56	
Abnormal Operating Procedures, Component Cooling Malfunctions	BOA PRI-6	Attach B, step 2	13	1	3.c, 5

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: Zion 7/13/92

Comment Type **Comment**

Question 77 Determine Tech Spec limitation on eqpt outage

The following conditions exist on Unit 1:

- RCS Temperature is 185ºF.
- S/Gs are drained
- The "A" loop of RH is in Shutdown Cooling.

What is the maximum time the B Train safety loop of Component Cooling Water could be isolated for required testing without requiring additional Technical Specification (ITS) ACTIONS?

(Assume all other requirements are met)

a. 1 hour.

b. 2 hours.

c. 7 hours.

d. 12 hours.

Answer D **Exam Level** S **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

026 Loss of Component Cooling Water

2.2 Equipment Control

2.2.24 Ability to analyze the affect of maintenance activities on LCO status. 2.6 3.8

Explanation of Answer For operable RCS loops in MODE 5 loops filled, requires ONE RH loop operable and either 1) one additional RHR loop operable OR 2) TWO SGs operable. Since NO SGs operable, the RHR loop must be operable. However, NOTES in the LCO allow ONE RHR loop to be inop for up to 2 hours for surveillance testing.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
RCS Loops - MODE 5	Byron 1 & 2 ITS	3.4.7 LCO Notes	3.4-14	A	
RCS Loops - MODE 5 - B 3.4.7	Byron 1 & 2 ITS BASES	Background, LCO	B 3.4-36, B 3.4-38	A	
Chp 19, Component Cooling System (CC)	Chp 19	I.B.6.f, III.B.2	10, 48	3	3.e, 14

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 78 Pressure controller step change

The following conditions exist on Unit 2:

- Reactor power is 100%
- Pressurizer pressure control is in automatic.

What is the immediate response of the pressure control system if the Master Pressure Controller setpoint is inadvertently changed to 2330 psig (step change)?

a. PORV RY455A opens and spray valves open.

b. PORV RY455A opens, spray valves open, and all heaters energize.

c. Spray valves open and proportional heaters go to minimum.

d. Spray valves close and proportional heaters go to maximum.

Answer d **Exam Level** B **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 2

027 Pressurizer Pressure Control Malfunction

AA1. Ability to operate and / or monitor the following as they apply to Pressurizer Pressure Control Malfunction:

AA1.01 PZR heaters, sprays, and PORVs 4.0 3.9

Explanation of Answer Setting the pot setting higher reduces the output from the controller and raises the demanded pressure setpoint. This reduction results in spray valve closure & heaters turning fully on.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Pzr Pressure Control	RY-2	PK-455A in "AUTO"		3	
Chp 14, Pressurizer (RY)	Chp 14	II.C.1.a.2), II.C.1.c.	53	3	21

Material Required for Examination

Question Source: New **Question Modification Method:** Significantly Modified

Question Source Comments: Calvert Cliffs 11/97

Comment Type **Comment**

Question 79 Non-Controlling channel failure

The following conditions exist on Unit 1:

- Reactor power is 100%
- All systems are in automatic
- Pressurizer pressure channels PT-456 and PT-458 reads normal
- Channel I Pressurizer Pressure Channel (PT-455) was declared inoperable and taken out of service with the appropriate bistables placed in the tripped condition .
- Controlling pressurizer pressure channel (PT-457) fails high

Assuming NO operator action, what is the plant response to the channel failure?

a. Both PORVs and both spray valves open resulting in a reactor trip from low pressurizer pressure followed by SI actuation.

b. The reactor will trip on high pressure, and safety injection will actuate on low pressure due to spray valve operation.

c. Pressurizer proportional heaters will de-energize and spray valves will open resulting in an OTdT runback prior to reactor tripping, and SI will actuate due to low pressurizer pressure.

d. Both PORVs and both spray valves remain closed while pressurizer heaters de-energize.

Answer b **Exam Level** B **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 2

027 Pressurizer Pressure Control Malfunction

AA2. Ability to determine and interpret the following as they apply to Pressurizer Pressure Control Malfunction:

AA2.15 Actions to be taken if PZR pressure instrument fails high 3.7 4.0

xplanation of Answer TWO PZR pressure channels will have HIGH PZR PRESSURE bistables actuated resulting in the reactor trip. The sprays will have modulated fully open resulting in actual pressure decreasing (PORV 1RY455A would have also opened on the failure of PT-457, but would close when the PZR pressure fell to 2185 psig PT-458 will actuate the low pressure interlock closing the PORV) until SI occurs at 1829 psig.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Pzr Pressure Control	RY-2	Pzr Press Channels schematic		3	
Chp 14, Pressurizer (RY)	Chp 14	III.C.2.d	81	3	21

Material Required for Examination

Question Source: New **Question Modification Method:** Significantly Modified

Question Source Comments: BV 8/91

Comment Type **Comment**

Question 80 Failed level channel low.

The plant is operating at 100% power with all control systems in AUTO. The following parameters are noted:

- Letdown Hx outlet flow (FI-132) - 75 gpm
- Charging Header flow (FI-121) - 87 gpm
- Total seal injection flow (FI-142 -FI -45) - 33 gpm

What is the effect on total seal injection flow initially if controlling Pzr level channel LT-459 fails LOW?

Total seal injection flow will...

a. decrease to 0 gpm.

b. decrease to approximately 20 gpm.

c. remain approximately 33 gpm.

d. increase to greater than 40 gpm.

Answer d **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 3 **SRO Group:** 3

028 Pressurizer Level Control Malfunction

AK3. Knowledge of the reasons for the following responses as they apply to Pressurizer Level Control Malfunction:

AK3.05 Actions contained in EOP for PZR level malfunction 3.7 4.1

Explanation of Answer The failure of the level instrument low increases charging flow and charging discharge header pressure. Since seal injection flow is normally increased by throttling close on CV182 to increase backpressure, the result is the same and seal injection flow will increase.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.	
CVCS notes	CV-2	CVCS ratings		2		
Operation with a Failed Instrument Channel	1BOA INST-2	Attachment C, step 1	1	56A	3	
Abnormal Operating Procedures, Operation with a Failed Instrument Channel		1BOA INST-2	Attachment C, step 1	35	1	3

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: Braidwood 1996 NRC exam. Modified premise from failed controller to failed level channel. Changed location of correct answer based on different response (increasing flow instead of decreasing flow).

Comment Type **Comment**

Question 81 AMS conditions

The following conditions exist on Unit 1:

- At t= 0 sec, Turbine load was decreased below 352 MW (30% power)
- At t=240 sec, The running main feedwater pump tripped.
  The reactor did NOT trip due equipment malfunction.
- At t=250 sec, All feedflow indications decrease to 0% flow
- At t=320 sec, All steam generator levels decrease below 15%.

Based on this information, AMS would...

a. initiate at t=320 sec.

b. initiate at t=345 sec.

c. initiate at t=360 sec.

d. NOT initiate because C-20 is cleared.

Answer b **Exam Level** B **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 1

029 Anticipated Transient Without Scram

2.4 Emergency Procedures / Plan

2.4.48 Ability to interpret control room indications to verify the status and operation of system, and understand how operator actions and directives affect plant and system conditions. 3.5 3.8

Explanation of Answer AMS remains armed for 6 minutes(360 sec) following decrease below 30%(C-20). The actuation signal is generated after 3/4 SGs level have fallen 3% below the LO-2 (reactor trip) setpoints of 18% for a period of 25 seconds. C-20 would clear @ t=360sec. AMS actuation occurs at 320 + 25 = 345 sec.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
.MS	PN-3	Logic #1 schematic		2	
Chapter 60c, AMS	Chapter 60c	I.B.1.b, II.B.1.f, 2.c	2, 9	2	2

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 82 Evaluation of SR NIS voltage failure

The following conditions exist on Unit 1:

- Reactor startup in progress
- Intermediate power range indication: 2.5E-5 amp N35 & 2.8E-5 amp N36
- SOURCE RANGE PERMISSIVE P-6 permissive light clear
- Source Range Channel N31 high voltage power supply fails to HALF its normal value

What indication(s) would be available to alert the operator to this failure?

a. None, until power is lowered below the P-6 setpoint, and then the Source Range N31 indication will indicate lower than expected.

b. None, until power is lowered below the P-6 setpoint, and then the Source Range N31 indication will indicate higher than expected.

c. Annunciator SR HIGH VOLTAGE FAILURE (1-10-B1) will remain in alarm when power exceeds P-10.

d. Annunciator SR HIGH VOLTAGE FAILURE (1-10-B1) will re-flash when the voltage source fails.

Answer a **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

032 Loss of Source Range Nuclear Instrumentation

AK1. Knowledge of the operational implications of the following concepts as they apply to Loss of Source Range Nuclear Instrumentation:

AK1.01 Effects of voltage changes on performance 2.5 3.1

Explanation of Answer Based on Gas filled detector curve (Region III), the number of events collected would drop (counts drop). Alarm and voltage input to SR detector is blocked until both IR NIS fall below the P-6 setpoint.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
SR HIGH VOLT FAILURE	BAR 1-10-B1	Setpoint, NOTEs	1	51	
Source Range Detector	NI-4	SR schematic, 1950V DC to Preamp, Gas filled detector six-region curve		4	
Chp 31, Source Range Nuclear Instrumentation	Chp 31	I.B.6.c.3), III.B.2.g, h	10, 64	1	11.b

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 83 Eval of failed IR channel on SU

The following conditions exists on Unit 2:

- Plant shutdown is in progress.
- Power range channels indicate: 9% (N41), 10% (N42), 11% (N43) , 11% (N44)
- Intermediate range channel N-36 fails HIGH.

When this failure occurs, what is the plant response this failure?

a. The reactor will trip on high IR flux, and source range trip will reinstate when N-35 decreases below P-6.

b. The reactor will trip on high IR flux, and source range trip will NOT be automatically reinstated.

c. The reactor will NOT trip immediately, but will trip when the source range trip is reinstated when N-35 decreases below P-6

d. The reactor will NOT trip, and source range trip will NOT be automatically reinstated.

