UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 23, 1982

MEMORANDUM FOR: Ashok Thadani, Chief
Reliability & Risk Assessment Branch, DST

FROM: Valeria H. Wilson, Management Analysis Branch
Planning & Program Analysis Staff, NRR

SUBJECT: FOIA 82-145 - REQUEST FROM DIANE CURRAN FOR A COPY OF A MEMO FROM THOMAS E. MURLEY TO HAROLD R. DENTON DATED 10/13/81

The subject FOIA request is enclosed for your action by March 26, 1982. Please provide documents which you might have that are subject to this request, along with a list of such documents. Also complete the FOIA time record form and return it with the documents.

If you believe the expected search time will exceed two hours, please contact me immediately. Also indicate which other NRR office might have documents subject to this request.

Valeria H. Wilson

Valeria H. Wilson
Management Analysis Branch
Planning & Program Analysis Staff

Enclosure:
FOIA Request

HARMON & WEISS

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SUITE 506

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OF COUNSEL
L. THOMAS GALLOWAY

March 17, 1982

FREEDOM OF INFORMATION
ACT REQUEST
FOIA-82-145
Rec'd 3-19-82

J.M. Felton, Director
Division of Rules and Records
Office of Administration
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Freedom of Information Act Request

Dear Mr. Felton:

Pursuant to the Freedom of Information Act, 5 U.S.C. §552 et seq., the Union of Concerned Scientists (UCS) through undersigned counsel, requests a copy of a memorandum from Thomas E. Murley to Harold R. Denton, Director, Nuclear Reactor Regulation, dated October 13, 1981. The memorandum, which discusses terminology regarding safety-related equipment, is referenced in a memorandum on that subject by Harold Denton, dated November 20, 1981.

UCS asks that any applicable fee be waived as provided for in NRC regulation 10 C.F.R. 9.14a, since disclosure of this information would be "in the public interest." UCS is a nonprofit organization concerned primarily with the health, safety and environmental problems associated with the development of nuclear technology. The organization, which has approximately 45,000 sponsors across the country, has long been active in representing the public interest in NRC licensing and regulatory proceedings. The requested information will assist UCS in continuing its work to improve and broaden the coverage of safety standards for nuclear power plants. UCS's contribution in this area through its 1977 Petition for Emergency and Remedial Action, calling for the environmental qualification of safety-related electrical equipment in nuclear power plants, has been acknowledged by the Commission. CLI-80-21, 11 NRC 707, 710 (1980). UCS has also litigated the issue of environmental qualification of safety-related equipment in the Three Mile Island Unit 1 restart proceeding. The information UCS receives through this request will be used in further licensing and regulatory actions, of which UCS will inform its sponsors and the public through its newsletters and other publications.

Dupe of ~~8204220118~~

If, for any reason, access to all or part of this information is denied, please describe the deleted matter in detail and specify the statutory basis for exempting the material. Please separately state your reasons for not invoking your discretionary powers to release the requested information in the public interest. Such statement will be helpful in deciding whether to appeal an adverse determination.

Thank you for your prompt reply.

Sincerely,

Diane Curran

Diane Curran

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NOV 2 0 1981

MEMORANDUM FOR: All NRR Personnel

FROM: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

SUBJECT: STANDARD DEFINITIONS FOR COMMONLY-USED SAFETY CLASSIFICATION TERMS

Litigation of one of the principal issues in the TMI-1 Restart Hearing brought to light the fact that there is not complete consistency among all elements of the NRR staff in the application of safety classification terms used frequently in the conduct of NRR's safety review and licensing activities. More specifically, it appears that terms "important to safety," "safety grade," and "safety-related" have been used at times interchangeably, or in ways not completely consistent with the definitions and usage of such terms in the regulations, and which do not fully reflect the intent of the regulations or current licensing practice.

Efforts have been underway for some months now to develop guidance for the consistent usage of these terms. These efforts have included: (a) review of a large number of Reg Guides and SRP's, in conjunction with parts of the regulations upon which they are based, for consistency in the application of safety classification terminology, (2) extensive discussions among cognizant NRR, RES (Stds. Devel.) and ELD representatives regarding proper interpretation and application of such terms, including consideration of alternative "standard" definitions and (3) consultation with the cognizant ACRS Subcommittee regarding these matters, and consideration by the full ACRS as well.

As a result of these efforts, I am endorsing and prescribing for use by all NRR personnel the standard definitions set forth in the enclosure to this letter. It should be noted that in connection with long-term efforts to develop means for ranking reactor plant systems with respect to degree of importance to safety, and in connection with related efforts to develop a graded Q.A. approach in reactor licensing, the general question of safety classifications and safety classification terminologies will be reexamined; and this could result in changes to the definitions set forth in the enclosure or perhaps in development of a completely new scheme in this regard. For the time being, however, the definitions in the enclosure should be considered "standard" and should be applied consistently by all NRR personnel in all aspects of our safety review and licensing activities and should be appropriately reflected in our regulatory guidance documents.

8212200448
PDR

It is expected that minor editorial revisions will have to be made to some existing Reg Guides and SRP's in order to make their wording consistent with these definitions. You should review the regulatory guidance documents within your purview in this regard and recommend the necessary changes; it is not expected that this will involve extensive revision efforts. I want to make clear that my interest here is only in establishing consistency in the language used by all cognizant groups within NRR in expressing our technical requirements. It is not my intention by this action to dictate new technical requirements, to modify existing technical requirements, or to broaden the existing scope of NRR licensing review.

Harold Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure:
Definition of Terms

DEFINITION OF TERMS

Important to Safety

- Definition - From 10 CFR 50, Appendix A (General Design Criteria) - see first paragraph of "Introduction."

  "Those structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public."

- Encompasses the broad class of plant features, covered (not necessarily explicitly) in the General Design Criteria, that contribute in important way to safe operation and protection of the public in all phases and aspects of facility operation (i.e., normal operation and transient control as well as accident mitigation).

- Includes Safety-Grade (or Safety-Related) as a subset.

Safety-Related

- Definition - From 10 CFR 100, Appendix A - see sections III.(c), VI.a.(1), and VI.b.(3).

  Those structure, systems, or components designed to remain functional for the SSE (also termed 'safety features') necessary to assure required safety functions, i.e.:

  (1) the integrity of the reactor coolant pressure boundary;

  (2) the capability to shut down the reactor and maintain it in a safe shutdown condition; or

  (3) the capability to prevent or mitigate the consequences of accidents which could result in potential off-site exposures comparable to the guideline exposures of this part.

- Subset of "Important to Safety"

- Regulatory Guide 1.29 providesan LWR-generic, function-oriented listing of "safety-related" structures, systems, and components needed to provide or perform required safety functions. Additional information (e.g., NSSS type, BOP design A-E, etc.) is needed to generate the complete listing of safety-related SSC's for any specific facility.

