ENCLOSURE 2

SAI-186-029-15

TECHNICAL REPORT

IMPROVEMENTS IN TRAINING AND REQUALIFICATION PROGRAMS AS REQUIRED BY TMI ACTION ITEMS I.A.2.1 AND II.B.4

for the

H. B. Robinson Steam Electric Plant, Unit No. 2

(Docket 50-261)

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I. INTRODUCTION

Science Applications, Inc. (SAI), as technical assistance contractor to the U.S. Nuclear Regulatory Commission, has evaluated the response by Carolina Power and Light Company for the H. B. Robinson Steam Electric Plant, Unit No. 2 (Docket 50-261) to certain requirements contained in post-TMI Action Items I.A.2.1, Immediate Upgrading of Reactor Operator and Senior Reactor Operator Training and Qualifications, and II.B.4, Training for Mitigating Core Damage. These requirements were set forth in NUREG-0660 (Reference 1) and were subsequently clarified in NUREG-0737 (Reference 2).*

The purpose of the evaluation was to determine whether the licensee's operator training and requalification programs satisfy the requirements. The evaluation pertains to Technical Assignment Control (TAC) System numbers 44195 (NUREG-0737, I.A.2.1.4) and 44545 (NUREG-0737, II.B.4.1). As delineated below, the evaluation covers only some aspects of item I.A.2.1.4.

The detailed evaluation of the licensee's submittals is presented in Section IV; the conclusions are in Section V.

II. SCOPE AND CONTENT OF THE EVALUATION

A. I.A.2.1: Immediate Upgrading of RO and SRO Training and Qualifications

The clarification of TMI Action Item I.A.2.1 in NUREG-0737 incorporates a letter and four enclosures, dated March 28, 1980, from Harold R. Denton, Director, Office of Nuclear Reactor Regulation, USNRC, to all power reactor applicants and licensees, concerning qualifications of reactor operators (hereafter referred to as Denton's letter). This letter and enclosures imposes a number of training requirements on power reactor licensees. This evaluation specifically addressed a subset of the requirements stated in Enclosure 1 of Denton's letter, namely: Item A.2.c, which relates to operator training requirements; item A.2.e, which concerns instructor requalification; and Section C, which addresses operator requalification. Some of these requirements are elaborated in Enclosures 2, 3, and 4 of Denton's letter. The training requirements under evaluation are summarized in Figure 1. The elaborations of these requirements in Enclosures 2, 3 and 4 of Denton's letter are shown respectively in Figures 2, 3 and 4.

As noted in Figure 1, Enclosures 2 and 3 indicate minimum requirements concerning course content in their respective areas. In addition, the Operator Licensing Branch in NRC has taken the position (Reference 3) that the training in mitigating core damage and related subjects should consist

*Enclosure 1 of NUREG-0737 and NRC's Technical Assistance Control System distinguish four sub-actions within I.A.2.1 and two sub-actions within II.B.4. These subdivisions are not carried forward to the actual presentation of the requirements in Enclosure 3 of NUREG-0737. If they had been, the items of concern here would be contained in I.A.2.1.4 and II.B.4.1.

Figure 1. Training Requirements from TMI Action Item I.A.2.1*

Program Element	NRC Requirements**
OPERATIONS PERSONNEL TRAINING	Enclosure 1, Item A.2.c(1) Training programs shall be modified, as necessary, to provide training in heat transfer, fluid flow and thermodynamics. (Enclosure 2 provides guidelines for the minimum content of such training.) Enclosure 1, Item A.2.c(2) Training programs shall be modified, as necessary to provide training in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged. (Enclosure 3 provides guidelines for the minimum content of such training.) Enclosure 1, Item A.2.c.(3) Training programs shall be modified, as necessary to provide increased emphasis on reactor and plant transients.
INSTRUCTOR REQUALIFICATION	Enclosure 1, Item A.2.e Instructors shall be enrolled in appropriate requalification programs to assure they are cognizant of current operating history, problems, and changes to procedures and administrative limitations.
PERSONNEL REQUALIFICATION	Enclosure 1, Item C.1 Content of the licensed operator requalification programs shall be modified to include instruction in heat transfer, fluid flow, thermodynamics, and mitigation of accidents involving a degraded core. (Enclosures 2 and 3 provide guidelines for the minimum content of such training.) Enclosure 1, Item C.2 The criteria for requiring a licensed individual to participate in accelerated requalification shall be modified to be consistent with the new passing grade for issuance of a license: 80% overall and 70% each category. Enclosure 1, Item C.3 Programs should be modified to require the control manipulations listed in Enclosure 4. Normal control manipulations, such as plant or reactor startups, must be performed. Control manipulations during abnormal or emergency operations must be walked through with, and evaluated by, a member of the training staff at a minimum. An appropriate simulator may be used to satisfy the requirements for control manipulations.

