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TECHNICAL EVALUATION REPORT

CONTROL OF HEAVY LOADS — PHASE II

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR POWER STATION UNITS 1, 2, AND 3

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FOREWORD

This Technical Evaluation Report was prepared by Franklin Research Center under a contract with the U.S. Nuclear Regulatory Commission (Office of Nuclear Reactor Regulation, Division of Operating Reactors) for technical assistance in support of NRC operating reactor licensing actions. The technical evaluation was conducted in accordance with criteria established by the NRC.

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1. INTRODUCTION

1.1 PURPOSE

This technical evaluation report documents a review of load handling equipment operated in the vicinity of spent fuel and equipment employed for reactor shutdown and fuel element decay heat removal at Browns Ferry Nuclear Power Station Units 1, 2, and 3. This review constitutes the second phase of a two-phase review instituted to resolve a generic issue pertaining to the safe handling of heavy loads at nuclear power plants.

1.2 GENERIC BACKGROUND

Generic Technical Activity Task A-36 was established by the Nuclear Regulatory Commission (NRC) staff to systematically examine staff licensing criteria and the adequacy of measures in effect at operating nuclear power plants to ensure the safe handling of heavy loads and to recommend necessary changes in these measures. This activity was initiated by a letter issued by the NRC staff on May 17, 1978 [1] to all power reactor licensees, requesting information concerning the control of heavy loads near spent fuel.

The results of Task A-36 were reported in NUREG-0612 [2]. The staff concluded from this evaluation that existing measures to control the handling of heavy loads at operating plants provide protection from certain potential problems but do not adequately cover the major causes of load handling accidents and should be upgraded.

To upgrade measures for the control of heavy loads, the staff developed a series of guidelines to implement a two-part objective. The first part of the objective, to be achieved through the implementation of a set of general guidelines expressed in NUREG-0612, Section 5.1.1, was to ensure that all load handling systems at nuclear power plants have been designed and are operated so that their probability of failure is appropriately small for the critical tasks in which they are employed. The results of the reviews associated with this part of the staff's overall objective were provided in a series of technical evaluation reports identified as Phase I reports. The second part

of the staff's objective, and the subject of this report, was to be achieved through guidelines expressed in NUREG-0612, Sections 5.1.2 through 5.1.5. The purpose of these guidelines was to ensure that, in the case of specific load handling systems used in areas where their failure might result in significant consequences, either (1) features have been provided, in addition to those required for all load handling systems, to make the potential for a damaging load drop extremely small or (2) conservative evaluations of load handling accidents indicate that the potential consequences of a load drop are acceptably small.

1.3 PLANT-SPECIFIC BACKGROUND

On December 22, 1980, the NRC issued a letter [3] to the Tennessee Valley Authority (TVA), the Licensee for Browns Ferry Units 1, 2, and 3, requesting the review of provisions for handling and control of heavy loads, the evaluation of these provisions with respect to the guidelines of NUREG-0612, and the provision of certain additional information to be used for an independent determination of conformance to these guidelines. The results of this independent evaluation with respect to general load handling equipment and procedures (Phase I) were provided on April 5, 1983 [4]. On September 28, 1982, TVA provided an initial Phase II report [5] concerning conformance with staff guidelines for specific load handling systems operated in areas where a load drop might result in significant consequences. On January 25, 1983 [6] and on March 28, 1983 [7], TVA supplemented that initial response. The information in References 5, 6, and 7 provided the basis for this technical evaluation report.

2. EVALUATION

This section presents an evaluation of critical load handling areas at Browns Ferry Units 1, 2, and 3. Separate subsections are provided to identify the criteria used in this evaluation and each of the plant areas considered. For each such area, relevant load handling systems are identified, Licensee-provided information related to the evaluation criteria or proposed alternatives is summarized and evaluated, and a conclusion as to the extent of compliance, including recommended additional action or requirements for additional information, as appropriate, is provided.

2.1 EVALUATION CRITERIA

The objective of this review was to determine if plant arrangements and load handling equipment design were such that either the likelihood of a load handling accident that could damage spent fuel or equipment used in reactor shutdown or fuel element decay heat removal is extremely small or that the consequences of such damage, should it occur, will be acceptable. Guidance contained in NUREG-0612, Sections 5.1.2, 5.1.3, and 5.1.5 (for pressurized water reactors) and in 5.1.4 and 5.1.5 (for boiling water reactors) forms the basis for the conclusions reached in this section and is briefly summarized as follows.