Answer d **Exam Level** B **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

033 Loss of Intermediate Range Nuclear Instrumentation

AA2. Ability to determine and interpret the following as they apply to Loss of Intermediate Range Nuclear Instrumentation:

AA2.04 Satisfactory overlap between source-range, intermediate-range and power-range instrumentation 3.2 3.6

Explanation of Answer Since reactor power is < P-10 setpoint (10% power), the IR trip setpoint at 25% EICA will be exceeded resulting in reactor trip. SR will NOT be reinstated automatically because only one IR channel will fall below P-6 and Two are required to remove P-6.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Intermediate Range	NI-3	IR High Flux schematic		4	
Chp 32, Intermediate Range Nuclear Instrumentation Sys	Chp 32	II.C.1, II.C.2	25-27	1	4.a, 4.c

Material Required for Examination

Question Source: New **Question Modification Method:** Significantly Modified

Question Source Comments: Watts Bar 8/94

Comment Type **Comment**

Question 84 S/G tube leak increase actions

The following conditions exist on Unit 1:

- Reactor power is 100%
- A known tube leak of .05 gpm exists in the A S/G.
- The time now is 1200.
- At 1300 the leak increases to .075 gpm.
- At 1400 the leak increases to 0.1 gpm.

What action, if any, is required as a result of the increasing S/G tube leak as directed by 1BOA SEC-8 "Steam Generator Tube Leak"?

a. Operation of the unit can continue at less than 30% power.

b. Operators must complete a reactor shutdown within 10 hours.

c. Operators must complete a reactor shutdown within 4 hours .

d. The unit must be tripped immediately per Technical Specifications.

Answer C **Exam Level** S **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

037 Steam Generator Tube Leak

2.1 Conduct Of Operations

2.1.32 Ability to explain and apply all system limits and precautions. 3.4 3.8

Explanation of Answer The leak rate in GPD 1200 - 72 gpd, 1300 - 108 gpd, 1400 - 144 gpd, change of 36 gpd/hr. Per SEC-8 action (evaluation) is required because rate > 50 gpd. Normally with rate < 150 gpd in each SG, operation may continue. With rate > 150 gpd, shutdown within 10 hours. BUT with consecutive leak rates showing > 25 gpd increase in one hour OR if rate > 10 gpm (14,400 gpd), TS 3.4.13 is applicable and shutdown is required within 4 hours while continuing with actions of SEC-8.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Steam Generator Tube Leak	1BOA SEC-8	step 6, 8	5-6	54A	
Abnormal Operating Procedures Steam Generator Tube Leak	1BOA SEC-8	6, 8.b	5-6	4	6, 9
BGP 100-4 Power Descention	BGP 100-4	NOTE step 1	5	03	2, 5

Material Required for Examination BOA SEC-8, pages 5 and 6. (Steps 6-8)

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 85 Loss of subcooling

BEP-3 "Steam Generator Tube Rupture" is being performed in response to a tube rupture on 2C S/G. The cooldown has just been completed but the target temperature value selected by the operators was higher than that stipulated in the procedure.

What condition could result because of this error?

a. Loss of RCS subcooling before RCS and ruptured S/G pressures are equalized.

b. Increase in pressure of the ruptured S/G with resultant lifting of the S/G Safety Valve.

c. Increase in pressure of the non-ruptured S/Gs with resultant lifting of their S/G Safety Valves.

d. Filling the Pressurizer solid during the subsequent depressurization.

Answer a **Exam Level** B **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

038 Steam Generator Tube Rupture

EK3. Knowledge of the reasons for the following responses as they apply to Steam Generator Tube Rupture:

EK3.06 Actions contained in EOP for RCS water inventory balance, S/G tube rupture, and plant shutdown procedures 4.2 4.5

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Emergency Operating Procedures, EP-3,Steam Generator Tube Rupture, ES 3.1-3.3	BEP 3	13,	17	2	1, 8
ERG Basis				6	

aterial Required for Examination

uestion Source: New **Question Modification Method:** Editorially Modified

Question Source Comments: Salem 6/94

Comment Type **Comment**

Question 86 Steamline isolation

The following conditions exist on Unit 1:

- The Unit was in MODE 3 at normal operating temperature and pressure prior to the event.
- A faulted steam generator has occurred.
- RCS hot leg temperatures - 547°F (A), 544°F (B), 545°F (C), 547°F (D)
- RCS cold leg temperatures - 545°F (A), 530°F (B), 543°F (C), 545°F (D)
- S/G pressures - 700 psig (A), 635 psig (B), 690 psig (C), 705 psig (D)
- S/G flow - 0.85 MLB/hr (B)
- Containment pressure (Channel) - 8 psig (1), 7.5 psig (2), 7.5 psig (3), 8 psig (4)

Based on these conditions, a main steam line isolation should...

a. have occurred because of the low pressure in at least ONE S/G.

b. have occurred because the steamline high negative rate occurred in S/G 1B.

c. NOT have occurred because Containment pressure is below the setpoint for the CNMT High-2 pressure signal.

d. NOT have occurred because THREE S/Gs have pressures above the isolation setpoint and do NOT indicate high steam flow.

Answer a **Exam Level** B **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

040 Steam Line Rupture

AA1. Ability to operate and / or monitor the following as they apply to Steam Line Rupture:

AA1.01 Manual and automatic ESFAS initiation 4.6 4.6

planation of nswer The steamline isolation signal is generated by the low pressure sensed on 2/3 pressure transmitters in any one SG. CNMT pressure is below the MSLI setpoint of 8.2 psig and steamline negative rate is blocked since initial condition has PZR pressure > P-11.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
ESF Setpoints	EF-2	Steamline Isolation Signals		5	
Chp. 23, Main Steam System	Chp. 23	II.C.2.c	52	2	5.d

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 87 Eval of Leak

The following conditions exist on Unit 1 following a trip from 100% power:

- Pressurizer level is 0%
- Pressurizer pressure is 1500 psig
- Containment Pressure is 16 psig.
- Tcold is 420°F for all loops.

Where is the location of the leak?

a. On one loop RCS cold leg.

b. On a Main Steam Line inside containment.

c. In a Steam Generator Tube.

d. On a feedwater line between the Feed Reg Valve and the associated Feed Water Isolation Valve, 1FW009.

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

040 Steam Line Rupture

AK1. Knowledge of the operational implications of the following concepts as they apply to Steam Line Rupture:

AK1.06 High-energy steam line break considerations 3.7 3.8

Explanation of Answer Secondary LOCA is indicated by the cold leg temperatures on all loops dropping to low temperature with the RCPs running. With LOCA expected cold leg temperature on affected loop is higher due to reversal of flow in that loop due to blowdown. Expected temperature drop is NOT as severe. The LOCA is inside CNMT as indicated by CNMT pressure rise. Feedline loop is outside CNMT.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
mergency Procedures, BEP-0 Reactor Trip Or Safety Injection, BEP ES-0.0 - 0.4	BEP-0	I.B.4.b	12	3	6, 7
Emergency Operating Procedures, BEP-2, Faulted SG Isolation	BEP-2	7	9	2	2

Material Required for Examination

Question Source: New **Question Modification Method:** Editorially Modified

Question Source Comments: St. Lucie 10/13/97

Comment Type **Comment**

Question 88 Eval of conditions

In accordance with BOA SEC-3, "Loss of Condenser Vacuum", which of the following sets of conditions requires the operator to trip the reactor?

a. LOW POWER TRIP BLOCKED P-8 annunciator - LIT
Turbine load - 200 MW
Condenser pressure - 5.2 " HgA

b. LOW POWER TRIP BLOCKED P-8 annunciator - LIT
Turbine load - 300 MW
Condenser pressure - 6.3" HgA

c. LOW POWER TRIP BLOCKED P-8 annunciator - CLEAR
Turbine load - 600 MW
Condenser pressure - 7.2" HgA

d. LOW POWER TRIP BLOCKED P-8 annunciator - CLEAR
Turbine load - 900 MW
Condenser pressure - 7.8" HgA

Answer b **Exam Level** B **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

051 Loss of Condenser Vacuum

AA2. Ability to determine and interpret the following as they apply to Loss of Condenser Vacuum:

AA2.02. Conditions requiring reactor and/or turbine trip 3.9 4.1

Explanation of Answer P-8 permissive active below 30% power (annunciator lit). At 480 MW and below, the minimum acceptable condenser pressure is 5.5 in HgA. At 600 MW minimum acceptable pressure is 7. 8 in HgA. At 610 MW and greater, minimum acceptable pressure is 8.0 in HG

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Loss of Condenser Vacuum	BOA SEC-3	step 5, Figure 1BOA SEC-3-1	5		
Abnormal Operating Procedures, Loss Of Condenser Vacuum	BOA SEC-3	5, Q4	11	1	5, 6

Material Required for Examination Figure 1BOA SEC 3-1

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 89 Operation of limited cross-tie

Which of the following actions will NOT maximize the available capacity of a limited crosstie source in a loss of all AC event on Unit 1 with only the 2A EDG available for the site?

a. Establishing Alternate SX cooling.

b. Stopping 2A AF pump

c. Throttling closed 1AF005E.

d. Stopping 1A AF pump.

Answer C **Exam Level** S **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

055 Station Blackout

EA1. Ability to operate and / or monitor the following as they apply to Station Blackout:

EA1.06 Restoration of power with one ED/G 4.1 4.5

Explanation of Answer Major 250V loads include emergency (DC) lube oil pumps for the main turbine and the turbine driven feed pump turbines. Venting the main generator is due to shutdown of the emergency seal oil pump which will allow the hydrogen to escape along the generator shaft unless the pressure is reduced by venting.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Loss of All AC Power	BCA-0.0	step 28, Attachment B Step 17	38,19 (78)	1A WOG-1 B	
BCA-0.0, 0.1, 0.2, 0.3 LOSS OF ALL AC POWER	BCA-0.0	28, Att B 17	38, 78	3	9
Chp 8b, 250 VDC	Chp 8b	IV.A.1	20	1	2

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

omment Type **Comment**

Question 90 Identification of RCP seal LOCA/cooldown

Select the primary reason for rapidly depressurizing the steam generators during a Loss of All AC.

- a. To provide maximum core cooling until power can be restored.
- b. To minimize RCS inventory loss from RCP seals.
- c. To enhance restoration of S/G level from the diesel driven AF pump.
- d. To increase subcooling of the RCS.

