June 30, 1966

A GUIDE FOR THE ORGANIZATION AND CONTENTS OF SAFETY ANALYSIS REPORTS

The following is intended as a guide in satisfying the requirements of § 50.34 of 10 CFR Part 50 with respect to the preparation and submission of Safety Analysis Reports and Preliminary Safety Analysis Reports by applicants for AEC licenses to construct and operate production or utilization facilities. It contains examples of the kind and general depth of information which the Commission desires in a Safety Analysis Report or Preliminary Safety Analysis Report. The examples and format are not intended as a rigid prescription. Material is included on matters not relevant to Reports for all facilities, and should be used in connection with a particular facility only to the extent pertinent. Conversely, Safety Analysis Reports and Preliminary Safety Analysis Reports are expected to include information other than that specified in the following guide if such information is necessary for the Report to meet the requirements set out in § 50.34. Reports incorporating information and/or using a format different from those set out below will be acceptable if they satisfy the requirements of § 50.34. The Report represents the principal communication between the applicant and the Commission about the facility and the manner in which it is to be operated. Effective communication is a key factor in the expeditious processing of license applications.

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GENERAL CONSIDERATIONS

This document presents guidance on the organization and contents of Safety Analysis Reports for nuclear power reactor facilities.

A. Applicability of Guide. This guide is intended to be applicable primarily to power reactors. While the examples given to illustrate desired depth of treatment and scope of coverage are drawn largely from pressurized water reactor technology, the basic system for the presentation of information is also applicable to other types of reactor facilities. What the guide does is to lay out a pattern for the presentation of information based upon the following sequence:

(1) Identification of the principal criteria for design of the facility and the design bases for those major systems and components significant to safety.

(2) Description of how it is intended that the plant be built and operated to satisfy the principal criteria and design bases.

(3) Systematic safety analysis and evaluation of the design that show plant performance objectives can be achieved and safety assured.

Such an approach to presentation of information is desired for all nuclear reactors.

B. Principal Purpose of Report. A principal purpose to be served by the preparation and submittal of a Safety Analysis Report is to inform the Commission of the nature of the facility and plans for its use. The Commission's interest is primarily that of public safety. The documentation of information is expected to be sufficient to permit a finding by the Commission's regulatory staff and the Advisory Committee on Reactor Safeguards, based on technical review of design and plans for operation, that the facility can be safely built and operated. An applicant is expected to have evaluated his facility and either conducted an experimental program or otherwise have satisfied himself from experiences with similar facilities or components thereof that the reactor can be built and operated without undue risk to the health and safety of the public. The Commission wants to understand the basis upon which this conclusion has been reached. The Safety Analysis Report is the principal document whereby this understanding is achieved.

The Commission's review of information presented in the Safety Analysis Report will be performed in the depth necessary to establish a basis for a finding in respect to safety of the proposed plant. Design methods and procedures of calculation will be examined to establish their validity. Spot checks of actual calculations and other procedures of design and analysis will be made to establish the validity of the applicant's analysis and evaluation of the design, and that the applicant has conducted his analysis and evaluation in sufficient depth and breadth to support findings in respect to safety.

C. Technical Specifications. The license issued by the Commission authorizing operation of a reactor defines in terms of "technical specifications" the broad framework within which operations are to be conducted. It is the intent that such conditions as are specified be largely safety-oriented. Therefore, the reasons for their inclusion and the bases for such specific limits as may be imposed, should be included in the Safety Analysis Report. In addition to supplying that backup information justifying the necessity for and sufficiency of the technical specifications proposed, the applicant should append the technical specifications themselves to the Report.

D. Additional Information. The worth of this guide will, of couse, depend to a great extent on the manner in which it is used. The users are urged to keep in mind the objectives and reasons for the Report as discussed above. The suggested examples are intended merely as guidance as to kinds of matters to be included and as to the approximate depth of coverage. In the event a particular item relevant to nuclear safety is not covered by the guide, such omission does not relieve the applicant of the obligation to include appropriate information in the Report. Ritualistic adherence to the suggested format and coverage should not be substituted for systematic and logical presentation of information associated with the evaluation of individual safety aspects peculiar to a particular plant. In any case in which any item of information not explicitly mentioned in the suggestions presented in Sections I to XV, inclusive, is necessary to a complete analysis and evaluation of the facility, that information should be included in the Report by insertion of additional sections or subsections.