*The requirements shown are a subset of those contained in Item I.A.2.1.
**References to Enclosures are to Denton's letter of March 28, 1980, which is contained in the clarification of Item I.A.2.1 in NUREG-0737.

Figure 2. Enclosure 2 from Denton's Letter

TRAINING IN HEAT TRANSFER, FLUID FLOW AND THERMODYNAMICS

1. Basic Properties of Fluids and Matter.

This section should cover a basic introduction to matter and its properties. This section should include such concepts as temperature measurements and effects, density and its effects, specific weight, buoyancy, viscosity and other properties of fluids. A working knowledge of steam tables should also be included. Energy movement should be discussed including such fundamentals as heat exchange, specific heat, latent heat of vaporization and sensible heat.

2. Fluid Statics.

This section should cover the pressure, temperature and volume effects on fluids. Example of these parametric changes should be illustrated by the instructor and related calculations should be performed by the students and discussed in the training sessions. Causes and effects of pressure and temperature changes in the various components and systems should be discussed in the training sessions. Causes and effects of pressure and temperature changes in the various components and systems should be discussed as applicable to the facility with particular emphasis on safety significant features. The characteristics of force and pressure, pressure in liquids at rest, principles of hydraulics, saturation pressure and temperature and subcooling should also be included.

3. Fluid Dynamics.

This section should cover the flow of fluids and such concepts as Bernoulli's principle, energy in moving fluids, flow measure theory and devices and pressure losses due to friction and orificing. Other concepts and terms to be discussed in this section are NPSH, carry over, carry under, kinetic energy, head-loss relationships and two phase flow fundamentals. Practical applications relating to the reactor coolant system and steam generators should also be included.

4. Heat Transfer by Conduction, Convection and Radiation.

This section should cover the fundamentals of heat transfer by conductions. This section should include discussions on such concepts and terms as specific heat, heat flux and atomic action. Heat transfer characteristics of fuel rods and heat exchangers should be included in this section.

This section should cover the fundamentals of heat transfer by convection. Natural and forced circulation should be discussed as applicable to the various systems at the facility. The convection current patterns created by expanding fluids in a confined area should be included in this section. Heat transport and fluid flow reductions or stoppage should be discussed due to steam and/or noncondensible gas formation during normal and accident conditions.

This section should cover the fundamentals of heat transfer by thermal radiation in the form of radiant energy. The electromagnetic energy emitted by a body as a result of its temperature should be discussed and illustrated by the use of equations and sample calculations. Comparisons should be made of a black body absorber and a white body emitter.

5. Change of Phase - Boiling.

This section should include descriptions of the state of matter, their inherent characteristics and thermodynamic properties such as enthalpy and entropy. Calculations should be performed involving steam quality and void fraction properties. The types of boiling should be discussed as applicable to the facility during normal evolutions and accident conditions.

6. Burnout and Flow Instability.

This section should cover descriptions and mechanisms for calculating such terms as critical flux, critical power, DNB ratio and hot channel factors. This section should also include instructions for preventing and monitoring for clad or fuel damage and flow instabilities. Sample calculations should be illustrated by the instructor and calculations should be performed by the students and discussed in the training sessions. Methods and procedures for using the plant computer to determine quantitative values of various factors during plant operation and plant heat balance determinations should also be covered in this section.

7. Reactor Heat Transfer Limits.

This section should include a discussion of heat transfer limits by examining fuel rod and reactor design and limitations. The basis for the limits should be covered in this section along with recommended methods to ensure that limits are not approached or exceeded. This section should cover discussions of peaking factors, radial and axial power distributions and changes of these factors due to the influence of other variables such as moderator temperature, xenon and control rod position.