Answer b **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

055 Station Blackout

EK3. Knowledge of the reasons for the following responses as they apply to Station Blackout:

EK3.02 Actions contained in EOP for loss of offsite and onsite power 4.3 4.6

Explanation of Answer The rapid cooling allows depressurizing the RCS reducing the leak rate via the RCP seals

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
LOSS OF ALL AC POWER	BCA-0.0	CAUTION 2 step 10	70	1A WOG-1 B	
BCA-0.0, 0.1, 0.2, 0.3 LOSS OF ALL AC POWER	BCA-0.0	CAUTION	70	3	4, 5

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 91 Reset of sequencer

How would the sequencer operate if a Safety Injection (SI) actuation occurs while the sequencer is sequencing loads in response to an ESF bus undervoltage condition?

a. There will be no change in operation; the undervoltage sequence overrides the SI sequence.

b. The undervoltage sequencing stops, the sequencer immediately resets and SI loads NOT already running will sequentially start.

c. The undervoltage sequencing stops, all started loads are shed, and SI loads will sequentially start.

d. The undervoltage sequencing completes its cycle, then resets to SI mode, and SI loads NOT already running will sequentially start.

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 3 **SRO Group:** 3

056 Loss of Off-Site Power

AA1. Ability to operate and / or monitor the following as they apply to Loss of Off-Site Power:

AA1.21 Reset of the ESF load sequencers 3.3 3.3

Explanation of Answer The UV sequence is stopped and the SARA sequencing is initiated from step 1.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
D/G Relaying	DG-2	SARA & SSR schematics, On a SI Signal Coincident with Loss of Off-site Power		1	
hp 9, Diesel Generators & Aux. Systems	Chp 9	III.D.1	47-48	1	7.c

Material Required for Examination

Question Source: New **Question Modification Method:** Significantly Modified

Question Source Comments: Vogtle - 5/91

Comment Type **Comment**

Question 92 Eval of electric bus status

The following conditions exist on Unit 1:

- Bus 141 is powered from its normal source
- D/G 1A surveillance is being performed with the D/G paralleled to the bus

What would occur if a failure of the undervoltage relay results in a sensed undervoltage condition on Bus 141?

a. SAT feeder breaker ACB 1412 and D/G feeder breaker ACB 1413 remain closed. The Safe Shutdown loads will NOT sequence and CANNOT be manually started from the control room.

b. SAT feeder breaker ACB 1412 and D/G feeder breaker ACB 1413 will open. After a 10-second delay, ACB 1413 will close and the Safe Shutdown loads will sequence.

c. SAT feeder breaker ACB 1412 will open but D/G feeder breaker ACB 1413 will remain closed. The Safe Shutdown loads will sequence normally.

d. SAT feeder breaker ACB 1412 will open but D/G feeder breaker ACB 1413 will remain closed. The Safe Shutdown loads will NOT sequence and CANNOT be manually started from the control room.

Answer d **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 3 **SRO Group:** 3

056 Loss of Off-Site Power

AA2. Ability to determine and interpret the following as they apply to Loss of Off-Site Power:

AA2.46 That the ED/Gs have started automatically and that the bus tie breakers are closed 4.2 4.4

xplanation of Answer On sensed UV, the SAT feeder breaker opens (and alternate feeder breaker would also have opened if closed) and the control switches for the safe shutdown loads will be locked out due to constant undervoltage signal.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Chp 4, A.C. Electrical Power Distribution System	Chp 4	II.A.2.h.7).b)	62, 67	2	10.a, 11

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 93 Eqpt affected on bus loss

On Unit 1 power is lost to 120 VAC Instrument Bus 111

How are the ESF and Safe Shutdown loads affected?

a. "A" Train ESF loads will NOT load on an SI signal, but Safe Shutdown loads will load on a U/V signal

"B" Train loads are NOT affected.

b. A" Train ESF loads will load on an SI signal, but Safe Shutdown loads will NOT load on a U/V signal.
"B" Train loads are NOT affected.

c. "A" Train ESF loads will NOT load on an SI signal, and Safe Shutdown loads will NOT load on a U/V signal.
"B" Train loads are NOT affected.

d. "A" Train AND "B" Train ESF loads will NOT load on an SI signal, but Safe Shutdown loads will load on a U/V signal.

Answer C **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

057 Loss of Vital AC Instrument Bus

AA2. Ability to determine and interpret the following as they apply to Loss of Vital AC Instrument Bus:

AA2.19 The plant automatic actions that will occur on the loss of a vital ac electrical instrument bus 4.0 4.3

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
' oss of Instrument Bus	1BOA ELEC-2	Table A, 1.a	7	56	
.OA ELEC-2	BOA ELEC-2	Table A.a	7	2	3.c, 7
Chp 60a - SSPS	Chp 60a	II.B.5		1	4, 9.b

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 94 Operations required for transfer

Which of the following sets of indications are available on the Remote Shutdown Panel?

a. Emergency boration flow, S/G level, and RCS wide range temperature.

b. Red and green lights for reactor trip breaker position indication, S/G pressure, and pressurizer level.

c. Main feedwater flow, letdown flow, and charging line pressure.

d. Containment pressure, charging flow, and auxiliary feedwater flow.

Answer a **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

068 Control Room Evacuation

AA1. Ability to operate and / or monitor the following as they apply to Control Room Evacuation:

AA1.12 Auxiliary shutdown panel controls and indicators 4.4 4.4

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
RSP PL04/5J	PN-1	Diagram		2	
CONTROL ROOM INACCESSIBILITY	1BOA PRI-5	Attachment A, step 1	1 (40)	57B	
Chp 62, Remote Shutdown Panel	Chp 62	II.A.1.a, b	10	3	4

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments

Comment Type **Comment**

Question 95 Major action categories

When inadequate core cooling exists, which of the following sets of actions states the proper sequence of the major action categories to be performed in accordance with BFR-C.1, "RESPONSE TO INADEQUATE CORE COOLING", for removing decay heat from the core?

a. Rapid secondary depressurization; reinitiation of safety injection; RCP restart.

b. Reinitiation of safety injection; rapid secondary depressurization; RCP restart.

c. Rapid secondary depressurization; RCP restart ; reinitiation of safety injection.

d. RCP restart; rapid secondary depressurization; reinitiation of safety injection.

Answer b **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

074 Inadequate Core Cooling

EK1. Knowledge of the operational implications of the following concepts as they apply to Inadequate Core Cooling:

EK1.03 Processes for removing decay heat from the core 4.5 4.9

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Function Restoration Procedures BFR-C.1, C.2, C.3	BFR-C.1	C	1	3	2, 3

Material Required for Examination

Question Source: New **Question Modification Method:** Editorially Modified

Question Source Comments: VC Summer 5/94

Comment Type **Comment**

Question 96 Actions for reducing activity

High coolant activity has been detected and chemistry has determined that it is due to corrosion product activation.

Identify the effect of placing the cation demineralizer in service.

The cation demineralizer...

- a. will remove lithium so it should NOT be used in this condition.
- b. will cause the activity level to decrease as soon as it is placed in service.
- c. is NOT effective in removing corrosion product activity.
- d. is less effective than the mixed bed demineralizer so it is placed in service ONLY if decontamination factor is less than 10.

Answer b or A **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

076 High Reactor Coolant Activity

AA2. Ability to determine and interpret the following as they apply to High Reactor Coolant Activity:

AA2.02 Corrective actions required for high fission product activity in RCS 2.8 3.4

Explanation of Answer The cation demin is highly effective in removing corrosion products from the coolant.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Abnormal Operating Procedures BOA PRI-4, Abnormal Primary Chemistry	1BOA PRI-4	B	1	1	1
Chp 15a - Chemical and Volume Control System	Chp 15a	II.A.1.k.3)	27	02	4

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 97 Interlocks affecting reestablishment of feed

The following conditions exist on Unit 2:

- Reactor power was 8% prior to the event below.
- A failure in the feedwater control system caused ONE S/G level to rise to 83%.
- The main turbine tripped.
- S/G levels have returned to their normal level range
- The Startup FW Pump is running

What are all the conditions that would have to be met to feed the S/Gs using the FW034's Feedwater Tempering Flow Control valves?

a. The FW Isolation Aux Relays would have to be reset and FW035 Feedwater Tempering Isol valves opened.

b. The reactor trip breakers would have to be cycled, the FW Isolation Aux Relays would have to be reset and FW035 Feedwater Tempering Isol valves opened.

c. The FW Isolation Main Relays and Aux Relays would have to be reset and FW035 Feedwater Tempering Isol valves opened.

d. The reactor trip breakers would have to be cycled and FW Isolation Main Relays and Aux Relays reset and FW035 Feedwater Tempering Isol valves opened.

Answer a **Exam Level** B **Cognitive Level** Application **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

E05 Loss of Secondary Heat Sink

EK2. Knowledge of the interrelations between Loss of Secondary Heat Sink and the following:

EK2.1 Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. 3.7 3.9

Explanation of Answer The P-14 signal, once clear, only maintains FWI signal via the FW Isol Aux relays if NO reactor trip signal is present. So resetting the FW Isolation Aux relay allows opening of FW035s (normal feed path at low power) and throttling of FW034s

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
ESF Setpoints	EF-2	To reset a FW Isolation after a P-14		5	
Feedwater Simple / SGWLC	FW-1 / FW-2	FWI signals, Pump interlocks, Flowpaths during startup		0	
Chp 61,Engineered Safety Features	Chp 61	II.C.6.a		2	7.c

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 98 Identification of heat removal process

The following conditions exist on Unit 1:

- A leak developed on the RCS loop C flow instrument piping.
- Coincident with the RCS leak, on the reactor trip a S/G PORV failed open and was later isolated.
- FR-P.1 was entered to due to an ORANGE PATH condition.
- SI actuated and has been reset.
- All RCPs are stopped.
- Conditions required to support an RCP start are met.

Under the current conditions starting the RCP will...

a. cause excessive thermal stresses in the stagnant loops.

b. cause a pressure surge that will aggravate the PTS condition.

c. provide mixing of the ECCS injection flow thereby decreasing the likelihood of PTS.

d. increase the RCS cooldown rate thereby increase the likelihood of PTS.

Answer C **Exam Level** B **Cognitive Level** Comprehension **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

E08 Pressurized Thermal Shock

EK2. Knowledge of the interrelations between Pressurized Thermal Shock and the following:

EK2.2 Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility. 3.6 4.0

Explanation of nswer

Reference Title	Facility Reference Number	Section	Page Number(s)	Revision	L. O.
Function Restoration Procedures BFR-P.1, P.2	BFR-P.1	5	19	0	3, 5
Status Trees	ST-1	Integrity		1	

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 99 Natural Circ conditions and limits

The following conditions exist on Unit 1:

- A natural circulation is in progress per BEP ES-0.2 "Natural Circulation Cooldown"
- Pressurizer pressure is being controlled using Aux. Spray and Pzr heaters
- As pressure is being lowered through 1300 psig, a rapid increase is noted in Pzr level
- Charging and letdown are in manual and are balanced

What actions are required to be taken by the operators?

a. Repressurize the RCS.

b. Isolate the SI Accumulators.

c. Increase the RCS cooldown rate.

d. Place excess letdown in service.