E. Previously Submitted Information. Attention is invited to 10 CFR § 50.32 wherein it is stated that an "applicant may incorporate by reference information contained in previous applications, statements or reports filed with the Commission: Provided, that such references are clear and specific." Copies of the material referenced should be appended to the Report.

F. Definitions. The following terms used in this guide have the following meanings:

(1) Principal Design Criteria means those fundamental architectural and engineering design objectives established for the project. As such, these criteria represent the broad frame of reference within which the more detailed plant design effort is to proceed and against which the end project will be judged.

(2) Design Bases means that information which identifies the specific functions to be performed by a major component or system in terms of performance objectives together with specific values or range of values chosen for controlling parameters as reference bounds or limits for design. Such limits may be restraints derived from generally accepted "state of the art" practices for achieving functional goals (such as a "no-center melting" restriction placed upon fuel design) or requirements derived from calculating the effects of a situation representing an upper limit which a component or system could reach under credible circumstances (such as peak pressure loading of a containment).

(3) Design Evaluation. A study of the functional and physical features of the major reactor plant systems and components to determine:

(a) Whether the design can or has met performance objectives with an adequate margin of safety.

(b) The identity of and susceptibility of failures, either in equipment or control over process variables, which could be initiating events for accidents.

(4) Safety Analysis. A study of the predicted response of the reactor plant to postulated failures to determine with reasonable assurance whether the plant has the capacity for preventing accidents or mitigating their effects sufficiently to preclude undue risks to public health and safety.

G. Table of Contents. To facilitate use of a Report, a table of contents should be provided. When a Report consists of more than one volume, the complete table of contents should be included in each of the volumes.

H. Drawings. Drawings, maps, diagrams, sketches, and charts should be employed whenever the information can be presented more adequately or conveniently by such means. Due concern should be taken to insure that all information presented in drawings is legible, symbols are defined, and drawings are not reduced to the extent that visual aids are necessary to interpret pertinent items of information presented in the drawings.

I. Abbreviations. Abbreviations should be used discriminately, should be consistent throughout the Report, and should be consistent with generally accepted usage. Any abbreviations not in general usage or unique to the proposed facility should be defined in each volume where they are used.

J. Page Assembly. The assembly of pages of the Report should be accomplished in a manner permitting the rapid removal and reinsertion of a given page, or insertion of a modified page, or of additional pages.

SECTION I.- INTRODUCTION AND SUMMARY

The first section of the Safety Analysis Report should present an introduction and summary. The purpose of this section is to provide a concise description of the facility, its principal design criteria, its design bases, principal operating characteristics, and safety implications. This section should enable the reader to obtain a reasonably accurate understanding of the facility without having to delve into the subsequent sections. Review of the detailed sections which follow can then be accomplished with better perspective and with recognition of the relative safety importance of each individual item to the overall facility design. The following are the types of information that should be included in this section:

A. Principal Design Criteria. Identification of the principal design criteria to which the facility was designed and is to be operated.

B. Characteristics. A tabulation of the most important design and operating characteristics of the facility extracted from succeeding sections of the Report, without details of description, calculation, or discussion. An indication of the principal similarities to other reactors (preferably previously designed or built reactors) and principal differences from such reactors.

C. Design Highlights. Identification of those features of the plant likely to be of special interest because of their relationship to safety. (Primarily in the Preliminary Safety Analysis Report, such items as unusual site characteristics, an uncommon solution to a particularly difficult engineering problem, or a significant extrapolation in the technology as represented by the design should be highlighted. Those features or components on which further technical information is required by 10 CFR § 50.35 should also be identified.)