Figure 3. Enclosure 3 from Denton's Letter

TRAINING CRITERIA FOR MITIGATING CORE DAMAGE

A. Incore Instrumentation

1. Use of fixed or movable incore detectors to determine extent of core damage and geometry changes.

2. Use of thermocouples in determining peak temperatures; methods for extended range readings; methods for direct readings at terminal junctions.

3. Methods for calling up (printing) incore data from the plant computer.

B. Excore Nuclear Instrumentation (NIS)

1. Use of NIS for determination of void formation; void location basis for NIS response as a function of core temperatures and density changes.

C. Vital Instrumentation

1. Instrumentation response in an accident environment; failure sequence (time to failure, method of failure); indication reliability (actual vs indicated level).

2. Alternative methods for measuring flows, pressures, levels, and temperatures.

   a. Determination of pressurizer level if all level transmitters fail.

   b. Determination of letdown flow with a clogged filter (low flow).

   c. Determination of other Reactor Coolant System parameters if the primary method of measurement has failed.

D. Primary Chemistry

1. Expected chemistry results with severe core damage; consequences of transferring small quantities of liquid outside containment; importance of using leak tight systems.

2. Expected isotopic breakdown for core damage; for clad damage.

3. Corrosion effects of extended immersion in primary water; time to failure.

E. Radiation Monitoring

1. Response of Process and Area Monitors to severe damages; behavior of detectors when saturated; method for detecting radiation readings by direct measurement at detector output (overranged detector); expected accuracy of detectors at different locations; use of detectors to determine extent of core damage.

2. Methods of determining dose rate inside containment from measurements taken outside containment.

F. Gas Generation

1. Methods of H_2 generation during an accident; other sources of gas (Xe, Ke); techniques for venting or disposal of non-condensibles.

2. H_2 flammability and explosive limit; sources of O_2 in containment or Reactor Coolant System.

Figure 4. Control Manipulations Listed in Enclosure 4.

CONTROL MANIPULATIONS

*1. Plant or reactor startups to include a range that reactivity feedback from nuclear heat addition is noticeable and heatup rate is established.

2. Plant shutdown.

*3. Manual control of steam generators and/or feedwater during startup and shutdown.

4. Boration and or dilution during power operation.

*5. Any significant (greater than 10%) power changes in manual rod control or recirculation flow.

6. Any reactor power change of 10% or greater where load change is performed with load limit control or where flux, temperature, or speed control is on manual (for HTGR).

*7. Loss of coolant including:

   1. significant PWR steam generator leaks
   2. inside and outside primary containment
   3. large and small, including leak-rate determination
   4. saturated Reactor Coolant response (PWR).

8. Loss of instrument air (if simulated plant specific).

9. Loss of electrical power (and/or degraded power sources).

*10. Loss of core coolant flow/natural circulation.

11. Loss of condenser vacuum.

12. Loss of service water if required for safety.

13. Loss of shutdown cooling.

14. Loss of component cooling system or cooling to an individual component.

15. Loss of normal feedwater or normal feedwater system failure.

*16. Loss of all feedwater (normal and emergency).

17. Loss of protective system channel.

18. Mispositioned control rod or rods (or rod drops).

19. Inability to drive control rods.

20. Conditions requiring use of emergency boration or standby liquid control system.

21. Fuel cladding failure or high activity in reactor coolant or offgas.

22. Turbine or generator trip.

23. Malfunction of automatic control system(s) which affect reactivity.

24. Malfunction of reactor coolant pressure/volume control system.

25. Reactor trip.

26. Main steam line break (inside or outside containment).

27. Nuclear instrumentation failure(s).

* Starred items to be performed annually, all others biennially.

of at least 80 contact hours* in both the initial training and the requalification programs. The NRC considers thermodynamics, fluid flow and heat transfer to be related subjects, so the 80-hour requirement applies to the combined subject areas of Enclosures 2 and 3. The 80 contact hour criterion is not intended to be applied rigidly; rather, its purpose is to provide greater assurance of adequate course content when the licensee's training courses are not described in detail.

Since the licensees generally have their own unique course outlines, adequacy of response to these requirements necessarily depends only on whether it is at a level of detail comparable to that specified in the enclosures (and consistent with the 80 contact hour requirement) and whether it can reasonably be concluded from the licensee's description of his training material that the items in the enclosures are covered.