Answer a **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

E09 Natural Circulation Operations

EK3. Knowledge of the reasons for the following responses as they apply to Natural Circulation Operations:

EK3.1 Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics. 3.3 3.6

Explanation of Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Emergency Procedures BEP-0 Reactor Trip or Safety Injection BEP ES-0.0 - 0.4	BEP ES-0.2 - 0.4	5, 12	3, 8	3	3, 4
Natural Circulation Cooldown	BEP ES-0.2	Step 14	10	1, WOG-1 B	

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Question 100 Reason for rapid S/G depressurization

Why are the S/Gs depressurized to less than 675 psig according to BCA-1.1, "Loss of Emergency Coolant Recirculation"?

a. To allow maximum AFW flow to the S/Gs.

b. To ensure adequate subcooling for restart of the RCPs.

c. To set up conditions for controlled injection to the RCS from the accumulators.

d. To decrease RCS temperature and pressure which reduces break flow in a LOCA condition.

Answer C **Exam Level** B **Cognitive Level** Memory **Facility:** Byron **ExamDate:** 9/14/98

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

E11 Loss of Emergency Coolant Recirculation

EA1. Ability to operate and / or monitor the following as they apply to Loss of Emergency Coolant Recirculation:

EA1.1 Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. 3.9 4.0

Explanation of Answer The concern is maximizing cooling volumes that supply water to RCS. By cooling RCS, depressurization of RCS can be initiated (while maintaining subcooling) to the point where the SI accumulators inject their volumes into the RCS.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Loss of Emer Coolant Recirc	BCA 1.1	28	24	1B WOG 1B	
BCA-Contingency Action 1.1 and 1.2	BCA-1.1	28	24	2	4

Material Required for Examination

Question Source: New **Question Modification Method:** Editorially Modified

Question Source Comments: South Texas 9/92

Comment Type **Comment**

GENERIC FUNDAMENTALS EXAMINATION
EQUATIONS AND CONVERSIONS HANDOUT SHEET

EQUATIONS

$\dot{Q} = \dot{m}C_p\Delta T$

$\dot{Q} = \dot{m}\Delta h$

$\dot{Q} = UA\Delta T$

$\dot{Q} \propto \dot{m}^3_{Nat\ Circ}$

$\Delta T \propto \dot{m}^2_{Nat\ Circ}$

$K_{eff} = 1/(1 - \rho)$

$\rho = (K_{eff} - 1)/K_{eff}$

$SUR = 26.06/\tau$

$$\tau = \frac{\overline{\beta} - \rho}{\lambda_{eff}\,\rho}$$

$$\rho = \frac{\ell^*}{\tau} + \frac{\overline{\beta}}{1 + \lambda_{eff}\tau}$$

$\ell^* = 1 \times 10^{-4}$ seconds

$\lambda_{eff} = 0.1$ seconds^{-1}

$P = P_o 10^{SUR(t)}$

$P = P_o e^{(t/\tau)}$

$A = A_o e^{-\lambda t}$

$CR_{S/D} = S/(1 - K_{eff})$

$CR_1(1 - K_{eff1}) = CR_2(1 - K_{eff2})$

$1/M = CR_1/CR_X$

$DRW \propto \phi^2_{tip}/\phi^2_{avg}$

$F = PA$

$\dot{m} = \rho A\hat{v}$

$\dot{W}_{Pump} = \dot{m}\Delta P \upsilon$

$E = IR$

Eff. = Net Work Out/Energy In

$$\upsilon(P_2 - P_1) + \frac{(\bar{v}_2^2 - \bar{v}_1^2)}{2g_c} + \frac{g(z_2 - z_1)}{g_c} = 0$$

$g_c = 32.2$ lbm-ft/lbf-sec^2

CONVERSIONS

---

1 Mw = 3.41×10^6 Btu/hr

1 hp = 2.54×10^3 Btu/hr

1 Btu = 778 ft-lbf

°C = (5/9)(°F - 32)

