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Docket No.: 50-412

NOV 14 1984

MEMORANDUM FOR: Thomas M. Novak, Assistant Director for Licensing, Division of Licensing

FROM: Daniel R. Muller, Assistant Director for Radiation Protection, Division of Systems Integration

SUBJECT: BEAVER VALLEY UNIT 2 SER INPUT FROM THE ACCIDENT EVALUATION BRANCH (AEB)

Plant Name: Beaver Valley Unit 2
Licensing Stage: OL
Responsible Branch: Licensing Branch No. 3; M. Ley, LPM
Target Date: SER Input-10/26/84
Review Branch: AEB
Review Status: Review Continuing

Enclosed (Enclosure 1) is the SER input for Beaver Valley Unit 2 from the Accident Evaluation Branch. Several open and confirmatory items are identified. There is a confirmatory item in Section 6.4, Control Room Habitability, relating to control room operator doses following design basis accidents. In addition, the applicant has not demonstrated the capability to mitigate steam generator tube rupture (SGTR) events (review question 450.10) and this is considered an open issue. Section 15.4.3 reflects the SGTR issue and references the Reactor Systems Branch input (the P.M. will need to supply the appropriate cross reference in this SER input). Further, the radiological consequences of the LOCA and rod ejection accidents exceed the SRP guideline values. The applicant will have to take appropriate measures for these accidents to reduce the dose estimates to less than the acceptance criteria of the Standard Review Plans. AEB has communicated with the applicant on several occasions to resolve these issues.

As I have previously communicated to you, we do not recommend going forward to ACRS until resolution of at least the LOCA evaluation is assured.

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| SURNAME | | | | | | | |
| DATE | | | | | | | |

The SALP input is provided as Enclosure 2.

Questions regarding this review can be directed to Ken Dempsey, x28941.

Original signed by
Daniel R. Muller

Daniel R. Muller, Assistant Director
for Radiation Protection
Division of Systems Integration

Enclosures:
As stated

cc w/encl.:
R. M. Bernero
G. Knighton
W. Gammill
L. Hulman

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| OFFICE | DSI:AEB | DSI:AEB | DSI:AEB | DSI:AD/RP | | | |
| SURNAME | KDempsey | TQuay TRQ | LGHulman | DMuller DRM | | | |
| DATE | 11/6/84 | 11/8/84 | 11/8/84 | 11/14/84 | | | |

## 6.4 Control Room Habitability

The criteria for the protection of the control room personnel under accident conditions are specified in General Design Criterion (GDC) 19. The applicant proposes to meet these requirements by incorporating shielding, emergency HVAC system, and self-contained breathing apparatus in the control room habitability design. The habitability design also includes storage for food and water, sanitary facilities, fire protection, and a remote shutdown capability. The technical support center (TSC), which serves to back up the control room functions during emergencies, is discussed in Section 12.3.

The design of other aspects of control room habitability systems are discussed in separate SER sections as indicated:

a. Explosion, fire and toxic gas in vicinity of plant - Sections 2.2.1-2.2.3;

b. Protection from wind and tornado effects - Section 3.3;

c. Flood design - Section 3.4;

d. Missile protection - Section 3.5;

e. Protection against the dynamic effects associated with postulated rupture of piping - Section 3.6;

f. Environmental qualification of equipment - Section 3.11;

g. Filter efficiency - Section 6.5.1;

h. Shielding; and TSC - Section 12.3;

i. HVAC systems analysis - Section 9.4.1;

j. Fire protection and remote shutdown capability - Section 9.5.1; and

k. Human engineering, control room envi onment, and communications - Section 18.

The Beaver Valley control room HVAC system is designed to automatically isolate upon receipt of a contaiment isolation signal or detection of chlorine in the outside air intake. These signals also initiate the bottled air supply system, which is capable of maintaining the control room at 1/8 inch water gauge positive pressure, or greater, for approximately 1 hour. Following the first hour, the control room envelope may be isolated in the event of a toxic gas release, or pressurized by the emergency outside air filtration system in the event of a radiation accident.