For a determination that the likelihood of damage is extremely small:

- The design of the load handling system (i.e., crane or hoist and underhook lifting devices) is consistent with, or equivalent to, the NRC staff criteria for single-failure-proof cranes identified in NUREG-0554 [8], or
- The plant physical arrangement is such that a crane operated in the vicinity of spent fuel or safety-related equipment is prevented from traveling to a position from which a load drop can be expected to damage such equipment.

For a determination that the potential consequences of damage following a load drop will be acceptable:

- In the case of potential damage to spent fuel, calculations have been provided to demonstrate that potential radiological doses at

the site boundary will not exceed 25% of the limits specified in 10CFR100 and that the post-accident configuration of the fuel will not result in a k_{eff} larger than 0.95.

- In the case of damage to the reactor vessel or spent fuel pool, it can be demonstrated that this damage will be limited to the extent that the fuel will not become uncovered.

- In the case of damage to equipment or components employed for reactor shutdown or fuel element decay heat removal, it can be demonstrated that the safety-related function of the affected system will not be lost consequent to a load drop.

2.2 REACTOR BUILDING HANDLING SYSTEMS

2.2.1 Load Handling Systems Capable of Carrying a Heavy Load (as defined in NUREG-0612) Within the Reactor Building

2.2.1.1 Summary of Licensee Statements and Conclusions

In Reference 5, the Licensee identified the reactor building crane and a 4-ton hook-type chain hoist as the load handling systems within the reactor building capable of carrying a heavy load. In Reference 6, the Licensee identified the 3-ton jib crane as being capable of swinging into critical electrical panels. In Reference 7, the Licensee revised the response it provided in a June 3, 1982 letter [9] to include a 24-ton gear-type chain hoist and indicated that this hoist is used to replace recirculation pump motors in the drywell.

2.2.1.2 Evaluation and Conclusion

The Licensee's statements and conclusions with respect to the load handling systems mentioned above are consistent with the information evaluated in Reference 4 and with the intent of NUREG-0612.

2.2.2 Reactor Building Crane

2.2.2.1 Summary of Licensee Statements and Conclusions

In Reference 5, the Licensee stated that the reactor building crane had been evaluated as having sufficient design features to make the likelihood of

a load drop extremely small. The Licensee provided an evaluation of the 125-ton reactor building crane with respect to the features of design, fabrication, inspection, testing, and operation as delineated in NUREG-0554 and supplemented by the identified alternatives specified in NUREG-0612 Appendix C. The Licensee also noted that this crane has previously been evaluated with respect to NRC Regulatory Guide 1.104 and Branch Technical Position APCSB9-1. The Licensee indicated that the evaluation of the reactor building crane design was based upon information provided in letters dated February 10, 1981 [10] and June 30, 1976 [11].

The Licensee also provided a heavy load/impact area matrix for the postulated drop of heavy loads from the reactor building crane within the reactor building.

2.2.2.2 Evaluation

The Licensee's evaluation of the 125-ton reactor building crane with respect to the features of design, fabrication, inspection, testing, and operation as delineated in NUREG-0554 and supplemented by alternatives specified in NUREG-0612, Appendix C was evaluated in a March 31, 1983 report [12]. The evaluation in Reference 12 included additional information provided by the Licensee in letters dated March 11, 1981 [13], May 27, 1981 [14], and December 14, 1982 [15]. In Reference 12, a point-by-point evaluation was completed using Licensee-supplied information. The overall conclusion drawn from that review was that TVA had satisifed the intent of the staff guidance by providing a reactor building crane which provides suitable protection from the effects of operator error and the failure of crane mechanical and electrical components. The evaluation of specific staff requirements indicated that in each issue associated with crane reliability, the Browns Ferry crane satisfied the staff requirements either directly or through features that can be assessed to be technically equivalent, or TVA had made a commitment to implement modifications which will satisfy the requirements, or TVA had made commitment to implement a test and inspection program to provide appropriate additional confidence in load handling reliability. Subsequent to the evaluation previously discussed, however, the NRC issued generic letter

83-42 [16], which provides an additional evaluation criterion not specifically stated in NUREG-0554. This letter notes that it will be the staff's policy to require a demonstration that no single failure in the crane electric power/control system will cause a load drop. This issue has not been addressed in available TVA submittals.