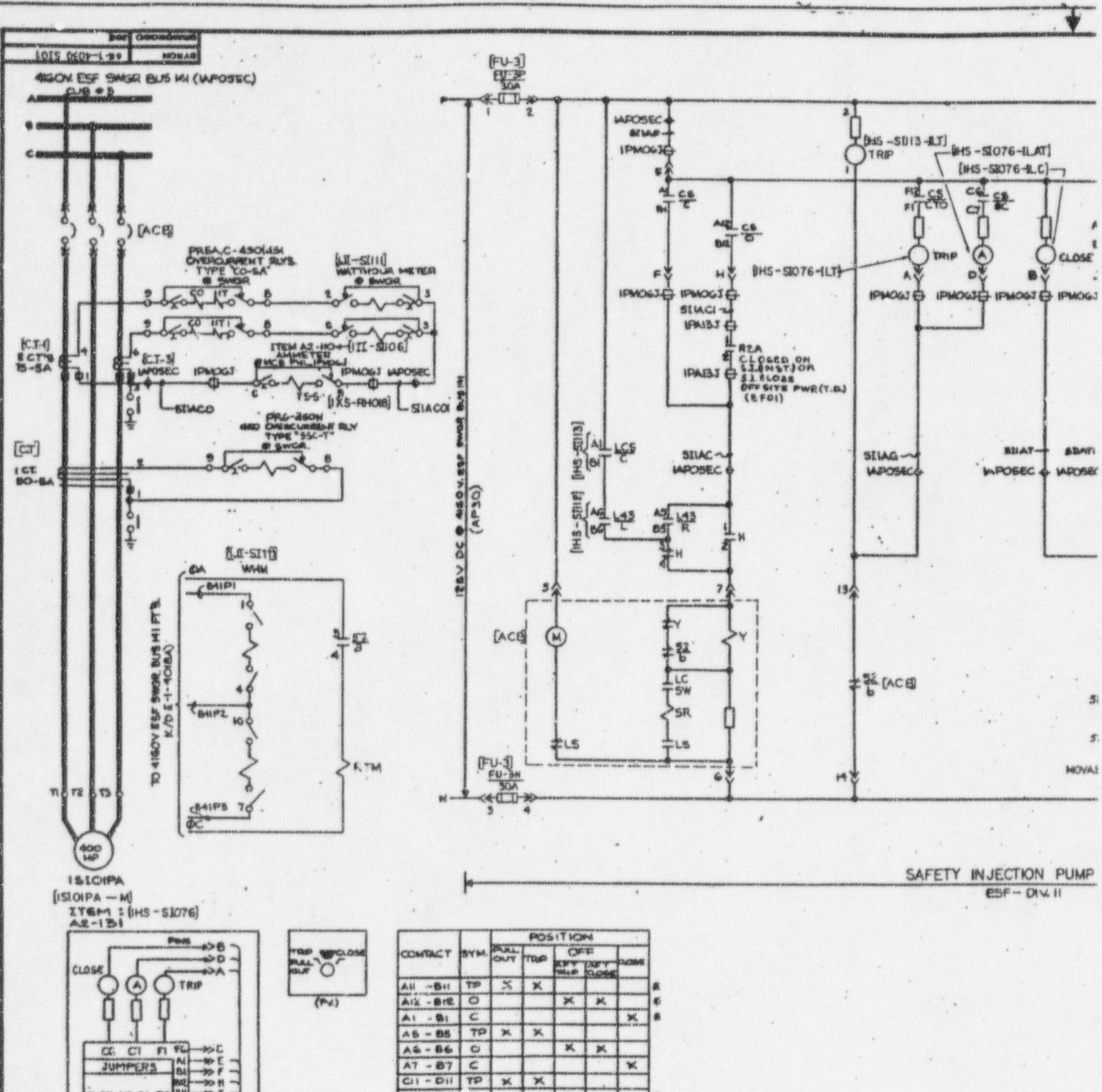
°F = (9/5)(°C) + 32

1 Curie = 3.7×10^{10} dps

1 kg = 2.21 lbm

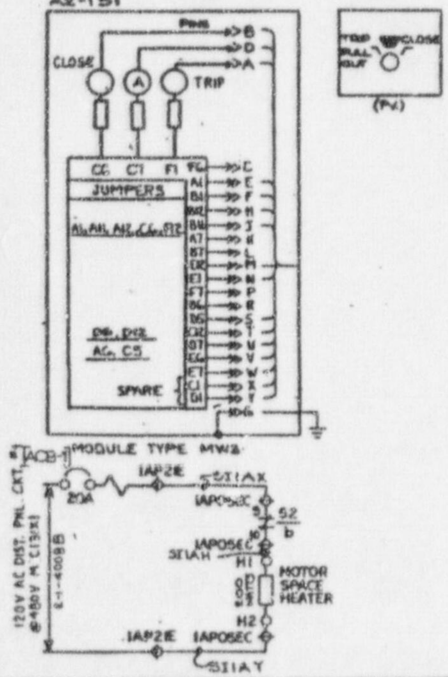
1 gal_{water} = 8.35 lbm

1 ft^3_{water} = 7.48 gal



SAFETY INJECTION PUMP

ESF – DIV. II

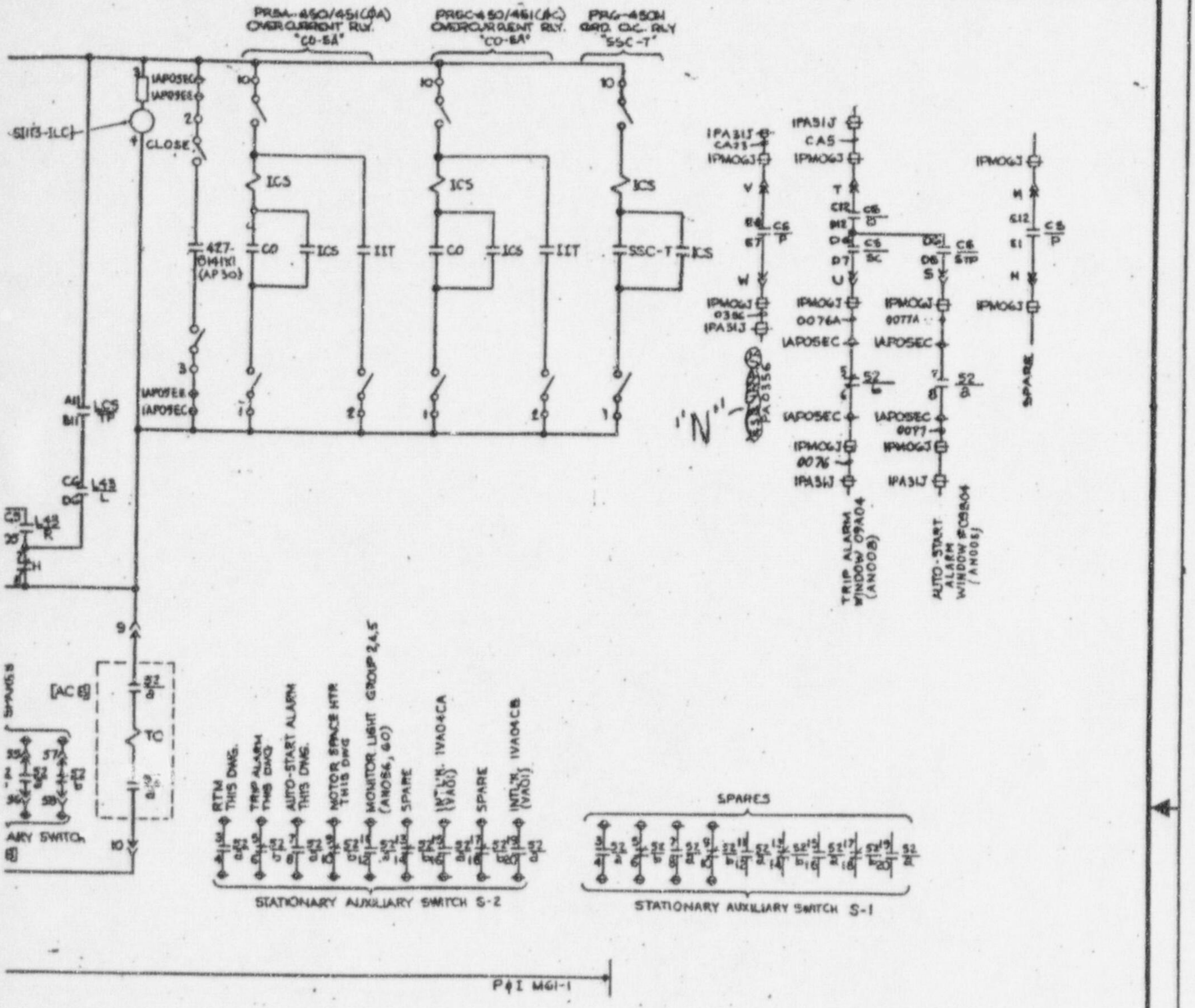


CONTACT	SYM.	POSITION					
		PULL OUT	TRIP	OFF		CLOSE	
				AFT TRIP	AFT CLOSE		
A11 - B11	TP	X	X				#
A12 - B12	O			X	X		#
A1 - B1	C					X	#
A5 - B5	TP	X	X				
A6 - B6	O			X	X		
A7 - B7	C					X	
C11 - D11	TP	X	X				
C12 - D12	O			X	X		#
C1 - D1	C					X	
C5 - C6	STP	X	X	X			
C6 - C7	SC				X	X	#
D5 - D6	STP	X	X	X			#
D6 - D7	SC				X	X	#
E12 - E1	P	X					
F12 - F1	CTO		X	X	X	X	#
E6 - E7	P	X					#
F6 - F7	CTO		X	X	X	X	

WESTINGHOUSE TYPE W-2 CONTROL SWITCH, 3 POSITIONS, SPRING RETURN TO CENTER, BASIC SWITCH S#5CMA264G02, PISTOL GRIP FIXED HANDLE

X - DENOTES CONTACT CLOSED
- DENOTES SWITCH CONTACT USED

CONTROL SWITCH DEVELOPMENT 'CS'
@ MLB PNL 1PM06J



B. G. TREECE 17830

NUCLEAR SAFETY RELATED
EQUIPMENT IS SHOWN ON THIS DRAWING.
REF. W.E.CO. DWG.271C620 SHT.26

SCHEMATIC DIAGRAM
SAFETY INJECTION PUMP 1A -
1SI01PA

BYRON/~~BRAIDWOOD~~ STATION —UNIT 1
COMMONWEALTH EDISON CO.
CHICAGO, ILLINOIS

SCALE NONE
DRAWN Louis J. Budzinski 6/26/74
CHECKED R P Herrmann 3-18-75
ENGINEER Alfred Hinal 3-18-75
APPROVED
JOB NO 4391

SARGENT & LUNDY
CHICAGO

DRAWING NO.
BYRON 6E-1-4030 SI01
BRAIDWOOD 6E-

REFERENCE DWGS.	
DWG. NO.	DESCRIPTION
E-1-4054 SERIES	INT/EXT W/D MCB ENG. SAFETY FEATURES 1PM06J
E-1-4122 SERIES	INT/EXT W/D ESF SEQUENCING AND ACTUATION CAB-TRAIN A 1PA13J
-1-4155 SERIES	INT/EXT W/D ANNUNCIATOR INPUT CAB. (ESF II) 1PA31J
-1-4611 SERIES	INT/EXT W/D 4160V ESF SWGR. BUS 141 1AP05E
E-1-4030 SERIES	S/D NOTES, LEGENDS, REFERENCE DRAWINGS
E-1-4661 SERIES	INT/EXT W/D 480V MCC 131X1 (1AP21E)
E-1-4665 SERIES	INT/EXT W/D 480V MCC 131X3 (1AP22E)

DRAWING RELEASE RECORD			
REV.	DATE	DESCRIPTION	DRAWN / CHECKED / ENGR. APPROVAL
M	6/16/87	ADDED COMPONENT TAGS REL. FOR RECORD ONLY	
N	9/1/87	FOR RECORD - INCORP. [illegible]	
L	1-8-87	ADDED COMPONENT TAGS REL. FOR RECORD ONLY	

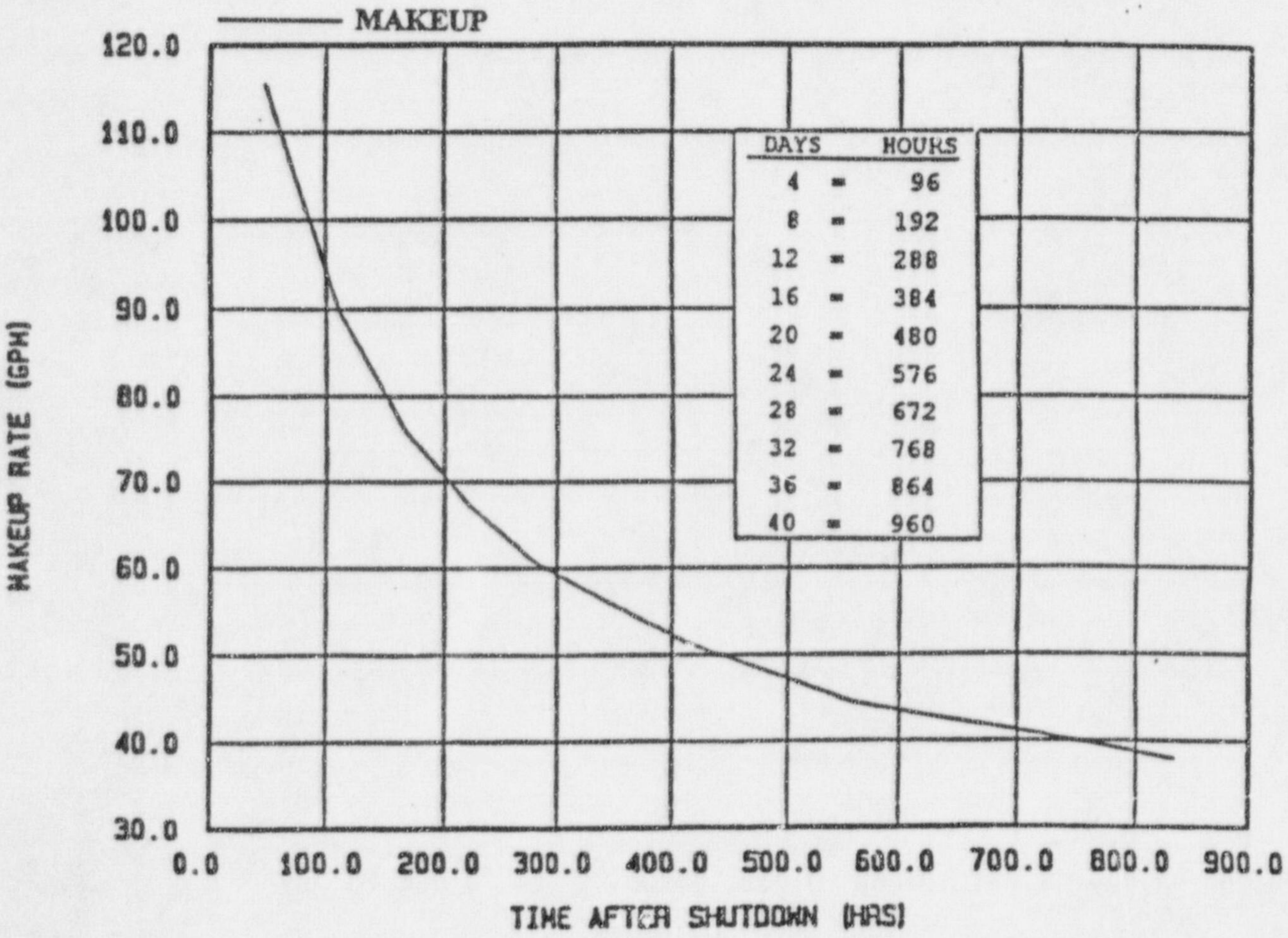
BYH-5236 (K), BBHE-10588 (H) BBHE-6006 (J) BYH-5568 (L) BYH-5613 (M) BBHE-8147 (G)

BBHE-7745 (F)

