SAI-186-029-11

TECHNICAL EVALUATION REPORT

IMPROVEMENTS IN REACTOR OPERATOR AND SENIOR REACTOR OPERATOR TRAINING AND REQUALIFICATION PROGRAMS

for the

R. E. Ginna Nuclear Power Plant

(Docket 50-244)

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Prepared By:

Science Applications, Inc.
1710 Goodridge Drive
McLean, Virginia 22102

Prepared for:

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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8205280327

Science Applications, Inc.

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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 Rochester Gas and Electric Corporation (RG&E) for the R. E. Ginna Nuclear Power Plant (Docket 50-244) to certain requirements contained in post-TMI Action Items I.A.2.1, Immediate Upgrading of Reactor Operator and Senior Reactor Operator Training and Qualification, 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 44164 (NUREG-0737, I.A.2.1.4) and 44514 (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

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.

This evaluation also encompassed Action Item II.B.4, Training for Mitigating Core Damage. The requirements imposed by this item are essentially identical to those of item A.2.c(2) of Enclosure 1 of Denton's letter. An evaluation of either is applicable to both. 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.

*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.
***This requirement is essentially the same as Action Item II.B.4 of 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 biannually.

As noted in Figure 1, Enclosures 2 and 3 indicate minimum requirements concerning course content in their respective areas for both initial training and requalification programs. 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 of at least 80 contact hours.* It 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 requirement 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 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.

Both Enclosure 3 and Action Item II.B.4 require that training for mitigating core damage be given to all operations personnel from the plant manager to the licensed operators.

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). Personnel with senior licensees 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 restrictive. Enclosure 4 requires about 32 manipulations over a two-year cycle while 10 CFR 55 Appendix A requires only 10 manipulations over a two-year cycle.

*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.

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 (RG&E) 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 Ginna plant, there were four submittals with attachments, for a total of eight items, which are listed below. The last three items were in a combined submittal in response to a request for additional information prepared by SAI, dated February 24, 1982, and transmitted by NRC to the licensee in a letter dated March 23, 1982.

1. Letter from L.D. White, Jr., Vice President Rochester Gas & Electric Corporation, to D.M. Crutchfield, Chief of Operating Reactors Branch #5, NRC. August 25, 1980. (1 pg, with enclosure: item 2). NRC Acc. No: 8009020086. (re: 05 additional TMI-2 related requirements).

2. "Response to NRC letter dated March 28, 80 (Qualifications of Reactor Operators)". R.E. Ginna Power Plant, Unit No. 1, Rochester Gas & Electric Corp. (3 pp, attached to item 1).

3. Letter from L.D. White, Jr., Vice President Rochester Gas & Electric Corporation, to P.F. Collins, Chief of Operator Licensing Branch, NRC. August 22, 1980. (1 pg, with enclosure: item 4). NRC Acc No: 8009050274 (Transmittal)

4. "R.E. GINNA Operator Requalification Program", Rochester Gas & Elec. Corp., Ginna Station, Procedure No: A-102.14, Rev. No: 3. Approved for use by the Plant Superintendent, May 21, 80. (13 pp, attached to item 3). NRC Acc No:8009050278

5. Letter from J.E. Maier, Rochester Gas & Electric Corporation, to D.M. Crutchfield, Chief of Operating Reactors Branch #5, NRC. March 13,81. (1 pg). NRC Acc No: 8103250242. (re: Training for mitigating core damage, NUREG-0737).

6. Letter from J.E. Maier, Vice President, Electric and Steam Production, Rochester Gas & Elect. Corp., to D. M. Crutchfield, Chief of Operating Reactors Branch #5, NRC. April 6, 82. (8 pp, with enclosures: item 7 & 8). NRC Acc No: 8204190066. (re:

Response to RAI letter dated 03/23/82, concerning NUREG-0737, items I.A.2.1, and II.B.4).

7. "Westinghouse Mitigating Core Damage" R.E. Ginna Power Plant, Unit No.1. Undated (10 pp, attached to item 6).

8. "Introduction to Physics, Thermodynamics, Fluid & Fluid Flow Principles" Ginna, 1980, Revision 3, 06/25/81. (4 pp, attached to item 6).

IV. EVALUATION

SAI's evaluation of the training programs at Rochester Gas and Electric Corporation's R. E. Ginna Nuclear Power Plant 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. TMI Action Item I.A.2.1: Upgrading of Reactor Operator and Senior Reactor Operator Training and Requalification Programs.

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 submittal of RG&E which addressed item A.2.c.(1) (submittal item 2) stated that the licensing training program (Administrative Procedure A-102.13) was revised on May 2, 1980, to include the necessary topics. No further details were available until RG&E submitted additional information (submittal items 6 and 8). In these submittals RG&E stated that their level of instruction was comparable with Enclosure 2 of Denton's letter. They also provided a table of contents for their instruction "Introduction to Physics, Thermodynamics, Fluid and Fluid Flow Principles." The table of contents does not have one-for-one correspondence with the items of Denton's Enclosure 2. It does, however, have a moderate level of detail and outlines a program which would appear to contain all of the required material.