The staff review of the Beaver Valley control room emergency HVAC system indicates an area, relating to control room operator thyroid dose following design basis accidents, that is unclear and needs to be addressed further by the applicant. The area in question is the lack of radiation detection capability at the outside air intake. The detectors may be necessary to initiate control room emergency systems, and maintain control room operator doses to within GDC-19 guidelines, in the event of accidents occurring outside the containment of Unit 2, such as steamline breaks and fuel handling accidents. The applicant has agreed to provide additional analysis demonstrating whether or not hardware modifications are needed and to make corrections if necessary. The staff considers this a confirmatory item.

Based upon satisfactory resolution of the confirmatory issue, the applicant will demonstrate that the control room habitability systems will adequately protect the control room operators in accordance with the requirements of 10 CFR Part 50, Appendix A, GDC 19. NUREG-0737, a summary of post TMI Action Items, contains requirements for control room habitability in Item III.D.3.4 that state the Standard Review Plan (SRP-NUREG-0800) should be used to assess control room habitability. By meeting the SRP, and referenced regulatory guides, the applicant has met the requirements of NUREG-0737, Item III.D.3.4.

6.5.2 Fission Product Removal and Control System

The quench spray system (QSS) provides for removal of certain post-accident fission products and depressurization of containment in the event of a LOCA. The system is redundant and covers about 78 percent of the containment volume. Following an appropriate accident signal (in less than 90 seconds following a LOCA initiation), the QSS is designed to initiate automatically; and the containment spray injection mode is automatically switched to the recirculation mode when a low level is reached in the reactor water storage tank (RWST).

Immediately following QSS initiation a NaOH solution from the chemical addition tank is added to the quench spray water, thereby improving the effectiveness for iodine removal during an accident by raising the pH of the spray to between 9.1 and 10.3. The pH of the sump water after caustic (NaOH) addition would be above 8.5. The minimum flow rate through the nozzles (SPRAYCO Model 1713A) with one of two trains operating is 2,950 gpm, which corresponds to an elemental iodine removal coefficient of 14.5 $hr^{-1}$. In accordance with the SRP, the staff has, however, limited the removal rate to 10 $hr^{-1}$ in its analysis for the duration of the accident. An iodine decontamination factor, defined as the ratio of iodine initially in the containment to that at a later time, of approximately 25 was estimated at the end of one hour, the time at which the containment was assumed to become subatmospheric. The assumptions used in the staff's LOCA analysis are shown in Table 15.4.2.

The staff evaluation of the containment spray system has demonstrated that the Beaver Valley containment spray system design meets the requirements of General Design Criterion (GDC) 41, "Containment Atmosphere Cleanup;" GDC 42, "Inspection of Containment Atmosphere Cleanup Systems;" and GDC 43, "Testing of Containment Atmosphere Cleanup System," of 10 CFR Part 50, Appendix A.

15.4 Radiological Consequences of Design Basis Accidents

The postulated design-basis accidents (DBAs) analyzed by the applicant to determine the effectiveness of Engineered Safety Features (ESFs) in mitigating the offsite radiological consequences are the same as those analyzed for previously licensed PWRs. To evaluate the effectiveness of the certain mitigative ESFs proposed for the Beaver Valley Power Station Unit 2, and to ensure that the radiological consequences of these accidents meet the applicable dose criteria, the staff has analyzed design basis loss-of-coolant (LOCA), fuel-handling, steamline break, steam generator tube rupture (SGTR), small line break and control rod ejection accidents. These DBAs were evaluated using the applicable SRP Sections and Regulatory Guides. The results of the DBAs evaluated by the staff are presented in Table 15.1. The data and assumptions used are listed in Tables 15.2-15.6.

The bases for estimated X/Q values (atmospheric dispersion estimated) are discussed in SER Section 2.3.4.

15.4.1 Loss-of-Coolant Accident (LOCA)

The applicant has analyzed a hypothetical design-basis LOCA and concluded that the combination of ESFs and distances to the exclusion area boundary (EAB) and to the outer boundary of the low population zone (LPZ) are sufficient to provide reasonable assurance that the radiological consequences of such an accident are within guidelines set forth in 10 CFR 100.11 (a)(1) and (2). The analysis included the following sources and radioactivity transport paths to the atmosphere:

(1) contribution from containment leakage; and

(2) contribution from post-LOCA leakage from ESF systems outside containment.

The staff has reviewed the applicant's provisions for and design of the containment system and the containment spray system as described in Sections 3 and 6 of this report. The staff independent analysis of the radiological consequences of a hypothetical design-basis LOCA is described below.