07

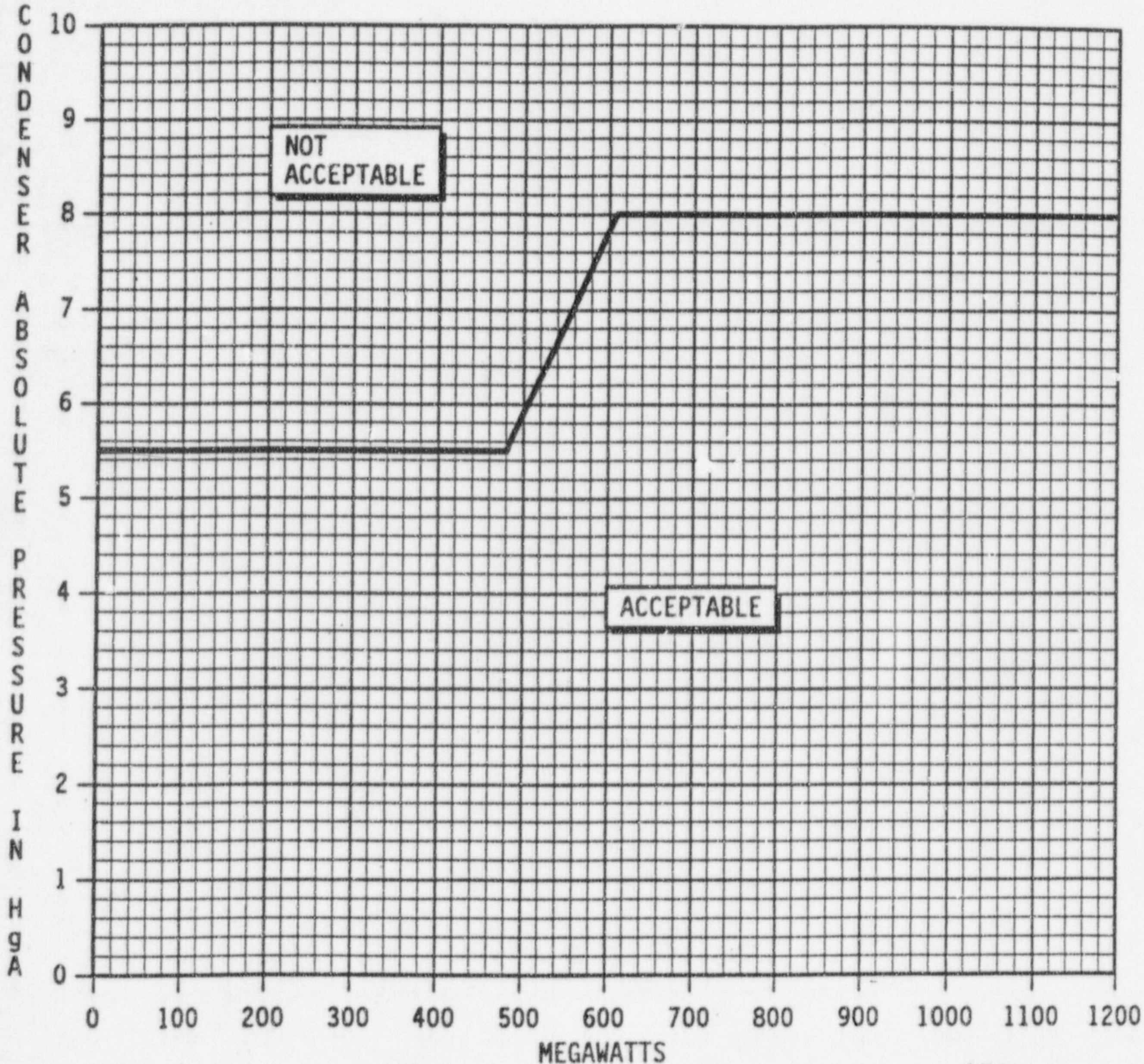
STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5	VERIFY ALL CONTROL RODS FULLY INSERTED:	Perform the following:
	a. All rod bottom lights - LIT	a. IF two or more rods are NOT fully inserted, THEN borate 1320 GAL (5500 GAL FROM RWST) for each rod NOT fully inserted per 1BOA PRI-2, EMERGENCY BORATION.
		b. Within 1 HOUR calculate Shutdown Margin per 1BOS 1.1.1.1.e-1, SHUTDOWN MARGIN SURVEILLANCE (ITS 1BOSR 3.1.1.1).

APPROVED
JAN 21 1998
B.O.S.R.



APPROVED
OCT 2 4 1997
B.O.S.R.

FIGURE 1BOA PRI 10-1
MINIMUM MAKEUP FLOW REQUIRED TO MATCH BOIL-OFF



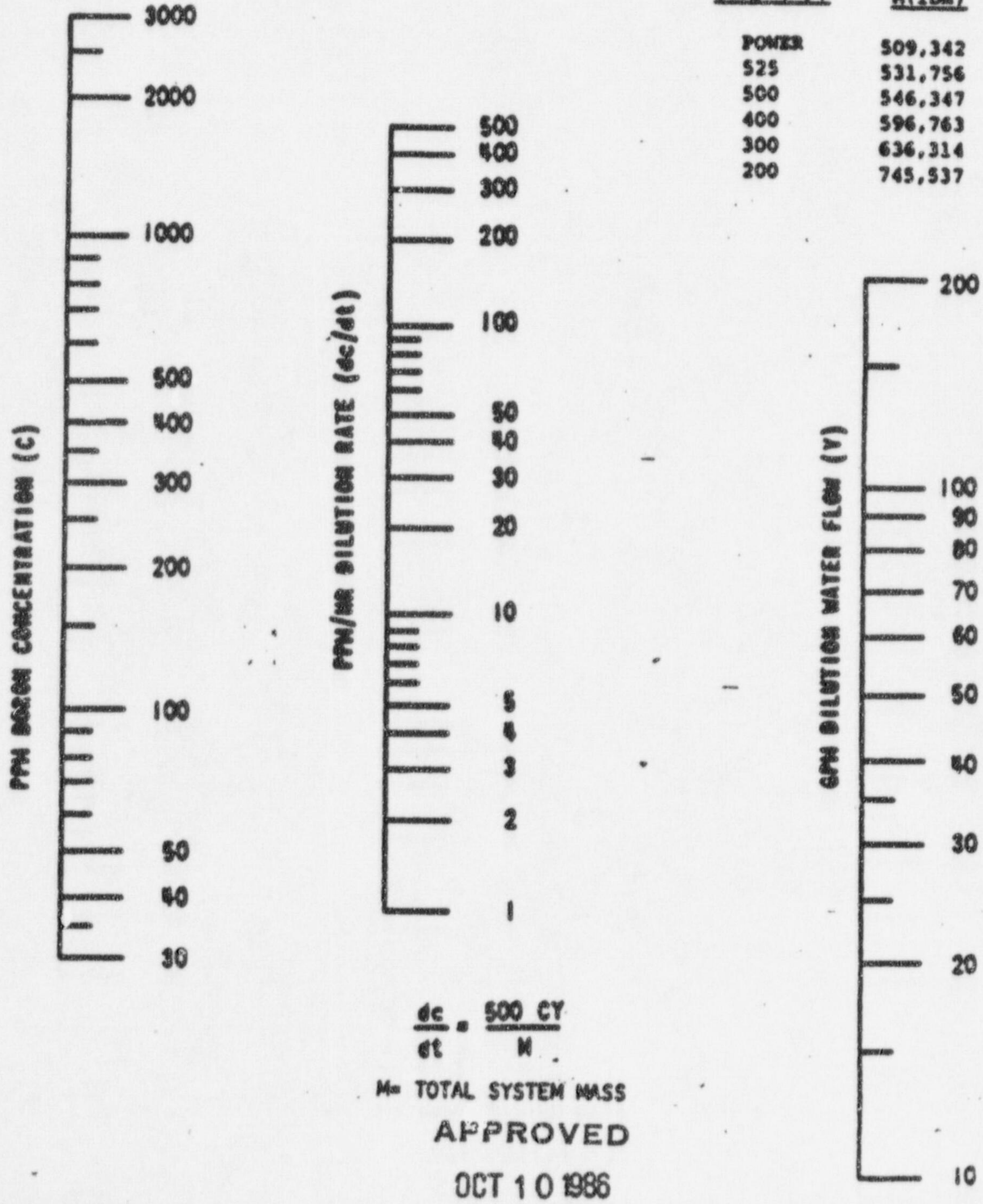
APPROVED
FEB 0 3 1994
B.O.S.R.

FIGURE 1BOA SEC-3-1
TURBINE LOAD -vs- CONDENSER PRESSURE

BCB-2
Figure 12
Revision 0

BORON DILUTION RATE NOMOGRAPH

TEMP (°F)	M(lbm)
POWER	509,342
525	531,756
500	546,347
400	596,763
300	636,314
200	745,537



$$\frac{dc}{dt} = \frac{500\ CY}{M}$$

M= TOTAL SYSTEM MASS

APPROVED
OCT 10 1986
B.O.S.R.