D. Research and Development Requirements. A resume of the results of any special research and development programs undertaken to establish the final design, and of any programs to be conducted during operation in order to demonstrate the acceptability of contemplated future changes in design or modes of operation.

E. Identification of Contractors. Identification of prime contractors for the design, construction, and operation of the nuclear reactor facility.

F. General Conclusions. A statement summarizing the general conclusions of the applicant with respect to the safety of the facility.

SECTION II - SITE

This section of the Safety Analysis Report should provide information relative to the site. It should consider pertinent natural geological, hydrological, and meteorological characteristics in conjunction with population distribution, land use, and site activities and controls.

Reactor plant designs are to varying degrees influenced by the particular site selected for the location of the reactor. The Commission staff is interested in knowing:

a. How site characteristics have influenced the design and the operating plans;

b. The evaluations that led to selection of site-influenced design criteria and the acceptance of their adequacy from a safety viewpoint;

c. The data (in summarized form) supporting key inputs and assumptions used in the design evaluation.

The Safety Analysis Report is expected to cover the foregoing matters, but not necessarily in the same section. The preferred presentation of material is to include in the site section a clear identification of site-influenced design and operating criteria and the backup data that influenced their selection. The safety evaluation that shows how the siting characteristics have been accommodated or otherwise factored into a safe design and planned mode of operation can be treated under "safety analysis" in a subsequent section with appropriate cross references.

Factors generally considered in site evaluation are identified in 10 CFR Part 100, Reactor Site Criteria. The requirements of 10 CFR Part 20, Standards for Protection Against Radiation, must also be considered, especially with respect to conditions that must be maintained during periods of normal operation of the facility. While many factors may be given consideration in selecting a particular site, it is essential that the factors more important to safety should be suitably emphasized.

The extent of the evaluation and the amount and detail of information provided on any particular factor should be directly proportional to the importance of that factor to the safety of operation: e.g., if liquid wastes are not normally released to the environs, little, if any, information need be included on the sizes and rates of flow of nearby rivers and streams. If most of the site area near the facility has been surfaced, and because of this and other reasons the likelihood of a radioactive release to the ground is remote, little, if any, information on the ion-exchange and filtering characteristics of the soil, and on the drainage characteristics of underground water need be included. Information should be included only as needed to support the safety evaluation.

The converse to the examples given above is also true. If a given factor is of particular importance to safety, the evaluation and information should be appropriately emphasized. If the site is in an active seismic region, sufficient information should be provided to support conclusions about design for seismic loadings and accelerations. If a certain type of destructive windstorm is

characteristic of the site area, appropriate information should be given to support evaluation of occurrence frequencies and storm effects which are reflected in the design bases and operating limits.

The following is illustrative of information that should be included in this section:

A. The Site and Adjacent Areas. For proper orientation and in support of evaluations, a certain amount of descriptive information concerning site geography and population and land usage is desired. This information should include, for example:

(1) The area of the site, and the state and county in which it is located.

(2) The nature, extent, and basis of control exercised by the applicant over the site, including ownership, and, if applicable, leasing arrangements, and arrangements with respect to fencing, posting, patrolling, and similar control mechanisms.

(3) A map of the site and surrounding areas showing the location of the reactor facility, the site boundaries, and prominent natural and man-made features. The area covered by the map should be a function of the reactor power level. It is suggested that the radius. in miles, of the area of coverage be approximately equal to the square root of the reactor thermal power rating in megawatts.

(4) A presentation of the resident population distribution as a function of distance and direction from the nuclear facility. Preferably the distribution should be presented graphically by insertion of appropriate numbers of persons resident in individual area segments of a selected grid system. The grid system should be centered on the reactor facility and should extend radially to a distance, in miles, approximately equal to the square root of the reactor thermal megawatt rating. The interior grid segmentation should be defined, as a minimum, by concentric circles with radii of one, two, three, four, five, and ten miles, and divided by at least sixteen equally-spaced radial lines. The grid segmentation may thereafter be progressively reduced with distance from the reactor, except that the distance between concentric circles should never exceed ten miles and the number of radial lines should never be less than four.