The Institute of Nuclear Power Operations (INPO) has developed its own guidelines for training in the subject areas of Enclosures 2 and 3. These guidelines, given in References 4 and 5, were developed in response to the same requirements and are more than adequate, i.e., training programs based specifically on the complete INPO documents are expected to satisfy all the requirements pertaining to training material which are addressed in this evaluation.

The licensee's response concerning increased emphasis on transients is considered by SAI to be acceptable if it makes explicit reference to increased emphasis on transients and gives some indication of the nature of the increase, or, if it addresses both normal and abnormal transients (without necessarily indicating an increase in emphasis) and the requalification program satisfies the requirements for control manipulations, Enclosure 1, Item C.3. The latter requirement calls for <u>all</u> the manipulations listed in Enclosure 4 (Figure 4 in this report) to be performed, at the frequency indicated, unless they are specifically not applicable to the licensee's type of reactor(s). Some of these manipulations may be performed on a simulator. Personnel with senior licenses may be credited with these activities if they direct or evaluate control manipulations as they are performed by others. Although these manipulations are acceptable for meeting the reactivity control manipulations required by Appendix A paragraph 3.a of 10 CFR 55, the requirements of Enclosure 4 are more demanding. Enclosure 4 requires about 32 specific manipulations over a two-year cycle while 10 CFR 55 Appendix A requires only 10 manipulations over a two-year cycle.

B. II.B.4: Training for Mitigating Core Damage

Item II.B.4 in NUREG-0737 requires that "shift technical advisors and operating personnel from the plant manager through the operations chain to the licensed operators" receive training on the use of installed systems to control or mitigate accidents in which the core is severely damaged.

*A contact hour is a one-hour period in which the course instructor is present or available for instructing or assisting students; lectures, seminars, discussions, problem-solving sessions, and examinations are considered contact periods. This definition is taken from Reference 4.

Enclosure 3 of Denton's letter provides guidance on the content of this training. "Plant Manager" is here taken to mean the highest ranking manager at the plant site.

For licensed personnel, this training would be redundant in that it is also required, by I.A.2.1, in the operator requalification program. However, II.B.4 applies also to operations personnel who are not licensed and are not candidates for licenses. This may include one or more of the highest levels of management at the plant. These non-licensed personnel are not explicitly required to have training in heat transfer, fluid flow and thermodynamics and are therefore not obligated for the full 80 contact hours of training in mitigating core damage and related subjects.

Some non-operating personnel, notably managers and technicians in instrumentation and control, health physics and chemistry departments, are supposed to receive those portions of the training which are commensurate with their responsibilities. Since this imposes no additional demands on the program itself, we do not address it in this evaluation. It would be appropriate for resident inspectors to verify that non-operating personnel receive the proper training.

* * * * *

The required implementation dates for all items have passed. Hence, this evaluation did not address the dates of implementation. Moreover, the evaluation does not cover training program modifications that might have been made for other reasons subsequent to the response to Denton's letter.

III. LICENSEE SUBMITTALS

The licensee (CP&L) has submitted to NRC a number of items (letters and various attachments) which explain their training and requalification programs. These submittals, made in response to Denton's letter, form the information base for this evaluation. For the Robinson plant, there were two submittals with attachments, for a total of 9 items, which are listed below.

1. Letter from R.B. Starkey, Jr., General Manager, H.B. Robinson S.E. Plant, Carolina Power & Light Co., to P.F. Collins, Chief of Operator Licensing Branch, NRC. July 11, 1980. (2 pp, with enclosures: items 2, 3, 4, 5, & 6). NRC Acc No: 8007220426. (re: Response to NRC letter dated March 28, 1980).

2. "Training Instruction NO. 201, H.B. Robinson Plant Operator Replacement Training Program" Carolina Power & Light Co., H.B. Robinson Plant, (Rev. 4). Approved by R.B. Starkey, Jr. May 19, 1980. (9 pp, attached to item 1). NRC Acc No: 8007220429.

3. "Training Instruction No. 201A, Training of Replacement Reactor Operators for NRC Examination

Without Reactor Startup Demonstration", Carolina Power & Light Co., H.B. Robinson Plant, (Rev. 4). Approved for R.B. Starkey, June 27, 1980. (16 pp, attached to item 1). NRC Acc No: 8007220433.