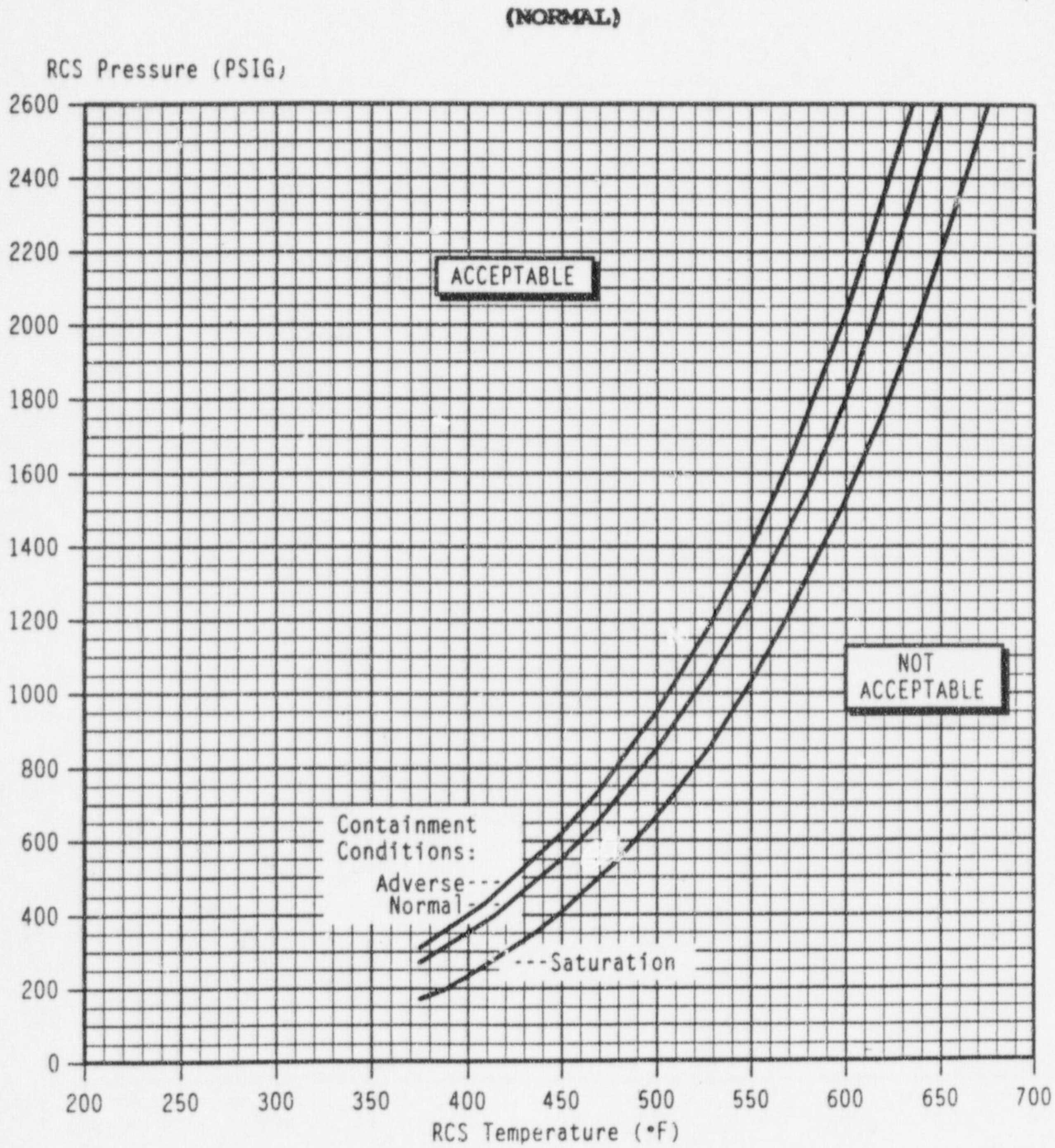


FIGURE 1BEP 1-1
RCS SUBCOOLING MARGIN

APPROVED
JAN 2 6 1998
B.O.S.R.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6	<u>CHECK CHANGE IN LEAK RATE SHUTDOWN REQUIREMENTS</u>:	
	a. Leak rate in - <u>ANY SG GREATER THAN 50 GPD</u>	a. **RETURN TO** procedure and step in effect.
	b. <u>TWO</u> consecutive leak rates - <u>AVAILABLE</u>	b. Continue with Step 7, <u>WHEN</u> <u>TWO</u> consecitive leak rates are available, <u>THEN</u> do Step 6c.
	c. Change in SG leak rate - <u>LESS THAN 25 GPD IN 1 HOUR</u>	c. **GO TO** Step 8 (Next Page).

<u>NOTE</u>

Trend leak rate during power reduction. Apply shorter shutdown time requirements if leakrate increases.

<u>NOTE</u>

Step 8 should be performed if rad monitors increase rapidly before subsequent samples can be obtained.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7	<u>CHECK LEAK RATE SHUTDOWN REQUIREMENTS</u>:	
	a. Leak rate in <u>EACH</u> SG - <u>LESS THAN 150 GPD</u>	a. Perform the following: 1) Be in MODE 3 within <u>10 HOURS</u>. 2) Perform a normal cooldown per the BGPs. 3) **GO TO** 1BGP 100-4, POWER DESCENSION.
	b. Continue to monitor leak rate	
	c. **RETURN TO** procedure and step in effect.	

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JAN 15 1998
B.O.S.R.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

8 **INITIATE UNIT SHUTDOWN**:

a. Refer to Tech Spec 3.4.6.2 (ITS: 3.4.13)

b. While continuing with this procedure, shutdown Unit to MODE 3 within 4 HOURS per BGPs

9 **IDENTIFY LEAKING SG**:

- o Decreasing feed flow with stable SG level in any SG
- o Unexpected rise in any SG narrow range level
- o High activity from any SG sample
- o Increasing trend on any Main Steamline radiation monitor:
  - o Main Steamline 1A:
    - o 1RT-AR022 Grid 1 4AA122
    - o 1RT-AR023 Grid 1 4AA123
  - o Main Steamline 1B:
    - o 1RT-AR022 Grid 1 4AB222
    - o 1RT-AR023 Grid 1 4AB223
  - o Main Steamline 1C:
    - o 1RT-AR022 Grid 1 4AC322
    - o 1RT-AR023 Grid 1 4AC323
  - o Main Steamline 1D:
    - o 1RT-AR022 Grid 1 4AD422
    - o 1RT-AR023 Grid 1 4AD423

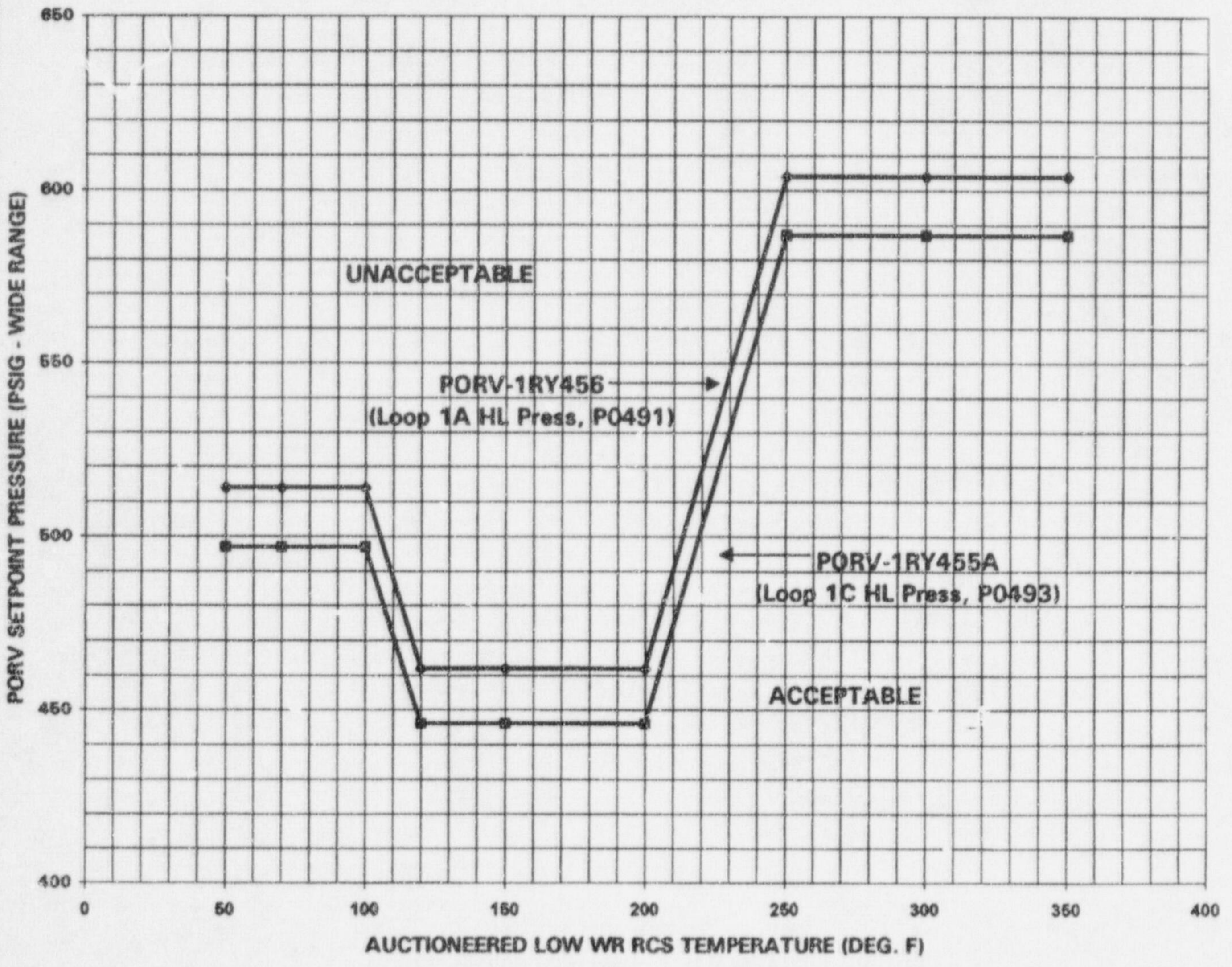
APPROVED
JAN 15 1998
B.O.S.R.

UNIT 1 (LTOPS)

LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM

PORV-1RY455A		PORV-1RY456	
AUCTION. LOW WR HL RCS TEMP. (DEG. F)	RCS WR PRESSURE (PSIG)	AUCTION. LOW WR CL RCS TEMP. (DEG. F)	RCS WR PRESSURE (PSIG)
50	497	50	514
70	497	70	514
100	497	100	514
120	446	120	462
150	446	150	462
200	446	200	462
250	587	250	604
300	587	300	604
350	587	350	604
450	2350	450	2350

(NOTE: To Determine RCS Pressure At RCS Temperatures Greater Than 350 F, Linear Interpolate Between 350 F and 450 F Data Points Shown Above.)



APPROVED

JAN 26 1998

REPORTING HAZARDOUS MATERIAL INCIDENTS

A. STATEMENT OF APPLICABILITY:

The purpose of this procedure is to implement current reporting and notification requirements for hazardous material incidents.

B. REFERENCES:

1. Byron Generating Station Emergency Plan (GSEP)
2. Title 10, Code of Federal Regulations, Part 20
3. Title 40, Code of Federal regulations, Part 302.4
4. Company Instruction 140, "PCB Management"
5. Hazardous Material (HAZMAT) Emergency Response Plan, Revision 1
6. Title 29, Code of Federal Regulations, Part 1910.120
7. RCRA Facility Plan for Byron Nuclear Power Station
8. Spill Prevention Control and Countermeasure Plan for Byron Station
9. BAP 1250-6, Reportable/Potentially Significant Event Screening and Notification
10. Reportability Manual
    a. Reportable Event SAF 4.1.
    b. Reportable Event ENV 2.1.
11. ESD Land Quality Compliance Manual

C. PREREQUISITES:

1. A HAZMAT reportable incident for radioactive materials is one in which there is ANY release exceeding Byron Station Technical Specifications.

> **NOTE**
>
> Radionuclides are listed as hazardous material with RQ units in curies in 40 CFR Table 302.4 Appendix B.