(5) Any significant effect on the above population distribution due to part time occupancy or seasonal variations.

(6) The general character of use of the areas within the appropriate limiting distance specified in subsection A.(3) above with

particular emphasis given to any specific usage that may entail special control requirements in the event of other than normal radioactivity releases from the facility.

(7) Estimates of anticipated future changes, during the intended period of operation of the facility, in both the population density and character of use of the areas within the limiting distances specified in subsection A.(4) above. The bases for the estimates should be explained.

(8) The nature of any activities conducted within the site boundaries, other than those directly related to the operation of the reactor facility, including the number of persons associated with each of the activities and the degree of control exercised over such persons. The grazing of domestic farm animals on site pasturage, the use of certain areas of the site for recreational purposes, and the conduct of commercial enterprises unrelated to reactor operation on the site are examples of the types of activities that should be identified.

B. Meteorology. Evaluation of the meteorology of the site and adjacent areas in order to:

(1) Identify predominating meteorological conditions that could have an important influence upon consequences of gaseous radioactive effluents.

(2) Explain the design bases and operating requirements established because of meteorological considerations.

(3) Provide a basis for evaluating the diffusion parameters selected for analysis of postulated accidental releases.

Within the context of the above, the following are illustrative of the type and extent of meteorological information that has been considered adequate:

(1) Seasonal wind characteristics, including speeds, directions, and frequencies. The frequency of occurrence and duration of essentially calm conditions should also be specified. Information should be given for winds at the surface and aloft to heights at least twice that of the height of the stack but, in any event, consistent with assumptions made for the diffusion of radioactivity releases from the facility. The information should preferably be presented in the form of wind roses oriented and sectionalized so as to permit direct associations with the population distribution grid selected in accordance with subsection A.(4) above.

(2) Precipitation information including prevalent wind directions associated with precipitation.

(3) Prevalent lapse and inversion conditions, including the frequency of occurrence, duration of typical conditions, and information as to the average and maximum vertical temperature gradients.

(4) Any peculiarities of the local meteorological conditions, including effects of terrain, relevant to the diffusion of released radioactive matter.

(5) Information as to the frequency of occurrence and effects of storms accompanied by high and violent winds, including tornadoes, hurricanes, and other storms.

(6) The source of meteorological information and, where used, a discussion of the applicability of data obtained from other than on-site locations. The types of data collected, the methods and frequencies of collection, and the results associated with such off-site and on-site meteorological data collection programs that might have been undertaken especially for the nuclear facility should be discussed.

C. Hydrology, Geology, and Seismology. The Commission is interested in the relationship of these factors to design and operating limitations. The extent of evaluation of the surface terrain and subsurface layers of earth should be consistent with the importance of these matters to the plant design and operation. In general, hydrology, from a safety viewpoint, has not been a predominant factor influencing plant design. Similarly, geological formations beneath the facility in general play the same role in the architectural engineering of the structures for reactors as for any major industrial facility, and hardly justify exhaustive treatment in facility design reports to be submitted to the Commission. Except for the unusual situation in which local hydrology or geology have particular influence on design, a great deal of information with respect to these matters need not be submitted in the Safety Analysis Report. A common practice has been the inclusion, in total, of survey reports of geological experts brought in by an applicant to determine site characteristics as a basis for further design. Such exhaustive reports should be referenced, but a summary with pertinent conclusions will, in general, suffice for the Safety Analysis Report. Emphasis should be on geological information explaining the need or the basis for any unusual design criteria because of geological anomalies.

It is expected that the seismic history of a site will be examined. The extent of evaluations submitted in the Safety Analysis Report and the amount of supporting information should be roughly proportional to the probability of a seismic event and to the intensity of its effects. The Uniform Building Code Seismic Probability Map (1958) provides an appropriate index to probability and intensity. The information submitted should provide explanations for such design requirements that may have been established because of seismic considerations.

The following are illustrative of matters which have been treated in submittals of information supporting and explaining design requirements established because of consideration of site surface and subsurface conditions:

D. Hydrology.

(1) The flood and inundation history of the area, including frequencies and causes.