4. "Training Instruction No. 203, Senior Reactor Operator Replacement Training Program" Carolina Power & Light Co., H.B. Robinson Plant, (Rev. 0). Approved by R.B. Starkey, Jr. May 25, 1980. (10 pp, attached to item 1). NRC Acc No: 8007220435.

5. "Training Instruction No. 902, H.B. Robinson SEG Plant, Instructors Requalification Program", Carolina Power & Light Co., H.B. Robinson Plant, (Rev. 0). Approved by R.B. Starkey, May 25, 1980. (3 pp, attached to item 1). NRC Acc No: 8007220438.

6. "10.2 Operator Requalification Program, H.B. Robinson Unit No. 2", Administrative Instruction, Volume 1, Section 10, Paragraph 10.2, Rev. 69. May 15, 1980. (7 pp, attached to item 1). Serial No. RSEP/80-1035

7. Letter from P.W. Howe, Vice President Technical Services, Carolina Power & Light Co., H.B. Robinson Plant, Unit 2, to S.A. Varga, Chief of Operating Reactors Branch #1, Division of Licensing, NRC. May 18, 1982. (2 pp, with enclosures: items 7 & 8). NRC Acc No 8205210193:. (re: Response to NRC's RAI dated April 13, 1982).

8. "Response to April 13, 1982 Letter", Enclosure. Undated. (3 pp, attached to item 7).

9. "Organizational Chart of Personnel Receiving Mitigating Core Damage Training", Attachment. Undated. (1 pg, attached to item 7).

The last three items were submitted in response to a request for additional information (Reference 6).

IV. EVALUATION

SAI's evaluation of the training programs at Carolina Power and Light Company's H. B. Robinson Steam Electric Plant, Unit No. 2, is presented below. Section A addresses TMI Action Item I.A.2.1 and presents the assessment organized in the manner of Figure 1. Section B addresses TMI Action Item II.B.4.

A. I.A.2.1: Immediate Upgrading of Reactor Operator and Senior Reactor Operator Training and Qualification.

Enclosure 1, Item A.2.c(1)

The basic requirements are that the training programs given to reactor operator and senior reactor operator candidates cover the subjects of heat transfer, fluid flow and thermodynamics at the level of detail specified in Enclosure 2 of Denton's letter.

The licensee has three distinct training programs, two for ROs (TI-201 and TI-201A) and one for SROs (TI-203). In all three cases, a major section entitled "Heat Transfer, Fluid Flow, Thermodynamics" has been added to the list of lecture topics. The subtopics indicated are precisely the numbered subtopics in Enclosure 2 of Denton's letter. In response (submittal item 8) to a request for additional information (Reference 6), the licensee indicated that 80 contact hours were involved in covering these subjects in the two RO training programs. The SRO program was inadvertantly omitted from the inquiry so the licensee did not respond in this case; however, a subsequent telephone inquiry by the NRC Project Manager to the licensee indicated that 80 contact hours were provided in the SRO program also (Reference 7). Consequently, SAI concludes that the licensee satisfies this requirement.

Enclosure 1, Item A.2.c(2)

The requirements are that the training programs for reactor and senior reactor operator candidates cover the subject of accident mitigation at the level of detail specified in Enclosure 3 of Denton's letter (see Figure 3 of this report).

Each of the three training programs has also been augmented by a major lecture section entitled "Mitigating Core Damage"; again, the subtopics are the numbered topics from the relevant enclosure (number 3) to Denton's letter. Submittal item 8 indicated that 35 hours were devoted to this subject in the RO training programs and, as before, this response was subsequently determined to apply also to the SRO program. When combined with the 80 hours devoted to heat transfer, fluid flow and thermodynamics, this is more than adequate to satisfy NRC's intentions. The licensee clearly meets this requirement.

Enclosure 1, Item A.2.c(3)

The requirement is that there be an increased emphasis in the training program on dealing with reactor transients.

The licensee asserts (in submittal item 8) that an increased emphasis on transients has been incorporated into both the RO and SRO training programs. These include 5 and 2 weeks, respectively, of simulator training and both include 2 weeks of transient and accident analysis. We conclude that the licensee meets the NRC requirement.

Enclosure 1, Item A.2.e

The requirement is that instructors for reactor operator training programs be enrolled in appropriate requalification programs to assure they are cognizant of current operating history, problems and changes to procedures and administrative limitations.