As indicated in Section 2.1 and NUREG-0612, Section 5.1.4, it is the staff's position that if the load handling system is to be considered highly reliable (i.e., it can be established that the likelihood of a load drop is extremely small), both the crane, underhook lifting devices, and attachment points should conform to the guidance of NUREG-0612, Section 5.1.6. The Licensee has not addressed the design capability of lifting devices or attachments points. As of this review, only the spent fuel shipping cask at the Browns Ferry plant has been reported to be of redundant design, including two lower load block attaching points. Other lifting devices and attachment schemes, which handle other critical loads (defined as loads which, if dropped, could result in an offsite dose in excess of 1/4 of 10CFR100 limits, or damage to fuel or fuel storage racks such that the average k_{eff} is greater than 0.95, or damage to the reactor vessel or spent fuel pool sufficient to cause water leakage to the extent that fuel could be uncovered, or damage to the equipment in safety systems sufficient to cause a loss of safe shutdown function), have not been addressed by the Licensee.

As an alternative to providing a single-failure-proof protection through lifting device design, NUREG-0612, Section 5.1.4 indicates that the effects of heavy load drops in the reactor building can be analyzed to show that the evaluation criteria of Section 5.1 are satisfied (i.e., demonstrate that certain heavy loads are not critical loads). The Licensee has provided a heavy load/impact matrix for various heavy loads carried by the reactor building crane. The heavy loads identified in this matrix are the same as those identified in Reference 4 with the following exceptions. The equipment pool shield plugs (50 tons), loaded skip box (1.25 tons), equipment pool covers (1.5 tons), motor generator sets (7.5 tons), surge tank plug (2.1 tons), reactor water cleanup (RWCU) demineralizer vault plugs and vessel head (6 tons), and the control rod rack (0.5 ton) have not been included in the

Licensee's heavy load/impact matrix. For the heavy loads included in the load/impact matrix, the Licensee has indicated that the likelihood of the reactor building crane load handling system failure is extremely small for each hazard; however, the Licensee has not indicated that analysis demonstrates that the load handling system failure (i.e., failure of lifting device) will result in consequences that satisfy the evaluation criteria of Section 5.1.

2.2.2.3 Conclusions and Recommendations

The Licensee has demonstrated, with the exception of specific consideration of single failures in the electric power/control system, that the reactor building crane has sufficient design features to make the likelihood of a load drop extremely small. The Licensee has not, however, demonstrated that the associated lifting devices comply with the guidelines of NUREG-0612, Section 5.1.6. As a consequence, the Licensee has not demonstrated that the reactor building crane load handling system meets the requirements of NUREG-0612, Section 5.1.4.

To demonstrate compliance with NUREG-0612, Section 5.1.4, the Licensee should address how the additional guidelines of Section 5.1.6(1) and 5.1.6(3) have been invoked at the Browns Ferry plant. For those lifting devices and interfacing lift points that do not satisfy these additional guidelines, the Licensee should analyze the effects of heavy load drops to show that the evaluation criteria of Section 5.1 are satisfied. In addition, the Licensee should ensure that all heavy loads handled by the reactor building crane are considered or should provide justification for exclusion of any heavy load previously identified.

2.2.3 <u>3-Ton Jib Crane</u>

2.2.3.1 Summary of Licensee Statements and Conclusions

In Reference 9, the Licensee exempted the 3-ton jib crane because the loads carried by this load handling system weigh less than 1000 lb. During an October 28, 1982 conference call with TVA, the NRC expressed a concern that

the 3-ton jib crane has a capacity to carry loads greater than this weight over Class IE equipment within its area of coverage. Following the conference call, the Licensee stated in Reference 6 that a design change request would be initiated by June 1, 1983 requiring a stop be installed to prevent the 3-ton jib crane from swinging into the critical electrical panels.