(2) The absorption and drainage characteristics of water in surface and underground areas, and the depths, estimated direction, and rates of flow of underground eaters.

(3) The characteristics of wells within the area, including depths and yields wherever nearby wells are an important source of water for human consumption or for agricultural purposes.

(4) Sizes and seasonal rates of flow of nearby streams, rivers, lakes, and reservoirs, and the character of their use. For lakes where the consequences of seiche activity are of importance, the characteristics thereof should be fully described.

(5) The source of hydrological information.

E. Geology.

(1) Nature of and results of test borings at the site.

(2) Ion exchange and filtering characteristics of the sale (particularly where liquid radioactive holdup tanks may be buried).

(3) Geological faulting of subsurface layers.

F. Seismology.

(1) General seismic history.

(2) Locations of geological faults with respect to site.

(3) Tsunamis history, if any.

SECTION III - REACTOR

A large measure of safety achieved in nuclear facilities is provided by those characteristics, either inherent or designed into the reactor, that make highly unlikely any significant release of radioactive fission products from the fuel. It is expected that an applicant will provide in this section an evaluation and supporting information to establish the capability of the reactor to perform throughout its lifetime under all normal operational modes, including both transient and steady state, without releasing other than acceptably small amounts of fission products to the coolant. In addition, this section should include a study of the potential of the reactor as a source of abnormal conditions and should provide supporting information for the analyses presented in Section XIV, Safety Analyses.

Broadly speaking, the information submitted should show how the principal design criteria are met by:

(1) Identifying the design bases and explaining the reasons therefor.

(2) Describing the reactor to show how the design bases have been satisfied.

(3) Showing through evaluations that design bases have been met with a reasonable margin for contingencies.

(4) Providing a basis for such limits upon operation that might be appropriate in the interest of safety.

With these objectives in mind, the following is presented for guidance to illustrate the kinds of information desired:

A. Design Bases. The bases upon which the design of the reactor was established, including such matters as:

(1) The functional or performance objectives for the reactor, such as average thermal power, core design lifetime, and fuel replacement program.

(2) Limits imposed upon the design by selection of specific values or ranges of values which are themselves safety bounds or control parameters which are safety bounds; e.g.:

(a) Nuclear Limits such as fuel burnup, reactivity coefficients, stability, excess reactivity, maximum controlled reactivity insertion rates, and control of power distribution.

(b) Reactivity Control Limits such as shutdown margins, stuck rod criteria, maximum rod speeds, chemical and mechanical shim controls, backup scram and emergency shutdown provisions.

(c) Thermal and Hydraulic Limits such as fuel and clad temperatures, critical heat flux ratio, flow velocities and distribution control, coolant and moderator voids and hydraulic stability.

(d) Mechanical Limits such as maximum allowable stresses, deflection, cycling and fatigue limits, fuel restraints (positioning and holddown), capacity for fuel fission gas inventory, maximum internal gas pressure, control rod clearances, material selection, radiation damage, and shock loadings.

The applicant should explain and substantiate his selection from the viewpoint of safety considerations. Where the limits selected are consistent with proven practice, a referenced statement to that effect would suffice; where the limits extend beyond present practice, an evaluation and explanation based upon developmental work and/or analysis are expected. These bases may be expressed as explicit numbers or as general conditions.

B. Reactor Design. Since the nuclear, thermal and hydraulic and mechanical aspects of a reactor are commonly developed by different design groups, a subdivision along such lines has been made below. Further, rather than describe the core in one section and evaluate it in another, the presentation which follows combines in the same subsection the description of hardware and its evaluation from the standpoint of the characteristic under discussion. This was done to make it easier to prepare materials on the core design. Alternate ways are, of course, acceptable.

(1) General Summary. A summary table of the important design and performance characteristics should be included.

(2) Nuclear Design and Evaluation

(a) Nuclear Characteristics of the Design

1. Cold and hot excess reactivity and shutdown margin with and without mechanical and chemical shims and with and without equilibrium xenon and samarium poisoning, for the clean condition and the maximum reactivity condition. If different, excess reactivity associated with temperature, moderator voids, and burnup should be indicated.