  Note: The term "safety-related" also appears in 10 CFR 50, Appendix B (Q.A. Program Requirements); however, in that context it is framed in somewhat different language than its definition in 10 CFR 100, Appendix A. That difference in language between the two appendices has contributed to confusion and misunderstanding regarding the exact meaning of "safety-related" and its relationship to "important to safety" and "safety-grade." A revision to the language of Appendix B has been proposed to clarify this situation and remove any ambiquity in the meaning of these terms.

Enclosure

Safety-Grade

- Term not used explicitly in regulations but widely used/applied by staff and industry in safety review process.
- Equivalent to "Safety-Related," i.e., both terms apply to the same subset of the broad class "Important to Safety."

DEFINITION OF TERMS

Important to Safety

. Definition (10 CFR 50 Appendix A - General Design Criteria)

"Those structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public"

. Encompasses the broad class of plant features, referred to (not necessarily explicitly) in the General Design Criteria, that contribute in important ways to safe operation and protection of the public in all phases and aspects of facility operation (i.e., normal operation, transient control, accident mitigation) including:

- Systems and components provided for normal operation and control of the plant, whose failure could directly cause or aggravate an accident (also could be called upon to help mitigate the consequences of accidents).

  Examples: Main Steam, Condensate and Main Feedwater, Reactivity Control, Primary Pressure Control, Main Turbine and Condenser, Major Plant Control Circulating Water System

- Major Casualty Control Systems

  Examples: Fire Protection, Emergency Lighting, Emergency Comm.

- Systems and components provided to contain and control radioactive waste/ effluents resulting from plant operation

  Examples: Radwaste Systems, Effluent Treatment Systems and Spent Fuel Storage/Cooling systems and structures

- Reactor Coolant Pressure Boundary

- Vital safety systems and Engineered Safety Features relied upon to control and mitigate consequences design basis accidents and other design basis events (e.g. failure of systems/components provided for normal operation and plant control, LOCA, ATWS, SSE, etc.)

  Examples: RPS, ECCS, RHR, AFW, Containment Spray, Containment Isolation, etc.

- Structures relied upon to protect vital safety systems and ESF's from effects of design basis accidents and design basis events (including those involving noctural phenomena, e.g. SSE, wind, flood, etc.)

  Examples: Primary Containment Bldg, other Seismic Category I structures

- Auxiliary and support systems <u>required</u> for the operation of vital safety systems and ESF's

  Examples: Component cooling water, emergency ac/dc power, emergency air, control room ventilation, etc.

. Includes Safety-Grade or Safety-Related as a subset

<u>Safety-Grade</u>

. Term not used explicitly in regulations, but widely used/applied by staff and industry in safety review process

. "Definition" (10 CFR 100 Appendix A Seismic and Geologic Siting Criteria)

- Those structures, systems, and components (designed to remain functional for the SSE) necessary to assure:

  (1) The integrity of the reactor coolant pressure boundary

  (2) The capability to shutdown the reactor and maintain it in a safe shutdown condition, or

  (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of this part.

. Subset of "Important to Safety"; equivalent to "Safety-Related"

<u>Safety-Related</u>

. Defined in regulations <u>only</u> in context of Q.A. program criteria

. Used widely by staff in many areas of safety review otther than Q.A. program review

. Definitions:

- 10 CFR 50 Appendix B - Quality Assurance Criteria

  "Structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public"

- SRP Section 7.1, III - Review Procedures

  "Safety-Related systems fall into three categories: basic safety systems, auxiliary supporting systems, and other systems important to safety.

  (1) Basic safety systems on those that directly perform a protective function (e.g. RPS, ECCS, other ESF's)

  (2) Auxiliary systems are those that must function to enable operation of the basic safety systems (e.g. component cooling, service water, ventilation and electric power that serve RPS & ESF's)

  (3) Other systems important to safety are those systems which operate to reduce the probability of occurrence of specific accidents, or to maintain the plant (including other safety systems) within the envelope of operating conditions postulated in the accident analyses as being required to assure full protection capability (e.g. cold loop startup interlocks, accum. tank isol. interlocks/P.I./alm., plant status indic./alm. necessary for initiating manual protective actions, etc)"

. In practice, staff has applied this term in a way that establishes it as generally equivalent to (if not identically equal to) "safety-grade." This is true in the Q.A. program review practices in the past, as in other safety review contexts.

 Obtained by Les Rubenstein on his trip to simulator at Sequoyah.

3.13 DEFINITIONS AND INTERPRETATION OF TERMS

This section defines terms which are used in describing the nuclear safety design of the plant. This section also describes the correlations TVA has made among terms contained in the NRC SAR Format, Regulatory Guides, Code of Federal Regulations, and ANS standards.

3.13.1 Definitions

The definitions given in this section are used within TVA to clarify and distinguish between the requirements placed on mechanical systems and components which are "safety related" or "important to nuclear safety." These definitions, which generally conform to those used within the nuclear industry as described in Section 3.13.2 clarify the relative degrees of importance to nuclear safety, identifying possible differences in design requirements.

a. Safety-Related Plant Features - Those structures, systems, and components which are important to safety because they perform either a primary or a secondary safety function.

   1. Primary Safety Function - That function of a structure, system, or component which is necessary to assure: (1) integrity of the reactor coolant pressure boundary, (2) capability to shut down the reactor and maintain it in a safe shutdown condition, or (3) capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures to a significant fraction of the guideline exposures of 10 CFR Part 100, reference [1]. Also included are supporting and auxiliary systems which must function to provide such assurance. 16

   2. Secondary Safety Function - That function of a portion of a structure, system, or component which must either: (1) retain limited structural integrity because its failure could jeopardize to an unacceptable extent the achievement of a primary safety function or because it forms an interface between Seismic Category I and non-Seismic Category I plant features or (2) perform a mechanical motion which is not required in the performance of a primary safety function but whose failure to act or unwanted action could jeopardize to an unacceptable extent the achievement of a primary safety function.

b. Seismic Category I - Those structures, systems, or components which perform primary safety functions. They are designed and constructed to assure achievement of their primary saf . unctions at all times including a concurrent Sa. S.utdown Earthquake (SSE).

c. Seismic Category I(L) - Those portions of structure, systems, or components which perform secondary safety functions to the extent that only limited structural integrity is required. They are designated as Seismic Category I(L) (i.e., limited seismic requirements) and are designed and constructed to assure achievement of their limited structural integrity at all times including a concurrent SSE. The limited structural integrity requirements associated with these plant features are either position retention (remain in place) or pressure boundary retention.