The licensee has instituted an Instructor Requalification Program (see submittal items 1 and 5) specifically in specifically in response to this requirement. This program satisfies NRC's requirement in an exemplary manner.

Enclosure 1, Item C.1

The primary requirement is that the requalification programs have instruction in the areas of heat transfer, fluid flow, thermodynamics and accident mitigation. The level of detail required in the requalification program is that of Enclosures 2 and 3 of Denton's letter. In addition, these instructions must involve an adequate number of contact hours.

As did the training programs, the licensee's requalification program (TI-902) includes two major new sections, one on heat transfer, fluid flow and thermodynamics and one on mitigating core damage. The subtopics correspond to those of Denton's Enclosures 2 and 3. In addition, some of the other sections cover material that is related to topics in mitigating core damage. Therefore, with regard to the basic subject matter, the program adequately reflects the requirements.

In 1981, CP&L's requalification program at Robinson involved about 84 contact hours of training relating to mitigation of core damage, distributed as follows: Heat Transfer, Fluid Flow, Thermodynamics, 40 hours, Mitigating Core Damage Training, 12 hours, Simulator Retraining relating to mitigation of core damage, 32 hours (50% of the total simulator retraining time). In 1982, the requalification program pertaining to mitigation of core damage will consist of about 41 hours, distributed as 6 hours, 3 hours and 32 hours, respectively, among the program elements noted above.

Noting again that some time devoted to other major sections of the program would relate to mitigating core damage, the Robinson program would entail in excess of 80 hours over the two year requalification cycle required by 10CFR55. Technically, this complies with the NRC requirement. This conclusion, however, depends on two assumptions: (1) that the annual training in future years does not drop below the 1982 level of about 41 hours, and (2) that all licensed operators participate each year (as opposed to an alternating year arrangement).

Enclosure 1, Item C.2

The requirement for licensed operators to participate in the accelerated requalification program must be based on passing scores of 80% overall, 70% in each category. The licensee's requalification program explicitly includes such a provision.

Enclosure 1, Item C.3

TMI Action Item I.A.2.1 calls for the licensed operator requalification program to include performance of control manipulations involving both normal and abnormal situations. The specific manipulations required and their performance frequency are identified in Enclosure 4 of the Denton letter (see Figure 4 of this report).

The licensee explicitly lists all the manipulations of Enclosure 4 (Figure 4 herein), except one (#6) which is not applicable to their reactor, in the requalification program. The program includes a commitment to the frequency of performance and other administrative provisions of the enclosure. The licensee meets all aspects of this requirement.

B. II.B.4 Training for Mitigating Core Damage

Item II.B.4 requires that training for mitigating core damage, as indicated in Enclosure 3 of Denton's letter, be given to shift technical advisors and operating personnel from the plant manager to the licensed operators. This includes both licensed and non-licensed personnel.

Submittal item 8 lists the topics covered in training for mitigating core damage. These adequately encompass the topics of Enclosure 3. The number of contact hours involved is 35. The licensed operators would also have received about 40 hours of training, via the 1981 requalification program, in heat transfer, fluid flow and thermodynamics. They also would have received about 50 hours of simulator training related to mitigating core damage. Together, these total more than 80 contact hours in mitigating core damage and related subjects. This requirement is therefore met for licensed operators.

At Robinson, the licensee identifies the General Plant Manager and the Shift Technical Advisors as non-licensed personnel required to be trained in mitigating core damage and indicate that these personnel have received the lectures. The requirement is therefore satisfied for non-licensed operations personnel.

V. CONCLUSIONS

SAI has evaluated the submittals by Carolina Power and Light Company to NRC, for the H. B. Robinson Steam Electric Plant, Unit 2, in response to NUREG-0737 items I.A.2.1 and II.B.4. We conclude there is reasonable assurance that the licensee has satisifed all of the associated requirements in his current training and requalification programs at the Robinson Plant.

V. REFERENCES

1. "NRC Action Plan Developed as a Result of the TMI-2 Accident." NUREG-0660, United States Nuclear Regulatory Commission. May 1980.

2. "Clarification of TMI Action Plan Requirements," NUREG-0737, United States Nuclear Regulatory Commission. November 1980.