2.2.3.2 Evaluation

In Reference 4, the Licensee's commitment to install a stop to prevent movement of a heavy load over or into Class IE equipment was considered sufficient to exclude the load handling system from further evaluation with respect to NUREG-0612, Section 5.1.1. To ensure that the evaluation criteria of NUREG-0612, Section 5.1 are satisfied for heavy load drops within the reactor building, additional guidelines were provided in Section 5.1.4. Reliance on plant physical arrangement to prevent a load handling system in the vicinity of safe shutdown equipment from traveling to a position from which a load drop can be expected to damage such equipment is a means of ensuring that the likelihood of damage is extremely small. The Licensee's commitment to install a permanent stop to prevent the 3-ton jib crane from swinging into Class IE equipment is consistent with the intent of NUREG-0612, Section 5.1.4.

2.2.3.3 Conclusion and Recommendations

The Licensee has demonstrated that the 3-ton jib crane load handling system meets the guidelines of NUREG-0612, Section 5.1.4.

2.2.4 <u>24-Ton Gear-Type Chain Hoist</u>

2.2.4.1 Summary of Licensee Statements and Conclusions

In Reference 9, the Licensee described this load handling device as a "twenty-four-ton capacity, low headroom, spur geared type chain hoist w/geared trolley and chain container." The Licensee indicated that this hoist is used to replace recirculation pump motors (20 tons each) in the drywell.

2.2.4.2 Evaluation

In Reference 5, the Licensee described the load handling system and indicated how compliance with the guidelines of NUREG-0612, Section 5.1.1 was to be accomplished. The purpose of Reference 9 was to inform the NRC that TVA had identified additional load handling systems which had not been included in its initial response concerning NUREG-0612, Section 5.1.1. However, the Licensee's response in Reference 9 did not address how the additional guidelines of NUREG-0612, Section 5.1.4 were to be satisfied.

2.2.4.3 Conclusion and Recommendations

The Licensee has not demonstrated that the 24-ton gear-type chain hoist load handling system meets the guidelines of NUREG-0612, Section 5.1.4. To demonstrate compliance with these guidelines, the Licensee should revise its response provided in Reference 5 to include the 24-ton gear-type chain hoist load handling system.

2.2.5 4-Ton Hook Type Hoist

2.2.5.1 Summary of Licensee Statements and Conclusions

In Reference 5, the Licensee has stated that the 4-ton hook-type chain hoist does not have sufficient design features to make the likelihood of a load drop extremely small. Because the 4-ton hook-type chain hoist has the physical capability of carrying loads over safe shutdown equipment, the Licensee provided a heavy load/impact matrix for the various heavy loads handled and described the consequences. For each postulated load drop and impact area, the Licensee stated that system redundancy and separation ensure that the safety-related function of the affected system is not precluded following the postulated load drops. The Licensee further explained that the load drops are postulated in separate pump rooms containing core spray pumps A and C or pumps B and D. Because of physical separation of the two loops, a load drop in one room would not adversely affect the other redundant system, provided that hand control valve HCV-75-1 is closed during repair on pumps A and C, and valve HCV-75-29 is closed during repair on pumps B and D.

2.2.5.2 Evaluation

Analysis of the effects of heavy load drops to demonstrate that the evaluation criteria of NUREG-0612, Section 5.1 are satisfied is consistent with the guidelines of Section 5.1.4 for load handling systems within the reactor building. The 4-ton hook-type chain hoist is used over hatches to remove various equipment from lower floors to the 565-ft elevation floor. The loads analyzed by the Licensee are consistent with those previously reviewed and identified in Reference 4. The results of the Licensee's load drop analysis indicate that the physical separation of the two core spray loops precludes a load drop in one room from adversely affecting the other redundant loop as long as the suppression pool suction valve is closed following the load drop. Review of the affected equipment following a load drop indicates that the load drop will not initiate an event that will require the use of the core spray system (i.e., damage to a core spray loop will not initiate a loss-of-coolant accident). Therefore, sufficient time should be available for the operator to assess which core spray loop has been damaged and to shut the suppression pool suction isolation valve in the affected loop.

2.2.5.3 Conclusions and Recommendations

The Licensee has demonstrated that Criterion IV of NUREG-0612, Section 5.1 is satisfied for the 4-ton hook-type chain hoist and that the load handling system meets the guidelines of NUREG-0612.