3.13.2 Correlation of Definitions

TVA has attempted to use the definitions in Section 3.13.1, throughout the FSAR. However, current practices with the nuclear industry utilize several different terms to identify plant features that are safety-related. This makes it difficult to maintain consistency throughout the FSAR. Therefore, TVA has assumed that the following terms which have been used in different places by the nuclear industry and the NRC are generally equivalent with the term "Safety-Related Plant Features" defined above. This general equivalency should not be construed as an endorsement of predetermined requirements associated with the following terms. The only requirements are those prescribed by TVA using TVA definitions. (Note: Some of the terms apply to only a portion of the broad structures, systems, and components and others refer to functions, however, the basic comment is generally the same). 16

a. "Safety-related structures, systems, and components" (i.e., plant features) as defined in 10 CFR Part 100, Appendix A, reference [1], and RG 1.29, reference [2], Section 3.2.2 in RG 1.70, reference [8].

b. "Structures, systems, and components important to safety," as defined in RG 1.29, reference [2], and RG 1.105, reference [3], and as used in 10 CFR Part 100, Appendix A, reference [1], and the General Design Criteria for Nuclear Power Plants, reference [4], and Sections 3.2.1 and 3.2.2 in RG 1.70, reference [8].

c. "Safety system," as defined in ANSI N18.2a, reference [5], and ANSI N212, reference [6].

d. "Structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public," as defined in 10 CFR Part 50, Appendix B, reference [7].

e. Basic components as defined in 10 CFR Part 21, paragraph 21.3(a), reference [9].

As stated earlier TVA has interpreted as equivalent the expressions "safety-related plant features," and "structures, systems, and components important to safety." In one exception to the general pattern above, Standard Review Plan 7.1, Section III, reference [10], gives a somewhat different definition of safety-related systems. This definition states:

"Safety-related systems fall into three categories: basic safety systems, auxiliary supporting systems, and other systems important to safety."

This definition indicates that "important to safety" is a subset of safety-related systems rather than being generally equivalent to safety related systems as indicated in items a and b of this section.

This section indicates a number of different terms are being used to define the single concept of safety-related plant features. TVA hopes that by stating the terms we have assumed are equivalent this will aid in clarifying the material presented in this FSAR. 16

REFERENCES

1. Nuclear Regulatory Commission. Seismic and Geologic Criteria for Nuclear Power Plants. Title 10, Code of Federal Regulations, Part 100 (10 CFR Part 100), Appendix A. Washington: GPO.

2. Nuclear Regulatory Commission. Seismic Design Classification. Regulatory Guide 1.29, Rev. 2. Washington: NRC, February 1976.

3. Nuclear Regulatory Commission. Instrument Setpoints. Regulatory Guide 1.105. Washington: NRC, November 1976.

4. Nuclear Regulatory Commission. General Design Criteria for Nuclear Power Plants. Title 10, Code of Federal Regulations, Part 50 (10 CFR Part 50), Appendix A. Washington: GPO.

5. American Nuclear Society. Revision and Addendum to Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants. ANSI N18.2a-1975. Hinsdale, Ill.: American Nuclear Society, March 19, 1975.

06-30-78

6. American Nuclear Society. Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants. ANSI N212. Hinsdale, Ill.: American Nuclear Society, May 1974.

7. Nuclear Regulatory Commission. Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants. Title 10, Code of Federal Regulations, Part 50 (10 CFR Part 50), Appendix B. Washington: GPO.

8. Nuclear Regulatory Commission. Regulatory Guide 1.70, Rev. 2 Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition. Washington: NRC September 1975. 16

9. Nuclear Regulatory Commission. Reporting of Defects and Noncompliance. Title 10, Code of Federal Regulations, Part 21 (10 CFR Part 21).

10. Nuclear Regulatory Commission. Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants LWR Edition. September 1975. NUREG-75/087.

Slides for ACRS Subc. Mtg Feb. 3, 1981

BACKGROUND

1. TMI RESTART HEARING ISSUE

   ° UCS CONTENTION #14

   "[redacted] SYSTEMS AND COMPONENTS WHICH CAN EITHER CAUSE OR AGGRAVATE AN ACCIDENT OR CAN BE CALLED UPON TO MITIGATE AN ACCIDENT MUST BE IDENTIFIED AND CLASSIFIED AS COMPONENTS IMPORTANT TO SAFETY AND REQUIRED TO MEET ALL SAFETY-GRADE DESIGN CRITERIA."

   ° THIS CONTENTION ADDRESSED IN TESTIMONY BY CONRAN PROVIDED TO ACRS (DSI)

2. ROGOVIN STUDY FINDING

   ° VOL II; SEC. A.3.B, PGS. 49-50 - QUALITY ASSURANCE

   "THE NRC LACKS DEFINITIONS FOR SAFETY-RELATED AS APPLIED TO EQUIPMENT, SYSTEMS, STRUCTURES AND SO FORTH NECESSARY TO ENSURE THAT APPENDIX B QUALITY ASSURANCE STANDARDS ARE IMPLEMENTED CONSISTENTLY. THE CONSEQUENCE HAS BEEN AN AD HOC UNCONTROLLED APPLICATION OF SAFETY-RELATED REQUIREMENTS TO EQUIPMENT OUTSIDE THE REACTOR PROTECTION SYSTEM AND THE ENGINEERED SAFETY FEATURES SYSTEMS."

   ° THIS PROBLEM ADDRESSED IN ACTION PLAN I.F. (OAP)
   - EXPAND "Q" LIST TO COVER EQUIPMENT IMPORTANT TO SAFETY
   - RANK EQUIPMENT IN ORDER OF ITS IMPORTANCE TO SAFETY

DEFINITION OF TERMS

Important to Safety

- Definition (10 CFR 50 Appendix A - General Design Criteria)

  "Those structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public"

- Encompasses the broad class of plant features, covered (not necessarily explicitly) in the General Design Criteria, that contribute in important ways to safe operation and protection of the public in all phases and aspects of facility operation (i.e., normal operation and transient control, as well as accident mitigation)

- Includes Safety-Grade or Safety-Related as a subset

Safety-Grade

- Term not used explicitly in regulations, but widely used/applied by staff and industry in safety review process

- "Definition" (derived from 10 CFR 100 Appendix A)

  - Those structures, systems, and components (designed to remain functional for the SSE) necessary to assure:

    (1) The integrity of the reactor coolant pressure boundary

    (2) The capability to shutdown the reactor and maintain it in a safe shutdown condition, or

    (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of this part.