15.4.1.1 Containment Leakage Contribution

The staff's calculation of the consequences of a hypothetical LOCA used the conservative assumptions of Positions C.1.a through C.1.e of Regulatory Guide 1.4, Revision 2, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors." The primary containment was assumed to leak at a rate of 0.1 percent per day for the first hour and, because the containment pressure would become subatmospheric within one hour, the leak rate was assumed to be zero percent per day after one hour. The fraction of core inventory available for release from the containment was assumed to be 25 percent for iodine and 100 percent for noble gases. The analysis took into account radiological decay during holdup in the containment and iodine removal by the sprays. Although the ESF areas are aligned and exhausted by the supplementary leak collection and release system (SLCRS) immediately following a LOCA, no credit was taken for filtration. A list of assumptions used in the calculation of the LOCA doses is given in Table 15.2.

15.4.1.2 Post-LOCA Leakage from ESF Systems Outside Containment

As part of the LOCA analysis, the staff has also evaluated the consequences of leakage of recirculated sump water because in the recirculation mode of operation, the sump water is circulated outside containment to the auxiliary building. If a leak should develop, such as a pump seal failure, a fraction of the iodine in the water could become airborne in the auxiliary building and exit to the atmosphere. Because the Beaver Valley Unit 2 ECCS area in the auxiliary building is served by the SLCRS, doses from passive failures were not considered (as specified in SRP Section 15.6.5, Appendix B).

In FSAR Table 15.6-9, the applicant has identified a value of $9.4 \times 10^{-3}$ gpm as the expected amount of leakage from the ECCS equipment following an accident. Using the guidance of Appendix B of SRP Section 15.6.5, the staff evaluated the potential radiological consequences from this release pathway assuming a routine leakage rate of twice the applicant's value ($1.9 \times 10^{-2}$ gpm). The staff will review the Beaver Valley Unit 2 Technical Specifications relative to the testing of ESF systems recirculating sump water outside containment to assure that the leakage outside containment for all these systems is maintained less than $9.4 \times 10^{-3}$.

15.4.1.3 Conclusions

The staff's calculated thyroid dose from a hypothetical LOCA exceeds the 10 CFR Part 100 guideline values for the exclusion area boundary. The applicant will have to take appropriate measure(s) to reduce this value to within 10 CFR Part 100 guideline values. When this open item is resolved, additional discussion will be provided.

15.4.2 Main Steamline Break Outside Containment

The staff and the applicant have both evaluated the radiological consequences of a postulated steamline break accident occurring outside containment and upstream of the main steam isolation valve. During the course of the accident, the shell side of the affected steam generator is assumed to stay dry since auxiliary feedwater flow to the affected steam generator would be blocked off under accident conditions. Using this assumed dry condition, all the iodine transported by the primary to secondary leakage (1 gpm) is assumed available to be released directly to the atmosphere. Although the contents of the secondary side of the affected steam generator would be vented initially to the atmosphere as an elevated release, the staff conservatively assumed that the entire release for the duration of the accident occurs at ground level.

The staff investigated three cases in accordance with the guidance of SRP Section 15.1.5, Appendix A. For Case 1, assuming a stuck rod, the applicant projected 1% fuel cladding failure. The staff, however, used a value consistent with that previously used on Westinghouse plants of 5% fuel cladding failures. For Case 2, that of a preaccident iodine spike, the staff assumed that previous reactor operation has resulted in a primary coolant concentration equal to a standard technical specification iodine spike limit of 60 μCi/gm DEI-131. For Case 3, that of an accident induced iodine spike, the staff assumed that an iodine spike occurs as a result of the accident, and that the iodine release rate from the fuel to the primary coolant following the accident is increased by a factor of 500. Furthermore, prior to the accident, the plant was assumed to be operating at a Technical Specification primary coolant equilibrium activity limit of 1.0 μCi/gm DEI-131. Finally, the staff assumed that plant cooldown took 8 hours and then the Decay Heat Removal system was initiated.

The staff assumptions are presented in Table 15.3 and calculated doses are presented in Table 15.1. The calculated doses are within the guideline values of SRP Section 15.1.5, Appendix A, and the staff finds the design to mitigate the consequences of a main steamline break outside containment acceptable.