2. Coefficients of reactivity for hot, cold, and intermediate temperature conditions:

a. Moderator temperature and void, overall and regional
b. Fuel Doppler
c. Fuel geometry and composition
d. Fuel thermal expansion.

3. Hot and cold reactivity worth of individual control rods and groups of rods for planned patterns and core operating modes with estimates of reductions in effectiveness during core lifetime.

4. Hot and cold reactivity worth of fuel assemblies and mechanical or chemical shims.

5. Hot and cold reactivity worth of any materials within the core or adjacent to it which could have a significant reactivity effect by a change in position as, for example, flooding of superheat reactors or movement of reflecting elements or flux suppression materials.

6. Maximum controlled reactivity insertion rates at startup and at operating conditions.

7. Gross and local radial and axial power distribution for different planned rod patterns with and without equilibrium xenon and samarium.

8. Power decay curve for full and partial scram or power cutback, if applicable, from least effective planned rod arrangement.

9. Minimum critical mass with and without xenon and samarium poisoning.

10. Neutron flux distribution and spectrum at core boundaries and at the pressure vessel wall.

(b) <u>Nuclear Evaluation</u>

1. The nuclear evaluation should include a description of the calculational methods employed in arriving at important nuclear parameters, with an estimate of accuracy by comparison with experiments or with the performance of other reactors. Also included should be discussion of the potential effects for those cases in which nuclear parameters such as excess reactivity, reactivity coefficients and reactivity insertion rates exceed present practice.

2. The stability of the reactor should be evaluated.

3. A safety-oriented discussion of all planned nuclear experiments and tests, both in critical assemblies and zero power and approach-to-power tests at the reactor site, should be included.

(3) Thermal and Hydraulic Design and Evaluation

(a) Thermal and Hydraulic Characteristics of the Design

1. Fuel and cladding temperatures, both local and distributed, with an indication of the correlation used for thermal conductivity and the method of employing hot channel factors.

2. Critical heat flux ratio, both local and distributed, with an indication of the critical heat flux correlation used, method of use, and method of employing hot channel factors.

3. Predicted core average and maximum void fraction and distribution.

4. Coolant flow distribution and orificing.

5. Core pressure drop and hydraulic loads during normal and accident conditions.

6. Correlations and physical data employed in determining important characteristics such as heat transfer coefficients and pressure drop.

7. Summary table of characteristics including important thermal and hydraulic parameters such as coolant velocities, surface heat fluxes, power density, specific power, surface areas, and flow areas.

(b) Thermal and Hydraulic Evaluation

1. Analytical techniques employed with an estimate of uncertainties.

2. Analysis of hydraulic stability.

3. Vibration analysis.

4. Analysis of the potential for and effect of sudden temperature transients on waterlogged elements or elements with high internal gas pressure.

5. Analysis of temperature transients that may cause bowing of fuel control rods or structure.

6. Analysis of the energy release and potential for a chemical reaction should physical burnout of fuel elements occur.

7. Analysis of the energy release and resulting pressure pulse should waterlogged elements rupture and spill fuel into the coolant.

8. Analysis of temperature distributions in solid moderators during normal and accident conditions with an evaluation of the potential for a chemical reaction and an estimate of the total stored energy in the form of heat.

(4) **Mechanical Design and Evaluation**

(a) Internal Layout

1. General assembly drawings showing the arrangement of the important components, positioning and support of the fuel assemblies, control rod and shim arrangement and support, and location of in-core instrumentation.

2. An evaluation which includes considerations such as stresses and deflections of critical pressure parts for pressure, temperature, shock, and hydraulic loads for both normal and accident conditions with particular emphasis on the dimensional integrity of control rod channels and the positioning of fuel assemblies.

(b) Fuel

1. A description and design drawings of the fuel assemblies and fuel elements showing arrangement, dimensions, critical tolerances, sealing and handling features, methods of support, fission gas spaces, and internal components.