2.3 LOAD HANDLING SYSTEMS IN OTHER AREAS

2.3.1 Load Handling Systems Capable of Carrying Loads in Plant Areas Outside of the Reactor Building

2.3.1.1 Summary of Licensee Statements and Conclusions

In Reference 5, the Licensee has identified the self-propelled truck crane as the only load handling system outside the reactor building capable of carrying a heavy load.

2.3.1.3 Evaluation and Conclusion

The Licensee's statement and conclusion with respect to the load handling system mentioned above are consistent with the information evaluated in Reference 4 and with the intent of NUREG-0612.

2.3.2 Self-Propelled Truck Crane

2.3.2.1 Summary of Licensee Statements and Conclusions

In Reference 5, the Licensee stated that the self-propelled truck crane does not have sufficient design features to make the likelihood of a load drop extremely small. Because the self-propelled truck crane has the physical capability of carrying loads over safe shutdown equipment, the Licensee provided a heavy load/impact area matrix. This matrix identified the various loads handled and described the consequences when these loads are dropped in specific areas. For each load postulated to drop (except the residual heat removal service water [RHRSW] pump) and impact a safety-related system, the Licensee stated that system redundancy and separation ensure that the safety-related function of the affected system is not precluded. For the postulated drop of the RHRSW pump, the Licensee is relying on site-specific considerations to eliminate the need to evaluate load drop consequences.

2.3.2.2 Evaluation

Analysis of the effects of heavy load drops to demonstrate satisfaction of the evaluation criteria of NUREG-0612, Section 5.1 is consistent with the guidelines of Section 5.1.5. At the Browns Ferry plant, the two general impact areas within the operating range of the self-propelled truck crane which contain equipment provided for safe shutdown or decay heat removal were considered. In each case, the loads analyzed were consistent with the heavy loads identified and reviewed in Reference 4.

The Licensee's analysis of load handling accidents in the vicinity of the diesel generator indicated that the potential consequences of such an accident would satisfy the relevant acceptance criterion. This analysis was based on the postulated drop of the reactor building exhaust fans onto the roof of

either the Units 1 and 2 or the Unit 3 diesel generator buildings. The air intake, filter, and exhaust for each diesel and the diesel generator air conditioning chilled water panels for the the Unit 3 diesel were identified as the safety-related equipment that might be damaged by such an accident. Because each diesel generator has an independent air intake, filter, and exhaust system, damage to any one of these systems would not impair the operation of the other diesel generators. Similar damage to one of the Unit 3 diesel generator air conditioning chilled water panels would not preclude the proper operation of other panels. In each case, the physical separation between independent or redundant systems has been found to be sufficient to prevent damage to more than one system. The Browns Ferry Updated FSAR [17] indicates that the operation of only three diesel generators in each building is required. Specifically, three out of the four diesel generators at Units 1 and 2 in parallel with three out of the four diesel generators at Unit 3 are sufficient to supply all required loads for the safe shutdown and cooldown of all three units in the event of loss of offsite power and a design basis accident in any one unit.

In the case of potential load handling accidents in the vicinity of the RHRSW system, the Licensee's response is not sufficient to support a determination that staff acceptance criteria have been satisfied. TVA's response regarding postulated drops of a condenser circulating water (CCW) pump, a CCW pump motor, or a fire pump does indicate that the physical separation of the RHRSW pumps is sufficient to preclude a common mode failure and the loss of system safety function. Retention of such capability in the case of the postulated drop of an RHRSW pump, however, appears to rely on unidentified site-specific considerations.

At the Browns Ferry plant, the RHRSW system is a fairly complex system providing substantial flexibility in operating modes. It consists of four pairs of RHRSW pumps assigned to the RHR systems for the three units, plus four additional pumps assigned to the emergency cooling water system (EECW). Each of the pump pairs feeds one independent RHRSW header which, in turn, feeds one RHR heat exchanger in each unit. Two of the individual (EECW-assigned) pumps feed one EECW header. The two remaining pumps feed the alternate EECW

header(s). Two RHRSW pumps and two RHR heat exchangers are required per unit to effectively remove afterheat under emergency conditions. Because of pump capacity limitations, one pump is necessary to supply enough coolant to serve one heat exchanger, and no single train's heat exchanger set can be operated in all three units simultaneously. This flexibility is further constrained, as identified in Reference 17, by limitations associated with allowable loadings of diesel generators and electrical boards such that certain multiple failures may disable the system. It is reasonable to believe that certain site-specific considerations, combined with the fact that damage to the RHRSW system will not initiate an event that will require the automatic initiation of the RHRSW system, can be relied upon to provide the degree of protection inherent in verbatim compliance with NUREG-0612; however, such consideration cannot be determined independently from the information provided.