- Subset of "Important to Safety"; equivalent to "Safety-Related"

Safety-Related

- Defined in regulations (10 CFR 50 Appendix B) in context of Q.A. program requirements only

  - SSC that prevent or mitigate consequences of postulated accidents that could cause undue risk to public health & safety

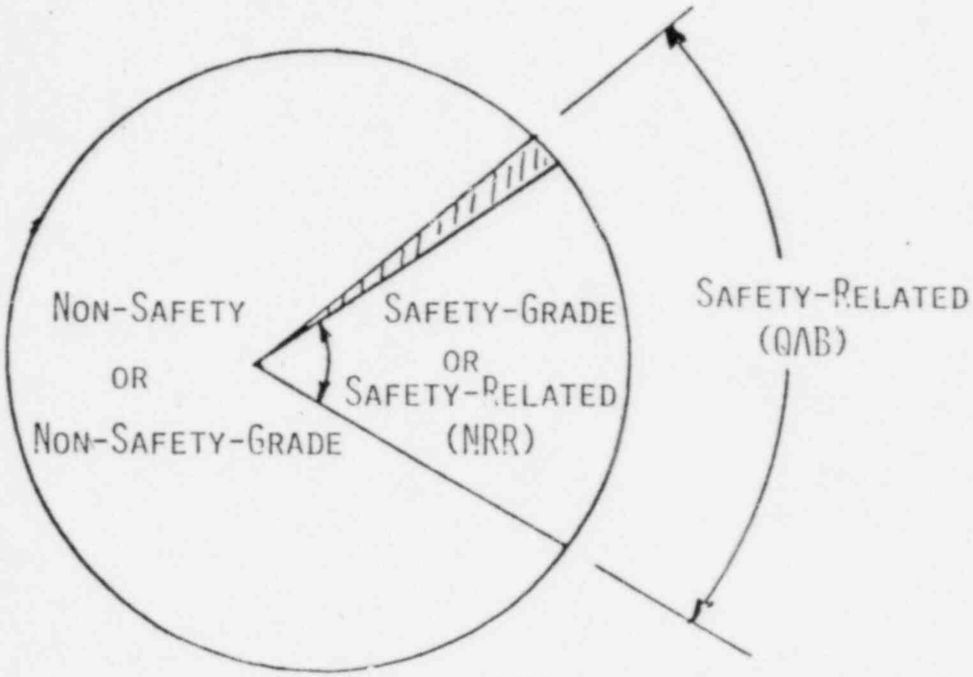
  - As applied by QAB, includes SSC identified in R.G. 1.29, plus other plant features that licensees identify as important to safety (e.g. radwaste sys.)

- Defined/used in many Reg. Guides & SRP's (e.g. SRP 7.1) in contexts other than Q.A.

  - In these contexts, term is equivalent to "safety-grade" above.

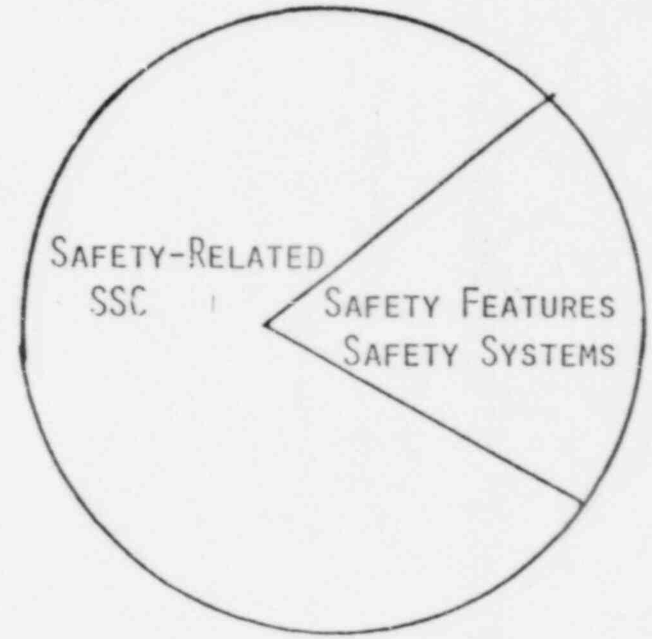
SAFETY CLASSIFICATIONS
NRC vs IAEA

IMPORTANT TO SAFETY
(ENTIRE CIRCLE)



NRR/QAB

IMPORTANT TO SAFETY
(ENTIRE CIRCLE)



IAEA

EXCERPT FROM LICENSEE'S MANUAL

Correlation of Definitions

However, current practices with the nuclear industry utilize several different terms to identify plant features that are safety-related. This makes it difficult to maintain consistency throughout the FSAR. Therefore, [redacted] has assumed that the following terms which have been used in different places by the nuclear industry and the NRC are generally equivalent with the term "Safety-Related Plant Features" defined above.

a. "Safety-related structures, systems, and components" (i.e., plant features) as defined in 10 CFR Part 100, Appendix A, reference [1], and RG 1.29, reference [2], Section 3.2.2 in RG 1.70, reference [8].

b. "Structures, systems, and components important to safety," as defined in RG 1.29, reference [2], and RG 1.105, reference [3], and as used in 10 CFR Part 100, Appendix A, reference [1], and the General Design Criteria for Nuclear Power Plants, reference [4], and Sections 3.2.1 and 3.2.2 in RG 1.70, reference [8]. WRONG!

c. "Safety system," as defined in ANSI N18.2a, reference [5], and ANSI N212, reference [6].

d. "Structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public," as defined in 10 CFR Part 50, Appendix B, reference [7].

e. Basic components as defined in 10 CFR Part 21, paragraph 21.3(a), reference [9].

REFERENCES

1. Nuclear Regulatory Commission. Seismic and Geologic Criteria for Nuclear Power Plants. Title 10, Code of Federal Regulations, Part 100 (10 CFR Part 100), Appendix A. Washington: GPO.

2. Nuclear Regulatory Commission. Seismic Design Classification. Regulatory Guide 1.29, Rev. 2. Washington: NRC, February 1976.

3. Nuclear Regulatory Commission. Instrument Setpoints. Regulatory Guide 1.105. Washington: NRC, November 1976.

4. Nuclear Regulatory Commission. General Design Criteria for Nuclear Power Plants. Title 10, Code of Federal Regulations, Part 50 (10 CFR Part 50), Appendix A. Washington: GPO.

5. American Nuclear Society. Revision and Addendum to Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants. ANSI N18.2a-1975. Hinsdale, Ill.: American Nuclear Society, March 19, 1975.

6. American Nuclear Society. Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants. ANSI N212. Hinsdale, Ill.: American Nuclear Society, May 1974.

7. Nuclear Regulatory Commission. Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants. Title 10, Code of Federal Regulations, Part 50 (10 CFR Part 50), Appendix B. Washington: GPO.

8. Nuclear Regulatory Commission. Regulatory Guide 1.70, Rev. 2 Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition. Washington: NRC September 1975.

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10. Nuclear Regulatory Commission. Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants LWR Edition. September 1975. NUREG-75/087.