15.4.3 Steam Generator Tube Rupture

The applicant has provided an analysis of the systems response and radiological consequences of a steam generator tube rupture (SGTR) accident. This analysis is based upon the ability to isolate the affected steam generator within 30 minutes. The staff has requested justification that the operator can take appropriate action within 30 minutes. The staff has also expressed concerns to the applicant regarding those systems in which his analysis takes credit in mitigating the consequences of a SGTR. In response, the applicant states that the Westinghouse Owners Group is investigating several SGTR licensing concerns and will address the staff's concerns through a generic resolution at a future date. Upon receipt of this additional information, the staff will complete the review of this open item and report the radiological consequences in a supplement to this SER. Section 15.___ of this Report provides additional discussion of the systems aspect of this accident.

15.4.4 Control Rod Ejection Accident

For this accident, a mechanical failure of the control rod drive mechanism is postulated. As a result of the failure, the reactor coolant system pressure would eject the rod cluster control assembly, drive the shaft to the fully withdrawn position, and primary coolant would leak to the containment. The consequence of this mechanical failure is a rapid positive reactivity insertion and a primary system depressurization. This would lead to an adverse core power distribution and localized fuel damage.

In analyzing this accident, the staff has used a conservative value of 10% fuel cladding failures and 0.25% fuel melting which is based on previous experience with similar PWRs. The consequences were evaluated using the guidance of SRP Section 15.4.8, Appendix A, and Regulatory Guide 1.77. The staff calculated doses via two release pathways:

through containment leakage and through the secondary system. In the case of the containment leakage pathway, the ejected rod assembly is assumed to puncture the control rod drive mechanism housing, with the activity released through containment leakage to the environment. The containment leakage is assumed to cease after one hour, the time it takes the containment to return to subatmospheric pressure. In the second case, all the released activity is assumed to be mixed with the primary coolant, with some of the activity transported to the secondary side of the steam generators through steam generator tube leaks. Since loss of offsite power is assumed, the activity would be released to the environment through a steam dump to the atmosphere.

The assumptions used to determine the consequences of this accident are presented in Table 15.4. The estimated 0-2 hour thyroid dose at the exclusion area boundary for the containment leakage pathway exceeds the guideline value of SRP Section 15.4.8, Appendix A and is considered an open item. The applicant will have to take appropriate measures to reduce the dose to within guideline values.

15.4.5 Fuel-handling Accident

For the analysis of a fuel-handling accident in the fuel pool, the staff assumed that a fuel assembly was dropped in the fuel pool during refueling operations and that all of the fuel rods in the dropped assembly were damaged, plus fifty rods in the second impacted assembly (as conservatively proposed by the applicant), thereby releasing the volatile fission gases from the fuel rod gaps into the pool. The fuel building exhaust system and filters, which are part of the SLCRS, would be in operation during fuel-handling, and the radioactive materials that escaped from the fuel pool were assumed to be released to the environment in a puff release with the iodine activity reduced by filtration through the SLCRS. The estimated offsite radiological consequences following the postulated accident are given in Table 15.1. The calculated doses are within the guidelines of SRP Section 15.7.4. The assumptions and parameters used in the analysis are given in Table 15.5. The dose model and dose conversion factors employed in the analysis were the same as those given in Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

The staff also evaluated the consequences of a fuel-handling accident inside containment. The applicant states that the time required for air to travel from the radiation monitor to the first containment isolation valve is greater than the closure time of the containment isolation valves plus the detector response time. Therefore, the design capability for rapid isolation of the containment provides assurance that virtually all the radioactive releases from such an accident would be contained in the primary containment, and no doses need to be reported in this SER.

The staff finds the design to mitigate the consequences of fuel-handling accidents acceptable.

### 15.4.6 Failure of a Small Line Carrying Primary Coolant Outside Containment

The applicant has provided an analysis of an accidental break in the CVCS letdown line outside containment, but downstream of the containment isolation valves. The applicant has postulated that the most severe pipe rupture with regard to radiological consequences outside containment would be a complete severance of a 2-inch letdown line in the Chemical Volume and Control System. The staff concurs in this assumption. This break would release up to 160 gpm of primary coolant to the auxiliary building, providing a release pathway to the environment. The time required for the operator to identify the accident and isolate the rupture is expected to be less than 15 minutes. This value is consistent with Draft Standard ANSI N660, "Time Response Design Criteria for Safety-Related Operator Actions." Indications, such as letdown line pressure downstream of the postulated break location and volume control tank level, will allow early detection of the failure by the operator.