2. An evaluation of the fuel design including considerations such as materials adequacy throughout lifetime; vibration analysis, fuel element internal pressure and cladding stresses during normal and accident conditions with particular emphasis upon temperature transients or depressurization accidents; potential for a waterlogging rupture; potential for a chemical reaction; fretting corrosion; cycling and fatigue; dimensional stability of the fuel and critical components during lifetime.

(c) Control System

1. Information on the design of the control system including a description and design drawings of the control rods and followers, rod drives, latching mechanisms, and assembly within the reactor.

2. An evaluation which includes considerations such as materials adequacy throughout lifetime; results of a dimensional and tolerance analysis of the system as a whole, including points of support in the vessel, core structure and channels, control rods and followers, extension shafts and drive shafts; thermal analysis to determine tendencies to warp; analysis of pressure forces which could eject rods from the core; potential for and consequences of a mechanical failure of critical components;

possible effect of violent fuel rod failures on control rod channel clearances; assessment of the sensititivy of the system to mechanical damage as regards its capability to continuously provide reactivity control; previous experience and/or developmental work with similar systems and materials; test and surveillance programs to demonstrate proper functioning at initial startup and throughout lifetime.

C. Safety Limits and Conditions. Either as part of the evaluation discussed in the previous subsection or in a separate treatment under this subsection, the identification of those aspects of design that significantly influence the maintenance of fuel integrity should be identified. Evaluations are expected to show the relationship between normal operating ranges planned for key variables affecting fuel integrity and limits beyond which operation will not proceed in the interests of safety.

Analysis and engineering judgment may lead to the conclusion that the fuel clad surface temperature is a primary safety parameter in this class. Consideration of failure mechanisms might have shown that a surface temperature limitation, if always met, would assure the integrity of the fuel clad. On the other hand, this parameter is not generally directly measurable. Therfore, parameters such as coolant effluent temperatures, coolant flow and coolant pressures might prove appropriate indirect indications. An evaluation of the primary and related parameters which shows a basis for selection of particular parameters upon which safety limits are placed is expected.

In similar fashion, evaluations are expected to reveal those conditions, if any, that might be considered as prerequisities to operation under specified modes. For example, it may not be advisable in the interests of safety to start up without having available certain protective circuitry or systems. On the other hand, redundancy may provide a flexibility in operations without significant effect on readiness for emergencies. Identifying such situations and showing through analysis that safety can be maintained may permit flexibility in licensed operation that might not otherwise be possible.

D. Tests and Inspections. The principal tests planned, if any, for the verification of characteristics of the core or its reactivity control features having safety implications should be identified. Emphasis should be given to those items such as developmental programs, preoperational and postoperational testing or surveillance programs which will be employed to substantiate the adequacy of the design to assure that safety limits will not be exceeded or to provide operational data in support of planned escalation in power or different operating modes.

SECTION IV - REACTOR COOLANT SYSTEM

This section of the Safety Analysis Report should provide information covering the reactor coolant system. By reactor coolant system is meant both the reactor coolant or coolants and all components that provide a boundary for coolant flow. In the case of direct-cycle plants, the extent of the reactor coolant system to be considered in this section should be limited to the boundary defined by the main isolation valves. The portions of the system beyond the isolation valves should be treated as part of the Steam and Power Conversion System in Section X.

Coolant flow is a fundamental requisite to reactor operation. Its loss while the reactor is at power or subsequent to power operation could lead to fuel melting and fission product release to the coolant. The coolant enclosure represents a principal safeguard whose integrity is important in the protection of public health and safety.

Evaluations, together with the necessary supporting material, should be submitted to show that the reactor coolant system is adequate to accomplish its intended objective and to maintain its integrity under conditions imposed by all foreseeable reactor behavior, either normal or abnormal. The information should permit a determination of the adequacy of the evaluations; that is, assurance that the evaluations included are correct and complete and all the evaluations needed have been made. Evaluations included in other sections that have a bearing on the reactor coolant system should be referenced.