SAFETY CLASSIFICATION vs APPLICABLE QUALITY STANDARDS

CHARACTERISTICS	SAFETY-GRADE	IMP. TO SAFETY (NOT SAFETY-GRADE)
Design/Fabr.	ASME Sec. III, Class 1, 2 & 3 (Reactor Coolant Press. Boundary, ECCS, AFW)	ASME Sec. VIII & ANSI B 31.1.0 (Radwaste System)
Redundancy/Diversity	Meets Single Failure Criterion (PPS, ESFAS, ESF's)	Not req'd to meet SFC (Reactor Control, ICS)
Seismic Design Classification	Seismic Category I (Containment Bldg., RCPB, ECCS, AFW)	"Seismic II/III - Bechtel" (Condensate & Feedwater, Circ water, MS)
Pwr. Source	Class IE (Critical Pwr Bus, Emer. Pwr for ESF's)	Non-Class IE (Non-essential power buss)
Quality Group Classif. (R.G. 1.26	Class A, B, C	Class D
Q.A. Program Req'ts	10 CFR 50 App. B	Will be graded Q. A. program

LEVEL OF QUALITY (RELIABILITY) REQUIRED ⟶

← DEGREE OF IMPORTANCE TO SAFETY

RPV (ASME Sec. III Class 1) —— Qual. Gr A

RCPB (ASME Sec. III Class 1) —— Qual. Gr A

RPS (IEEE 279)

Class IE Pwr (Crit. Pwr. Bus)

ECCS, RHR, Cont. Spray (ASME Sec. III Class 2) —— Qual. Gr. B

ESFAS

Class IE (Interuptable Emer. Pwr)

AFW (ASME Sec. III Class 3) —— Qual Gr. C

ESF Support Sys.

CCW, Serv. Water (ASME Sec. III Class 3) —— Qual. Gr. C

Spent Fuel Cooling Sys.

Safety - Grade

Reactor Coolant Pumps

Reactivity Control Sys.

RCS Press. Contr. Sys. (PORV, PZR Htrs)

Integrated Contr. Sys.

Normal Heat Transfer Sys.

Turbine and Condenser

Main Steam Sys.

Condensate & Feedwater Sys.

Secondary Plant Contr. Sys

Fire Protection Sys.

Rad Waste Mgmt. Sys. (ASME Sec. VIII Div. 1) —— Qual. Gr. D

Casualty Control Sys.

Emer. Lighting

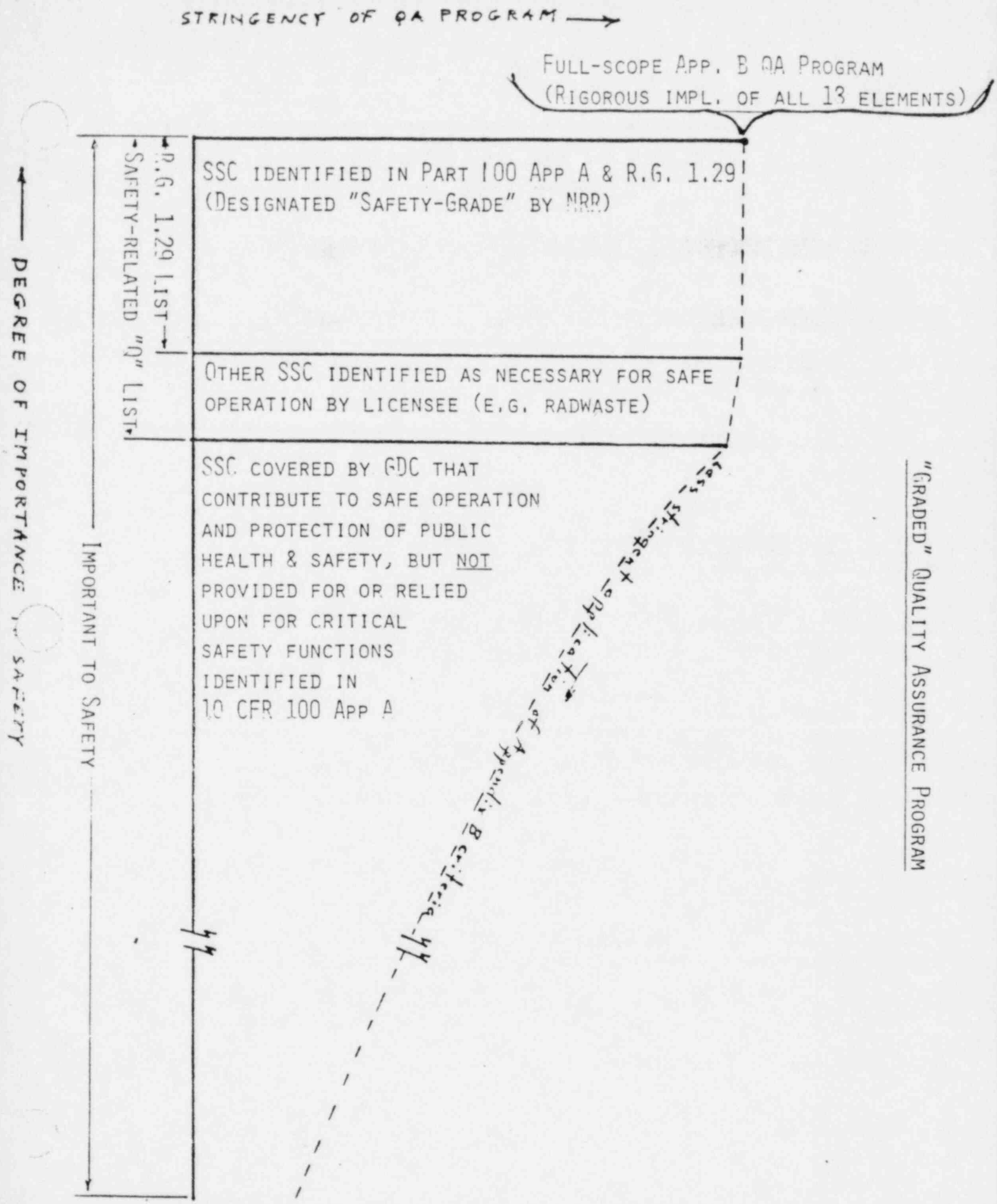
Emer. Communications

Important to Safety

ELEMENTS OF QA PROGRAM

(APPENDIX B)

1. ORGANIZATION
2. QUALITY ASSURANCE PROGRAM
3. DESIGN CONTROL
4. PROCUREMENT DOCUMENT CONTROL
5. INSTRUCTIONS, PROCEDURES, AND DRAWINGS
6. DOCUMENT CONTROL
7. CONTROL OF PURCHASED MATERIALS, EQUIPMENT, AND SERVICES
8. IDENTIFICATION AND CONTROL OF MAT'LS, PARTS, AND COMPONENTS
9. CONTROL OF SPECIAL PROCESSES
10. INSPECTION
11. TEST CONTROL
12. CONTROL OF MEASURING AND TEST EQUIPMENT
13. HANDLING, STORAGE AND SHIPPING
14. INSPECTION, TEST AND OPERATING STATUS
15. NON-CONFORMING MAT'LS, PARTS, OR COMPONENTS
16. CORRECTIVE ACTION
17. QUALITY ASSURANCE RECORDS
18. AUDITS