3. The NRC requirement for 80 contact hours is an Operator Licensing Branch technical position. It was included with the acceptance criteria provided by NRC to SAI for use in the present evaluation. See letter, Harley Silver, Technical Assistance Program Management Group, Division of Licensing, USNRC to Bryce Johnson, Program Manager, Science Applications, Inc., Subject: Contract No. NRC-03-82-096, Final Work Assignment 2, December 23, 1981.

4. "Guidelines for Heat Transfer, Fluid Flow and Thermodynamics Instruction," STG-02, The Institute of Nuclear Power Operations. December 12, 1980.

5. "Guidelines for Training to Recognize and Mitigate the Consequences of Core Damage," STG-01, The Institute of Nuclear Power Operations. January 15, 1981.

6. Letter, Steven A. Varga, Chief Operating Reactors Branch #1, Div. of Licensing, USNRC to J. A. Jones, Vice Chairman, Carolina Power and Light Co., Subject: Upgraded SRO and RO Training and Training for Mitigating Core Damage - Request for Additional Information, April 13, 1982.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO NUREG-0737, ITEM II.B.2.2

SHIELDING MODIFICATIONS FOR VITAL AREA ACCESS

CAROLINA POWER AND LIGHT COMPANY

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

DOCKET NO. 50-261

INTRODUCTION

Following the accident at TMI-2, the NRC staff developed Action Plan NUREG-0660, and "Clarification of TMI Action Plan Requirements", NUREG-0737, to provide for improved safety at nuclear power plants. NUREG-0737, Item II.B.2.2, directed all licensees to perform a design review of plant shielding and to provide for adequate post accident access to vital areas by design changes, increased temporary or permanent shielding, or post accident procedural controls.

The plant shielding design review for the H. B. Robinson Unit 2 facility was described by Carolina Power and Light Company in its letter to the NRC dated December 31, 1979. Supplemental letters clarifying the licensee's response to this item were submitted March 31, 1980, and December 31, 1980. The following evaluation contains the results of the post implementation review of the shielding study for NUREG-0737, Item II.B.2.2, entitled, "Plant Shielding Modifications for Vital Area Access."

EVALUATION

In response to NUREG-0737 Item II.B.2.2, "Plant Shielding Modifications for Vital Area Access", a design review of the H. B. Robinson Unit 2 plant shielding was performed. In accordance with the requirements, radiation source terms were specified, systems assumed to contain high levels of radioactivity as a result of a postulated accident were determined, vital areas requiring access were identified, and dose rates in various plant areas and vital areas were calculated.

The licensee's response to NUREG-0737, Item II.B.2.2 was reviewed during NRC Region II inspections 50-261/82-04 and 50-261/82-14. The assumptions and methodology employed by the licensee in the shielding design review were found to be consistent with the requirements. Source terms were based on source term requirements contained in NUREG-0737. The systems identified as potentially containing high concentrations of radioactivity following an accident were found to be consistent with system functions.

Licensee responses to this item were dated December 31, 1979, March 31, 1980, and December 31, 1980. The licensee identified areas which would require access or occupancy in order to mitigate the consequences of the postulated accident. Each area was evaluated in the plant shielding design review to ensure that these areas would be accessible without exposing an individual to radiation in excess

of GDC 19 criteria. The licensee identified the control room, technical support center, and hot chemical laboratory as vital areas requiring continuous occupancy and calculated the maximum dose rate in any of these areas to be less than 15 mrem/hr.

Due to the findings of the shielding design study, the licensee determined that no shielding modifications are required. The inspector noted that CP&L had not completed installation of their Post Accident Sampling System (PASS) which will be located in a vital area outside the hot chemistry laboratory in the auxiliary building. Evaluation of shielding requirements for post accident sampling will be conducted in conjunction with NUREG-0737, Item II.B.3, "Post Accident Sampling."

Three emergency procedures were reviewed in order to verify that vital area access or occupancy would be permissible for operations necessary to mitigate the consequences of the postulated accident. The procedures reviewed included emergency procedures for reactor coolant system depressurization, containment venting, and systems sampling. It was determined that vital areas will be accessible for required operations.

CONCLUSION

Based on our review, which included site inspections, we have concluded that the H. B. Robinson Unit 2 plant modifications, procedural control, accessibility of vital areas following an accident and shielding design meet the staff's requirements for NUREG-0737 Item II.B.2.2 and therefore are acceptable.

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