2.3.2.3 Conclusion

The Licensee has demonstrated that Criterion IV of NUREG-0612, Section 5.1 is satisfied for each load handled by the self-propelled truck crane except for the postulated drop of a RHRSW pump. To demonstrate that this criterion is satisfied for all loads handled by the self-propelled truck crane, the Licensee should describe the site-specific considerations that eliminate the need to consider the postulated drop of a RHRSW pump onto an operating RHRSW pump.

3. CONCLUSION

This summary is provided to consolidate the results of crane-specific evaluations presented in Section 2. It is not meant as a substitute for the specific conclusions reached in the various subsections of Section 2. It is provided to allow the reader to focus on the key topics which should be addressed in seeking to resolve issues where the degree of load handling reliability provided by cranes at the Browns Ferry plant was not found to meet the objectives of NUREG-0612. This section addresses issues for which the information provided is felt to be inadequate to support a definitive conclusion and issues wherein the information provided has been evaluated as proposing an approach inconsistent with the the guidance of NUREG-0612.

3.1 INFORMATION ISSUES

The information provided by the Licensee has been assessed as insufficient to support an independent conclusion that load handling reliability is consistent with the evaluation criteria of Section 2.1 in the following areas:

Reactor Building Crane Lifting Devices (Section 2.2.2)

The Licensee should identify how the guidelines of NUREG-0612, Sections 5.1.6(1) and 5.1.6(3) have been satisfied for lifting devices and associated lift interfaces for heavy loads handled by the reactor building crane for which the analysis of the drop of such loads has not demonstrated that the criteria of NUREG-0612, Section 5.1 have been satisfied. Where such analyses have been employed to demonstrate that additional design features are unnecessary, the results of these analyses should be provided.

Reactor Building Crane Electrical Power/Control System (Section 2.2.2)

The Licensee should verify that a single failure in the reactor building crane electrical power/control system will not result in a load drop as discussed in Reference 16.

Reactor Building 24-Ton Gear-Type Chain Hoist (Section 2.2.4)

The Licensee should identify how the guidance of NUREG-0612, Section 5.1.4 is satisfied for operation of the reactor building 24-ton gear-type chain hoist.

Self-Propelled Truck Crane (Section 2.3.2)

The Licensee should identify the site-specific considerations which lead to the conclusion that lifts of an RHRSW pump in the vicinity of the RHRSW system cannot, in the event of a load drop, cause a loss of RHRSW system safety functions.

3.2 APPROACH ISSUES

This review has revealed no issues wherein the approach or position taken by the Licensee, based on information provided thus far, is inconsistent with the staff's objectives as expressed in the evaluation criteria of Section 2.1.

4. REFERENCES

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2. "Control of Heavy Loads at Nuclear Power Plants"
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6. D. S. Kammer (TVA)
   Letter to D. V. Vassallo (NRC)
   Subject: Item-by-Item Description of TVA's Response Presented to Mr. Clemenson (NRC) in Conference Call on October 28, 1982
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   Letter to D. V. Vassallo (NRC)
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10. L. M. Mills (TVA)
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11. H. G. Parris (TVA)
Letter to A. Schwencer (NRC)
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13. L. M. Mills (TVA)
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14. L. M. Mills (TVA)
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Subject: Revised Response to Several Questions Addressed in Letter of February 10, 1981
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15. L. M. Mills (TVA)
Letter to D. V. Vassallo (NRC)
Subject: Item-by-Item Description of TVA's Response Presented to Mr. Clemenson (NRC) in Conference Call on October 27, 1982
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16. D. G. Eisenhut (NRC)
Letter to all Holders of Operating Licensees, Application for Operating Licensees and Holders of Constructor Premit for Power Reactor (Generic Letter 83-42)
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17. L. M. Mills (TVA)
Letter to H. R. Denton (NRC)
Subject: Updated Final Safety Analysis Report
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