STRINGENCY OF QA PROGRAM →
FULL-SCOPE APP. B QA PROGRAM
(RIGOROUS IMPL. OF ALL 13 ELEMENTS)
SSC IDENTIFIED IN PART 100 APP A & R.G. 1.29
(DESIGNATED "SAFETY-GRADE" BY NRR)
OTHER SSC IDENTIFIED AS NECESSARY FOR SAFE OPERATION BY LICENSEE (E.G. RADWASTE)
SSC COVERED BY GDC THAT CONTRIBUTE TO SAFE OPERATION AND PROTECTION OF PUBLIC HEALTH & SAFETY, BUT NOT PROVIDED FOR OR RELIED UPON FOR CRITICAL SAFETY FUNCTIONS IDENTIFIED IN 10 CFR 100 APP A
R.G. 1.29 LIST
SAFETY-RELATED "Q" LIST
IMPORTANT TO SAFETY
DEGREE OF IMPORTANCE TO SAFETY
"GRADED" QUALITY ASSURANCE PROGRAM
Less stringent application of Appendix B criteria

EFFECT OF QUANTITATIVE SAFETY GOAL ON DISTINCTION AMONG VARIOUS PLANT SYSTEMS

- EVALUATION OF DOMINANT EVENT SEQUENCES COULD RESULT IN NEEDING TO UPGRADE THE SIGNIFICANT FAILURE CONTRIBUTORS (E.G. NON-SAFETY SYSTEMS) TO MEET SAFETY GOALS.

- MORE EMPHASIS ON EVALUATION OF NON-SAFETY SERVICE SYSTEM INTERFACE EFFECTS — COMMON CAUSE OR LINKING FAILURES.

UCS Interrogatory 150

UCS 150 ... of Interrogatories

Explain the present Staff position on UCS Contention 14.

Response

As noted in Reference 1,* "Current practice in the licensing of nuclear power plants is to apply design requirements to one class of components, equipment, systems, and structures, the so-called safety-grade class, but not to another nonsafety-grade class. This system of classification is based on the premise that things can be classified either as important to safety (that is, the function is credited in the analysis of a design basis event or is specified in the regulations) or not important to safety." Reference 1 also states that "there is a general requirement that failure of nonsafety-grade equipment or structures should not initiate or aggravate an accident" and that "the term 'failure' when applied to nonsafety-grade equipment has generally been defined as 'failure to operate upon demand'". The above general requirement is not in the Standard Review Plan (SRP). For example, Reference 2, page 17, cites Section 7.7 of the SRP which gives a general guideline for reactor control systems** However, as noted in Reference 2, there are no guidelines in the SRP that generally apply to the many other nonsafety systems. It is also stated in Reference 2 that "the Task Force will reassess this approach to consider the need to expand the regulatory coverage to other systems such as the power conversion system and the auxiliary systems."

In References 1 and 2, the staff has presented detailed discussions of possible weaknesses of this "current practice." The quote from Reference 2 given in UCS

*References 1-5 can be found after the response to Interrogatory 153.

**References 3, 4, and 5 contain discussions of the staff position with respect to control systems and credit for nonsafety-grade in the steam line break accident.

Contention 14 is a portion of these discussions and deals in part with the role played by the nonsafety-grade systems in initiating the series of events involved in the TMI-2 accident and used in mitigation of the accident in ways not previously considered in the safety analysis.

In the discussion of recommendations with respect to improvements in plant design, the Lessons Learned Task Force stated (Ref. 1, page 3-3):

> "The Task Force concludes that comprehensive studies of the interaction of non-safety-grade components, equipment, systems and structures with safety systems and the effects of these interactions during normal operation, transients, and accidents need to be made by all licensees and license applicants (see Recommendation 9). This would constitute a significant alteration of the current unresolved safety issue concerning systems interaction. The Office of Standards Development has previously been requested to develop a Regulatory Guide that would specify generic requirements for some safety-related systems that do not presently fall within the safety-grade classification. This effort would have to be closely coordinated with the study by licensees that we are now recommending. In the interim, the effects of the abnormal conditions that accompany transients and accidents on the operation and failure of non-safety-grade items should be reviewed by all licensees to determine if there are any probable adverse interactions. The extent of simultaneous interactions considered in this review should reflect the number of non-safety systems simultaneously exposed to conditions for which they were not designed. Equipment identified as the cause of unacceptable interactions should be appropriately modified to reduce the probability of

that interaction, or the safety system that is adversely affected should be modified to cope with the interaction. In either event, operating procedures and operator training must be expanded to include consideration of the possible permutations and combinations of non-safety-grade system interactions with safety systems."

It is expected that the results of studies performed in accordance with Recommendations 8 and 9 of Reference 1 will be reflected in changes to the Standard Review Plan dealing with the staff position on nonsafety-grade systems.

In summary, as quoted in UCS Contention 14, the NRC staff has identified a need to examine whether further requirements may be necessary to improve the current capability for use of nonsafety systems during transient or accident situations. This need has been translated into the program described in Action Plan IIC of NUREG-0660. The present position of the Staff is that satisfactory completion of the short-term actions and reasonable progress in the long-term actions identified in the Order will provide reasonable assurance that the facility can be operated without endangering the health and safety of the public.

Interrogatory 151

Does the current position differ from the position of the Staff in any prior cases? If so, identify the case(s), explain the prior position, and explain the basis for the change in position.

Response

The current position differs from the previous position with respect to the staff actions already taken and planned as discussed in the response to Interrogatory 150 and the reference cited in that response. The basis for the changes in position are discussed in detail in the references.

UCS Interrogatory 152

Identify any members of the Staff who dissent from the present Staff position on UCS Contention 14. Explain the reasons for which any Staff members dissented from the present Staff position on UCS Contention 14.

Response

Mr. Demetrois Basdekas raised technical issues involving nonsafety-grade equipment which are discussed in detail in References 3, 4, and 5. No other members of the staff are known who dissent from the present staff position on UCS Contention 14.

UCS Interrogatory 153

Identify the specific sections and page numbers of the SER and/or FSAR for TMI, Unit 1, which are relied upon in formulating the Staff position on UCS Contention 14.

Response

The SER and FSAR for TMI, Unit 1 were not relied on in formulating the staff position on UCS Contention 14.

References

1. NUREG-0585, "TMI-2 Lessons Learned Task Force Final Report," October 1979.

2. NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," July 1979.

3. Memo, H. Denton, NRC, to Commissioner J. F. Ahearne , NRC, "Safety Implications of Control Systems and Plant Dynamics," October 22, 1979.

4. NUREG-0153, "Staff Discussion of Twelve Additional Technical Issues Raised by Responses to November 3, 1976 Memorandum From Director, NRR to NRR Staff," December 1976.

5. NUREG-0138, "Staff Discussion of 15 Technical Issues Listed in Attachment to November 3, 1976 Memo from Director, NRR to NRR Staff," November 1976.