Based on a 15 minute isolation time, a total of 18,000 lbm of primary coolant could be released. The staff estimates that 40 percent of the hot reactor coolant would flash into steam upon entering the auxiliary building atmosphere, and assumed an equal fraction of the dissolved iodine fission products would become airborne. The staff conservatively assumed that the airborne iodine and dissolved noble gases can escape directly to the environment at ground level, without decay or filtration. Other assumptions are given in Table 15.7. The radiological consequences for this postulated DBA are provided in Table 15.1. The estimated doses are a small fraction (not more than 10%) of the 10 CFR Part 100 exposure guidelines, are in accordance with Regulatory Guide 11.1 and, therefore, meets the acceptance criteria of Standard Review Plan 15.6.2. The staff finds the applicant's design for mitigating the radiological consequences of a failure of small lines carrying primary coolant outside containment acceptable.

15.4.7 Spent Fuel Cask Drop Accident

The applicant has stated that the spent fuel cask will not be lifted more than 30 feet above any surface during the entire transfer operation under normal operating conditions. Based on this commitment, no radiological release is anticipated from such a drop and, therefore, no doses need to be evaluated in accordance with the acceptance criteria of SRP Section 15.7.5.

Table 15.1

Radiological Consequences of Design Basis Accidents

| Postulated Accident | Exclusion Area Boundary, Rem Thyroid | Exclusion Area Boundary, Rem Whole Body | Low Population Zone, Rem Thyroid | Low Population Zone, Rem Whole Body |
|---|---|---|---|---|
| Loss of coolant:** | | | | |
| Containment Leakage 0-1 hour* | | | | |
| Total Containment Leakage | | | | |
| ECCS Component Leakage | | | | |
| Tota' | | | | |
| Steam Line break outside secondary containment: | | | | |
| w/5% fuel cladding failure (Case 1) | 114 | 5.5 | 14 | 0.8 |
| w/pre-accident iodine spike (Case 2) | 40 | 5.5 | 3.9 | 0.8 |
| w/concomitant iodine spike (Case 3) | 27 | 5.5 | 2.6 | 0.8 |
| Control rod ejction | | | | |
| Containment leakage pathway* | | | | |
| Secondary system release pathway | 36 | 4.0 | 4.6 | 0.3 |
| Fuel handling accident | | | | |
| In fuel handling area | 30 | 2.1 | 1.0 | 0.1 |
| Small line break | 21 | 0.1 | 0.7 | 0.3 |
| Steam generator tube rupture** | | | | |
| Case 1 (DEI-131 at 60 μCi/gm) | | | | |
| Case 2 (DEI-131 at 1 μCi/gm) | | | | |

* Containment leakage occurs for only one hour--the time it takes the containment to become subatmospheric

** Open items

Table 15.2

Assumptions Used in the Calculation of LOCA Doses

Containment Leakage

| | |
|---|---|
| Power level (MWt) | 2766 |
| Operating time, years | 3 |
| Fraction for core inventory available for containment leakage, % | |
| iodine | 25 |
| noble gases | 100 |
| Initial iodine composition in containment, % | |
| elemental | 91 |
| organic | 4 |
| particulate | 5 |
| Containment leak rate, %/day | |
| 0-1 hour | 0.1 |
| after 1 hour | 0 |
| Containment volume, $ft^3$ | |
| sprayed volume | $1.4 \times 10^6$ |
| unsprayed volume | $4.0 \times 10^5$ |
| Containment mixing rate ($hr^{-1}$) | 2 |
| Containment spray system | |
| maximum allowable elemental iodine decontamination factor | 100*** |
| Spray removal coefficients, $hr^{-1}$ | |
| elemental iodine | 10 |
| particulate iodine | 0.65 |
| organic iodine | 0 |
| Relative concentration values (X/Q), $sec/m^3$ | |
| 0-2 hours at the exclusion area boundary* | $1.9 \times 10^{-3}$ |
| 0-8 hours at the low population zone boundary** | $8.8 \times 10^{-5}$ |
| 8-24 hours at the low population zone boundary** | $6.4 \times 10^{-5}$ |
| 24-96 hours at the low population zone boundary** | $3.2 \times 10^{-5}$ |
| 96-720 hours at the low population zone boundary** | $1.2 \times 10^{-5}$ |