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). SAI has examined the submittals of RG&E and has found that most of the elements identified in Enclosure 3 of Denton's letter are explicitly identified in the response to SAI's request for additional information. A few items are not explicitly identified, these being "alternate measurement methods," "gas generation," and "corrosion." It seems reasonable to expect that these unidentified items are covered in the training program because (1) they are explicitly identified in the Westinghouse submittal to NRC of July 23, 1980 (Reference 5), dealing with accident mitigation, and (2) RG&E has arranged for Westinghouse to teach an accident

mitigation training course for Ginna personnel through August 1981. This analysis suggests that the requirements were met through August 1981. Assuming this or a similar program has been used since August 1981, the Ginna Training program still meets the NRC requirements.

RG&E does not describe the extent of their training programs in terms of "contact hours," but rather in terms of days. We estimate that the training in mitigating core damage and related subjects (including heat transfer, fluid flow and thermodynamics) involves in excess of 100 contact hours. In view of NRC's criterion for 80 contact hours, we take this as further evidence that the training programs at Ginna satisfy NRC's requirements regarding course content and level of detail.

Another requirement relative to accident mitigation training is that it be given to all operating personnel from the plant manager down to the licensed operators and also to the plant technical advisors. Again, based on information supplied by RG&E in their response to SAI's request for information, it appears that this requirement is satisfied at the Ginna plant. Specifically, this training is given to personnel holding the following positions: plant superintendent, assistant superintendent, operations engineer, operations supervisor, shift supervisor, head control operator, control operator, technical assistant for operational assessment, and shift technical advisor.

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 submittal of RG&E dated August 25, 1980, (submittal item 2) addressed this requirement by saying that the training program was revised to meet the requirement. No additional details were provided in this particular submittal. In a later submittal (submittal item 6), RG&E stated that two additional weeks of training are involved. This increase in program length is associated with an increase in scope, the most significant change being an increase in simulator use.

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 RG&E submittal of August 25, 1980 stated that the administrative procedure A-102.14 was revised to accommodate the requirement. The procedure itself stated that instructors shall participate in a program to keep them current in plant changes which include procedure changes, current operating history, current R.E. Ginna LERs and other relevant LERs.

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.

RG&E's submittal of August 25, 1980 (submittal item 2) stated that the requirement for including these materials in the requalification program was met by modifying the requalification program (Administrative Procedure A-102.14). The program defined in A-102.14 listed lectures on "heat transfer, fluid flow and thermodynamics" and "mitigating core damage during accidents." No further details were available until RG&E submitted a response on April 6, 1982 (submittal items 6 and 8). In this response, RG&E attached the table of contents for the instructions given in these two general areas. The details were the same as discussed and analyzed previously in items A.2.c.(1) and A.2.c.(2). Any judgment made about technical adequacy for those items would be applicable for this item. Estimates of the contact hours involved with this item have been made based on information supplied by RG&E. The estimate is that 88 contact hours are involved which is greater than the necessary number of hours according to this NRC criterion.

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. According to the submittal of August 25, 1980 (submittal item 2), the training program at Ginna has been judged to meet the requirement.

Enclosure 1, Item C.3

TMI Action item 1.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 RG&E submittal of August 25, 1980 (submittal item 2) stated that Administrative Procedure A-102.14 had been modified to meet the requirement. In procedure A-102.14, all of the appropriate Enclosure 4 manipulations are included with control manipulation titles similar or identical to those of Enclosure 4. The frequency of the manipulation performance is also compatible with the requirements of Enclosure 4.

B. TMI Action Item II.B.4 Training for Mitigating Core Damage

Training for mitigating core damage in both initial training and requalification programs as required by II.B.4 is essentially identical to that required by I.A.2.1 under Enclosure 1 items A.2.c(2) and C.1. Our evaluation of the Ginna programs with respect to the latter two items is therefore applicable to II.B.4.

V. CONCLUSIONS

Based on SAI's evaluation as discussed above, we conclude that the training programs at the R.E. Ginna Nuclear Power Plant very likely meet the requirements of NUREG-0737: I.A.2.1 as delineated in Section II of this report; and of NUREG-0737: II.B.4. Specific areas NRC inspectors should emphasize during their audits to verify this conclusion are:

- the level of detail in the heat transfer, fluid flow and thermodynamic course which are taught in both the training and requalification programs, and

- the continuation of an accident mitigation program which is comparable to the initial one instituted with the vendor.

VI. 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 position regarding the requirement for 80 contact hours is an informal one. 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, J. J. Evans, Manager, Westinghouse Electric Corporation Nuclear Training Services, to Paul Collins, Chief, Operator Licensing Branch, Division of Reactor Licensing, Nuclear Regulatory Commission, with attachments A through I, July 23, 1980.