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Interrogatory 155

Identify **all systems** and components presently classified as non-safety-related which contributed to the cause of the TMI-2 accident, aggravated the accident or were called upon to attempt to mitigate the accident. Discuss their role in the accident sequence.

Response

A complete list of all the non-safety related systems and components ~~which contributed~~ to the cause of the accident, aggravated the accident or were called upon to mitigate the accident has not been generated by the staff. As complete a list of components and systems available that were utilized and a discussion of their role in the accident sequence is contained in the following documents:

TMI-1

- Investigation into the March 28, 1979 Three Mile Accident by the Office of Inspection and Enforcement, NUREG-0600, August, 1979.
- Technical Staff Analysis Report on Summary Sequence of Events to the President's Commission on the Accident at Three Mile Island, October 1979.
- A Report to the Commissioners and to the Public, Volume II, Part II, (Draft). Mitchell Rogavin.
- Analysis of Three Mile Island Unit 2 Accident - NSAC-1, July 1979.

Interrogatory 156

Does the staff agree that some systems and components presently classified as non-safety-related can have an adverse effect on the core because they can directly or indirectly affect temperature, pressure, flow and/or reactvity?

Response

The staff agrees that some systems and components presently classified as non-safety-related can have an effect on the core because they can directly or indirectly affect temperature, pressure, flow and/or reactivity. However, the extent or degree to which these non-safety-related systems or components can adversely affect the core is still an on going review as identified in Draft NUREG-0660. On September 19, 1979 all licensees of light water reactors were requested to determine if an unreviewed safety question related to interaction of safety grade and non-safety grade equipment existed at their nuclear plants. We were concerned that consequential control system failures following a high energy line break (HELB) might cause consequences of the HELB to become more severe than previously expected. The staff has done a preliminary review of the licensee responses and the review to date has not specifically identified safety problems; that is, no event sequence clearly leads to an unacceptable consequence. (NRC Memorandum to D. G. Eisenhut, Acting Director, Division of Operating Reactors, from P. S. Check, Reactor Safety Branch, Division of Operating Reactor, dated December 17, 1979).

Non-safety systems and components ~~related to the core cooling systems are~~ the Auxiliary Feedwater System (AFW) and pressurizer power operated relief valves (PORV). ~~The AFW is related to the core cooling system~~ for mitigating small break accidents. For certain small breaks the steam generators remove heat from the primary system to lower the pressure to the high pressure

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emergency core cooling system (ECCS) automatic initiation setpoint. However, two other options are available to the operator as follows: 1) the operator can manually initiate the PORV to lower the primary system pressure to the automatic initiation setpoint of the high pressure injection; or 2) the operator can manually initiate the high pressure injection as discussed in NUREG-0565.

Interrogatory 157

Which of the short and/or long term measures recommended by the staff are directed toward preventing adverse effects on the integrity of the core caused by non-safety-related systems and components? How will these measures correct the deficiencies identified in NUREG-0578, Section 3.2? What is their schedule for implementation?

Response

The answer to Interrogatory 157 can be found in January 11, 1980, Status Report on the Evaluation of Licensee's Compliance with the NRC Order dated August 9, 1979 Metropolitan Edison Company, et al., Three Mile Island Nuclear Station Unit 1 Docket No. 50-289. These short and/or long term answers recommended by the staff are listed in a Status Summary Table on pages B-4 through B-8. The short term items that are directed toward preventing adverse effects on the integrity of the core caused by non-safety related systems are the following items: all items 1a through 1e; IE Bulletins 79-05A - all items except 79-05A-6, 9, 10, and 12; IE Bulletin 79-05B, all items except 79-05B-6; IE Bulletin 79-05C all items; Item 8 Lessons Learned Short Term items 2.1.1, 2.1.3.a, 2.1.3.b, 2.1.7.a, 2.1.7.b and 2.1.9.

The long term items that are directed toward preventing adverse effects on the integrity of the core caused by non-safety related systems are long term items 1 and 2.

measures identified in this status report are directed towards improving the operators understanding and response to small break LOCA and undercooling vents that could lead to core damage and, through training utilizing the pro- ed developed, enable the operator to mitigate the consequences of the event ith the components and systems available. Other measures identified in the tatus report which correct the deficiencies identified in Section 3.2 of NUREG-0578 re: improved reliability, quality, and availability of components and systems, utomation of AFW system startup on loss of main feedwater, and reactor hutdown on loss of main feedwater and turbine trip; and the addition of instrumentation o aid the operator in the detection of degraded conditions and upgrading of the reliability of existing instrumentation that would aid the operator in assessing lant status during transients.

he status of the applicable short and long term measures identfied that are irected towards preventing adverse effects on the integrity of the core are given n the report referenced in the first paragraph of this response in the sections or ponding to each of the measures identified.

Interrogatory 158

Does the staff propose to classify the reactor coolant pumps as safety-related? If not, explain your answer fully.

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Response

The NRC does not intend to reclassify the reactor coolant pumps as safety related. The reactor coolant pumps are not required to mitigate the consequences of an accident or transient. How about "failure to stop" or "restart"

Interrogatory 159

Does the staff propose to classify the steam generators as safety-related? If not, explain your answer fully.

Response

The steam generators are classified as safety related. Please note that the staff's response to this question set forth in its January 10, 1980 preliminary response is incorrect.

Interrogatory 160

Which of the General Design Criteria applying to safety-related systems, structures and components does the staff propose to apply to the reactor coolant pumps? If the staff proposes to apply less than all of them, explain why those not applied have been excluded.

Response

The reactor coolant pump housing forms part of the reactor coolant system pressure boundary and therefore has to comply to the requirements of GDC 1, 2, 3, 4, 13, 14 and 31 which are all the GDC which apply to the reactor coolant pressure boundary. The pump flywheel integrity has to comply to the requirements of GDC 4. These components have always been required to conform to the GDC identified.

Interrogatory 161

Answer the same question as 160 above with regard to the steam generators.

Response

The steam generators form part of the reactor coolant system pressure boundary and therefore have to comply to the requirements of GDC 1, 2, 3, 4, 13, 14 and 31, which are all the GDC which apply to the reactor coolant pressure boundary.

Interrogatory 161

Answer the same question as 160 above with regard to the steam generators.

Response

The steam generators form part of the reactor coolant system pressure boundary and therefore have to comply to the requirements of GDC 1,2, 3, 4, 13, 14 and 31, which are all the GDC which apply to the reactor coolant pressure boundary.

Interrogatory 162

Explain how the Staff can assure the adequate protection of the public health and safety when systems and components, which are classified as non-safety-related but already have been demonstrated to contribute to the aggravation or mitigation of the TMI-2 accident have not been classified as components important to safety and required to meet all safety grade design criteria.