* Exclusion area boundary = 527 meters

** Low population zone boundary = 5800 meters

*** Decontamination factor is approximately 25 at the time the containment becomes subatmospheric

Table 15.2

Assumptions Used in the Calculation of LOCA Doses (continued)

ECCS Leakage Outside Containment

| | |
|---|---|
| Power, MWt | 2766 |
| Sump volume, gal | $8.3 \times 10^5$ |
| Flash fraction | 0.1 |
| Leak rate (twice the maximum operational leakage), gpm | 0.019 |
| Leak duration, hr | 720 |
| Delay time, hr | 0.08 |
| Filter efficiency, % | 95 |

Table 15.3

Assumptions Used to Evaluate the Radiological Consequences Following a Postulated Main Steamline Break Accident Outside Containment

| | |
|---|---|
| Power, MWt | 2766 |
| Failed fuel fraction (Case 1), % | 5 |
| Preaccident dose-equivalent I-131 in primary coolant (Case 2), μCi/gm | 60 |
| Preaccident dose-equivalent I-131 in primary coolant (Case 3), μCi/gm | 1 |
| Primary to secondary leak rate, gpm | 1 |
| All of the 1 gpm leak occurs in the affected steam generator | |
| All the iodine transported to the shell side of the steam generator by the leakage is lost to the environment without decay. | |
| Iodine release rate from the fuel increases by a factor of 500 as a result of the accident (Case 3) | |
| Duration of the accident, hrs | 8 |

Table 15.4

Assumptions for Rod Ejection Accident

| | |
|---|---|
| Power level, MWt | 2766 |
| Failed fuel, % | 10 |
| Melted fuel, % | 0.25 |
| Iodine partition factor | 100 |
| Steam generator tube leak rate, gpm | 1 |
| Containment leakage, %/day | 0.1 |
| Time containment goes subatmospheric after accident, hrs | 1 |
| Time at which the primary and secondary pressures equalize, hrs | 8 |
| Peaking factor | 1.65 |

Table 15.5

Assumptions Used for Estimating the Radiological Consequences Following a Postulated Fuel Handling Accident

| | |
|---|---|
| Power level, MWt | 2766 |
| Number of fuel rods damaged | 314 |
| Total number of fuel rods in core | 41,448 |
| Radial peaking factor of damaged rods | 1.65 |
| Shutdown time, hours | 100 |
| Inventory released from damaged rods, % Iodines and noble gases | 10 |
| Pool decontamination factors | |
| Iodines | 100 |
| Noble gases | 1 |
| Iodine fractions released from pool, % | |
| Elemental | 75 |
| Organic | 25 |
| Iodine removal efficiencies for fuel building exhaust system, % | |
| Elemental | 95 |
| Organic | 95 |

Table 15.6

Assumptions Used in Accidents Involving Small Line Breaks Outside Containment

| | |
|---|---|
| Coolant released, lbs mass | 18,000 |
| Fraction of coolant released flashed to steam % | 40 |
| Primary coolant concentration, microcuries/gm dose-equivalent I-131 | 1 |
| Spiking factor (iodine release rate multiplier) | 500 |
| Letdown rate, gpm | 120 |
| Primary coolant volume, $ft^3$ | 8,416 |

ENCLOSURE 2

SALP INPUT FROM THE ACCIDENT EVALUATION BRANCH FOR BEAVER VALLEY UNIT 2
SER

A. Licensing Activities

1. Management Involvement in Assuring Quality

Rating: 2
Reviews were generally timely, thorough, and technically sound. Records are generally complete, well maintained and available.

2. Approach to Resolution of Technical Issues from a Safety Standpoint

Rating: 2
NRC effort neeeded to obtain some acceptable solutions. Responses for the most part were viable and sound.

3. Responsive to NRC Initiatives

Rating: 3
There are a few longstanding regulatory issues attributable to the licensee.

4. Staffing (including Management)

Rating: N/A

5. Reporting and Analysis of Reportable Events

Rating: N/A

6. Training and Qualification Effectiveness

Rating: N/A

7. Overall Rating for Licensing Activity Functional Area: 2

Average