2. The Shift Manager will be provided with a list of individuals that are qualified to perform roles in a HAZMAT Emergency Event. This list will also be included in the Byron Station Pre-Incident Plan for HAZMAT Emergency Response. Positions listed will include:

    a. HAZMAT Emergency Coordinator

    b. HAZMAT Specialist(s)

    c. On-Scene Incident Commander(s)

    d. Safety and Health Advisor

3. Prior to performing ANY off-site notifications as directed in this procedure, the following individuals should be contacted:

    a. Station Duty Officer

    b. Station Manager

    c. Nuclear Duty Officer

    d. NRC Resident

D. PRECAUTIONS:

1. Rescue operations and evacuation of personnel to prevent injury shall be the primary concern in the event of a hazardous material event.

2. Release of any petroleium product into a public waterway is considered a reportable release into the environment.

E. LIMITATIONS AND ACTIONS:

1. Environmental Services (ESD) will perform any additional follow-up reporting that may be required.

F. MAIN BODY:

NOTE

If any notifications have previously been satisfied per BZP 310-5, Acting Station Director or Station Director, a follow-up call should be made which includes information requested on BAP 3000-16T1, Hazardous Material Spill Report Log.

1. The Shift Manager, or designee, with the assistance from the Rad Waste Foreman and/or a qualified HAZMAT specialist shall DETERMINE the reportability of any hazardous material or petroleum product release off-site or spill on-site.

NOTE

Refer to BAP 1250-6, Reportable/Potentially Significant Event Screening and Notification, and the Reportability Manual Reportable Events SAF 4.1 and ENV 2.1 for notification requirements to the NRC Operations Center.

The incident is reportable by this procedure if one or more of the following situations occur:

a. Any petroleum product or hazardous material spill condition that meets ALL the criteria listed below:

1). The material spilled is equal to or greater than the Reportable Quantity (RQ) as listed in Table 1 of this procedure or 40 CFR Table 302.4.

2). The spill occurs OUTSIDE a building (except for PCB releases in which case ANY release must be reported).

3). The material is not restricted to a container or structure designed to contain it.

4). The material is not treated by the Wastewater Treatment System and not discharged in accordance with the Byron Station NPDES Permit.

CAUTION

Release of any petroleum product into a public waterway is considered a reportable release into the environment.

b. Any amount of petroleum product is released into a public waterway (Rock River).

c. Any incident that has been classified as a HAZMAT Emergency Event as documented on BAP 3000-15T1, Hazardous Material Risk Evaluation.

d. Any PCB spill, indoor or outdoor, with PCB content of >500 ppm.

NOTE

Transformer insulating oil and bushings used at Byron Station have been tested and certified to contain <500 ppm PCB and are labeled as such.

e. Any release of radioactive materials exceeding Byron Station Technical Specification.

NOTE

Table 1 of this procedure lists the REPORTABLE QUANTITIES of the most likely spilled hazardous materials for Byron Station.

2. Upon determination that a hazardous material incident is reportable, the Shift Manager, or designee, should prepare for notifications by completing sections 1 through 8 of BAP 3000-16T1, Hazardous Material Spill Report Log.

    a. Provide name of individual who will serve as the on-site contact and the outside telephone number to Byron Station plus extension of the on-site contact.

        1). The HAZMAT Emergency Coordinator should be the on-site contact for spills classified as a HAZMAT Emergency Event.

        2). The NPDES Coordinator should be the on-site contact for events that affect the station's NPDES Permit.

    b. Provide the location of the spill. Include:

        1). Byron Nuclear Power Station
        4450 North German Church Road
        Byron, IL. 61010

        2). Ogle County

        3). Any body of water that is or may be affected (Rock River)

   c. DATE and TIME of the spill or release

   d. Extent of INJURIES. If any

   e. TYPE of material and QUANTITY involved

   f. CAUSE of the spill, if known

   g. Extent of contamination (brief description of area affected)

   h. RESPONSE and PRECAUTIONARY measures being taken

3. NOTIFY the HAZMAT Emergency Coordinator or the on-call HAZMAT Specialist if not already done so during initial response activities as directed in BAP 3000-15, Hazardous Material and Chemical Spill Response, Section F.5.

4. NOTIFY the following of this procedure of the incident and off-site notification requirements:

   a. Station Duty Officer

   b. Station Manager

   c. Nuclear Duty Officer

   d. NRC Resident

   e. NRC Operations Center

5. NOTIFY Environmental Services Department (ESD).

   a. Phone numbers to reach Environmental Services are:

      1). Normal Business Hours: 8-4463 or 8-4464.

      2). Backshifts and Weekends (Mobilecomm Pager):

         a). 1-312-712-0638 (Listen for beep tone)

         b). Touchtone cap code number 04629 (Listen for ring and 3 beeps)

         c). Touchtone the phone number for ESD Duty person to call, not to exceed 10 digits - (815) 234-5441

         d). Touchtone the # button and wait for busy signal before hanging up.

> **NOTE**
>
> Environmental Services Department will perform the remaining notifications to the appropriate off-site agencies.

3). Document the NAME of the ESD individual contacted and the TIME and DATE on BAP 3000-16T1, Hazardous Material Spill Report Log.

4). If ESD does not return page within 30 minutes, proceed with the required notifications in Step F.6 of this procedure.

6. NOTIFY the following off-site agencies and document the NAME of the individual contacted, TIME, DATE and REPORT NUMBER the agency assigns on BAP 3000-16T1, Hazardous Material Spill Report Log.

a. NATIONAL RESPONSE CENTER:
PHONE: 1-800-424-8802

b. ILLINOIS EMERGENCY SERVICES AND DISASTER AGENCY (IESDA):
PHONE: 1-800-782-7860

c. OGLE COUNTY EMERGENCY SERVICES AND DISASTER AGENCY (OC ESDA)
PHONE: 732-3201 (Manned by the Ogle County Sheriff)

1). If unable to reach OGLE COUNTY ESDA contact:
BYRON FIRE DEPARTMENT
PHONE: 234-4911

d. For a PCB release or spill only, also notify:
UNITED STATES ENVIRONMENTAL PROTECTION AGENCY (USEPA)
REGION V
PHONE: 1-312-353-2318

7. FORWARD the completed Hazardous Material Spill Report Log, BAP 3000-16T1, to the HAZMAT Emergency Coordinator.

G. CHECKOFF LISTS:

1. BAP 3000-16T1, Hazardous Material Spill Report Log

2. BAP 3000-15T1, Hazardous Material Risk Evaluation

TABLE 1

REPORTABLE QUANTITIES FOR SPILLED HAZARDOUS MATERIALS AND PETROLEUM PRODUCTS USED AT BYRON STATION

Spills equal to or greater than the amounts listed below must be reported to the National Response Center. This report is required by the CERCLA/SUPERFUND Regulations listed in 40 CFR 302.4. The volumes listed below are based on the concentrations normally found at Byron Station. If a spilled substance is not on this list or is of a different concentration, contact the HAZMAT Emergency Coordinator and provide the concentration of the spilled substance so that a calculation can be made to determine reportability.

To be reportable, the amount spilled in a 24 hour period or less must be equal to or greater than the amount listed as REPORTABLE QUANTITY, and must meet conditions listed in Section F.1. of this procedure.

MATERIAL	COMMON NAME	% CONC AT STATION	REPORTABLE QUANTITY
Acetone	Acetone	100 %	758 gal.
Ammonium Hydroxide	Ammonium Water	30 %	455 gal.
Asbestos, friable	Asbestos	N/A	1 #
Calcium Hypochlorite	Sanuril	65 %	15 #
Chlorofluorocarbons	Freon	100 %	400 gal.
Ethylene Glycol	Anti-Freeze	100 %	15 gal.
Hydrazine	Hydrazine	35 %	16 oz.
Hydrochloric Acid	Muratic Acid	30 %	1738 gal.
Mercury	Quicksilver	100 %	1 oz.
Methanol	Synasol	100 %	758 gal.
Methyl Ethyl Ketone	MEK	100 %	745 gal.
Mineral Spirits	Stoddard Solvent Petroleum Naptha	100 %	15 gal.
Morpholine	Morpholine	22 %	400 gal.
Di-Octyl Phthalate	Capacitor Fluid Dielectrol II Selectrol I	100 %	500 gal.
1,2,4-Trichloro Benzene	Capacitor Fluid Dielectrol II Selectrol I	100 %	8 gal.
Polychlorinated Biphenyl	PCB Aroclor Askerl		1/2 gal.

TABLE 1
(Continued)

MATERIAL	COMMON NAME	% CONC AT STATION	REPORTABLE QUANTITY
Paint, Containing 15 % Butyl Acetate 20 % Xylene			500 gal.
Sodium Hydroxide	Caustic Soda	32 %	278 gal.
Sodium Hydroxide	Caustic Soda	50 %	157 gal.
Sodium Hydroxide	NALCO STABREX ST 70	1 - 5 %	2700 gal.
Sodium Hypochlorite	Bleach	12.5 %	78 gal.
Sodium Nitrite	NALCO 8338/1359+	20/40 %	38 gal.
Sodium Nitrite	LCS-20 / LCS-60	10/20 %	120 gal.
Styrene Monomer	Styrene	100 %	132 gal.
Sulfuric Acid	Sulfuric Acid	93 %	66 gal.
Toluene	Toluene	100 %	140 gal.
1,1,1 Trichloroethane	Edison Solvent	100 %	91 gal.
Xylene	Xylene	100 %	140 gal.
Zinc Chloride	NALCO 1360/1360T	10/20 %	500 gal.
Petroleum Products	Turbine Oil		15 gal.
	Crankcase Oil		15 gal.
	Fuel Oil		15 gal.
	Diesel Fuel		15 gal.
	Mineral Oil		15 gal.
	Gasoline		15 gal.
	Kerosene		15 gal.
Betz Powerline 3610	Power Line 3610		15 gal.

THE FOLLOWING IS A LIST OF OTHER WATER TREATMENT CHEMICALS USED THAT ARE NOT REPORTABLE:

NALCO	1250	Carbohydrazide (Oxygen Scavenger)
	1338	Sodium Bromide
	1340	Sure Cool Dispersant
	1383-T	Sodium Phosphonate
	1385	Dyna Cool Scale Inhibitor
	7338	Gluteraldahyde
	7349	Nalsperse Biodispersant
	8103	Organic Polymer, Coagulant
	8182	Acrylic Polymer
	8306	Potassium Phosphate
	8307	Phosperse - plus Inhibitor
	9383	Scale Inhibitor
	94UF193	Methoxypropylamine (MPA)

(Final)