Response

The TMI-2 accident involved a sequence of events which went well beyond those considered in the current design basis accidents and involved both safety-grade systems and nonsafety-grade systems. Reviews of the TMI-2 accident by the TMI-2 Lessons Learned Task Force and evaluations of operating plants by the Bulletins and Orders Task Force resulted in a large number of short-term recommendations and actions dealing with plant analysis, design and operation which provide substantial additional protection for the public health and safety. In view of the large number of staff actions already taken and planned, reference is made to the following staff reports for detailed discussions of these actions: NUREG-0578, NUREG-0565, NUREG-0585, NUREG-0626, NUREG-0635, NUREG-0611, and NUREG-0660.

With respect to this interrogatory which deals with nonsafety-grade equipment, the staff recognized the need to upgrade some systems which had not previously been considered part of the reactor protection system or engineered safety features. A number of short-term actions were taken to provide additional protection for the public health and safety. Actions taken with respect to the pressurizer PORVs, block valves and associated instrumentation and controls are discussed in the response to Interrogatory 47. In addition, actions were taken

(a) provide redundant emergency power for the pressurizer level indication instrument channels and part of the pressurizer heaters (NUREG-0578, Section 2.1.1) and (b) provide automatic initiation of the auxiliary feedwater system (NUREG-0578, Section 2.1.7).

Specific changes to upgrade the auxiliary feedwater system in TMI-1 are discussed in "Evaluation of Licensee's Compliance with the NRC Order Dated August 9, 1979, Metropolitan Edison, et al., Three Mile Island Nuclear Station Unit 1, Docket No. 50-289."

In addition to these short-term actions, the staff recognized the need for a long-term review of safety classifications and qualifications (NUREG-0585, Recommendation 9). As noted in NUREG-0585, this review would include all nonsafety components, systems, and structures and would include conditions of normal operation, anticipated operational occurrences, and design basis accidents. Work on this review will be done under Task II.C of NUREG-0660.

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the subject matter questioned. In lieu thereof, at Staff's option, a copy of each such document and study may be attached to the answer.

D. Explain whether the Staff and/or any independent contractor are presently engaged in or intend to engage in further research or work which may bear on the issues covered in the interrogatory. If so, please identify such research or work and the person(s) responsible therefor.

E. 1) Identify the expert(s), if any, whom the Staff intends to have testify on the subject matter covered in the interrogatory. State the qualifications of each such expert. 2) Present a summary of each expert's proposed testimony on each UCS Contention. 3) Identify all cases in which any such expert has previously testified and state the subject matter of such testimony.

Answer each of the following five preliminary questions for <u>every</u> Contention:

1. Explain the present Staff position on UCS Contention N (N=1-20).
2. Does the current position differ from the position of the Staff in any prior cases? If so, identify the case(s), explain the prior position, and

explain the basis for the change in position.

3. Identify any members of the Staff who dissent from the present Staff position on UCS Contention N. Explain the reasons for which any Staff members dissented from the present Staff position on UCS Contention N.
4. Identify the specific sections and page numbers of the SER and/or FSAR for TMI, Unit 1, which are relied upon in formulating the Staff position on UCS Contention N.
5. Identify all sections and page numbers of the SER and/or FSAR which contain subject matter pertaining to UCS Contention N.

CONTENTION 1

1-5. Answer each of the five preliminary questions with respect to Contention 1 and number the answers 1-5.

6. Explain whether or not natural circulation is an adequate means for removing decay heat from the reactor core in the event of a small loss-of-coolant-accident ("LOCA").
7. Explain in detail which of the short or long term measures recommended by the Director of Nuclear Reactor Regulation will prevent the formation of voids in the reactor cooling system as occurred at TMI-2.
8. Does the Staff take the position that implementation of the short term and/or long term

to reflect developments since 1971 and to accord more fully with current Staff policy in this area. . ." (Sl.op. at 9).

a. Have any such recommendations been provided to the Commission? If so, supply them.

b. If no such recommendations have yet been provided to the Commission, why not? When will they be provided?

c. Provide any draft memoranda or recommendations which have been prepared in response to the Commission's directive

d. Identify the Staff members who have been assigned to work on a response to the Commission's directive.

e. What are the "developments since 1971?"

f. What is "current Staff policy in this area?"

CONTENTION 14

150-154. Answer each of the five preliminary questions with respect to Contention 14 and number the answers 150-154.

155. Identify all systems and components presently classified as non-safety-related which contributed to the cause of the TMI-2 accident, aggravated the accident or were called upon to attempt to mitigate the accident. Discuss their role

156. Does the Staff agree that some systems and components presently classified as non-safety-related can have an adverse effect on the core because they can directly or indirectly affect temperature, pressure, flow and/or reactivity? Identify all such systems and components related to the core cooling systems.

157. Which of the short and/or long term measures recommended by the Staff are directed toward preventing adverse effects on the integrity of the core caused by non-safety-related systems and components? How will these measures correct the deficiencies identified in NUREG-0578, Section 3.2? What is their schedule for implementation?

158. Does the Staff propose to classify the reactor coolant pumps as safety-related? If not, explain your answer fully.

159. Does the Staff propose to classify the steam generators as safety-related? If not, explain your answer fully.

160. Which of the General Design Criteria applying to safety-related systems, structures and components does the Staff propose to apply to the reactor coolant pumps? If the Staff proposes to apply less than all of them, explain why those not applied have been excluded.

161. Answer the same question as #160 above with regard to the steam generators.

162. Explain how the Staff can assure the adequate protection of the public health and safety when systems and components, which are classified as non-safety-related but already have been demonstrated to contribute to the aggravation of mitigation of the TMI-2 accident have not been classified as components important to safety and required to meet all safety grade design criteria.

CONTENTION 16

163-167. Answer each of the five preliminary questions with respect to Contention 16 and number the answers 163-167.

168. What basic assumptions and methodology were used to define the 10 mile emergency planning zone for the plume exposure pathway? What were the parameters of the accident assumed to occur? What assumptions were made about meteorology?

169. Provide the NRC and EPA Staff input into NUREG-0396, EPA 520/1-78-016. "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans. . ." This should include draft and final memoranda, reports and other documents.

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148., 149.	The Staff objects to these interrogatories on the ground that they probe into generic matters not specifically related to this proceeding. To the extent that UCS is inquiring into whether the Staff will recommend that any environmental consequences of Class 9 accidents be considered in this proceeding, that answer will be provided in response to the Board's "Frist Special Prehearing Conference Order" dated 12/18/79.
Contention 14	
155. - 157.	Matter is under review; response will be in supplemental testimony.
158. - 159.	No; the explanation for the response will be in supplemental testimony.
160., 161.	Response will be set forth in supplemental testimony.
162.	Matter is under review; response will be in SE and its supplements, as well as supplemental testimony.
Contention 16	To be provided.

Respectfully submitted,

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Counsel for NRC Staff

Lucinda Low Swartz
Counsel for NRC Staff

Dated at Bethesda, Maryland
this 10th day of January, 1980