The following are illustrative of the types of evaluations and supporting information that should be included in this section:

A. Design Bases. The bases upon which the design of the reactor coolant system was established, including, for example:

(1) The performance objectives of the system and its components from which the design parameters are derived for both the normal and transient conditions considered.

(2) The design pressure, temperature, seismic loads, and maximum system and component test pressures for the system and individual components.

(3) Design cyclic loads which the system is expected to sustain and their estimated frequency during the service lifetime, including consideration of startup and shutdown operations, power level changes, emergency and recovery conditions, switching operations (one or more coolant loops), and hydrostatic pressure tests.

(4) The design service life of the system, including (a) estimates of operating time at specific levels of power production and (b) estimated life of components of reactor coolant system which, by virtue of design, will be accessible for inspection, repair, or replacement.

(5) The codes and classifications (for pressure-retaining components only) applied in the design, construction, inspection, and testing of the primary coolant system, the primary side of the auxiliary or emergency reactor systems, the primary side of engineered safeguard systems connected to the primary system, and the interconnecting piping systems.

The selection of the specific values or ranges of values for the variables such as pressure, temperatures, loadings, etc. should be explained from the standpoint of safety.

B. <u>System Design and Operation</u>. The system description and major operational features of the reactor coolant system, including such items as:

(1) A schematic flow diagram of the reactor coolant system denoting all major components, principal pressures, temperatures, flow rates, and coolant volume under normal steady state full power operating conditions.

(2) A piping and instrumentation diagram of the reactor coolant system and the primary sides of the auxiliary or emergency fluid systems* and engineered safeguard systems* interconnected with the reactor coolant system, delineating on the diagram:

(a) The extent of the systems located within the containment,

(b) The points of separation between the reactor coolant (heat transport) system and the secondary (heat utilization) system, and

(c) The extent of isolability of any fluid system as provided by the use of isolation valves between the radioactive and non-radioactive sections of the system.

(3) An elevation with principal dimensions of the reactor coolant system in relation to the supporting or surrounding concrete structures from which a measure of the protection afforded by the arrangement and the safety considerations incorporated in the layout is gained.

(4) A cross sectional assembly of the major components of the reactor coolant system, such as the reactor vessel with internals, pressurizer, steam generators, heat exchangers, pumps, and the principal valves.

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*The appropriate information pertaining to these individual systems should be included under Sections IX and VI, respectively.

(5) An identification and location of all pressure-relieving devices (all such devices may be indicated on the piping and instrumentation diagram referenced in subsection B(2)) as applied to:

(a) The reactor coolant system,

(b) The primary side of the auxiliary or emergency systems interconnected with the primary system, and

(c) Any blowdown or heat dissipation system connected to the discharge side of the pressure-relieving devices.

(6) Protection provided for the principal components of the reactor coolant system against environmental factors (e.g., fires, flooding, missiles, seismic effects) to which the system may be subjected.

(7) The materials of construction exposed to the reactor coolant and their compatibility with the coolant and contaminants or radiolytic products to which the system may be exposed.

(8) The materials of construction of reactor coolant systems and their compatibility with external insulation or the environmental atmosphere in the event of coolant leakage.

(9) The maximum allowable normal and emergency heating and cooling rates imposed on the reactor coolant system and, in particular, the reactor vessel, to limit thermal loadings within design specifications.

(10) The provisions made for leak detection of radioactive products in areas embracing the reactor coolant system which are inaccessible and where the possibility of coolant leakage exists.

(11) The additives to the primary coolant system such as inhibitors whose principal function is directed toward corrosion control within the system.

C. System Design Evaluation. *Evaluations of the reactor coolant system, as a whole, in support of the system's capability to provide or maintain, under normal or transient operating conditions, the integrity required as a primary barrier and operational dependability throughout design lifetime, including evaluations of:

(1) The extent of designed-in safety factors, or the margin of capacity, provided beyond that required to meet performance specifications (e.g., coolant pump coastdown characteristics in event of pump failure, or the minimum number of reactor coolant or recirculation loops in operation to satisfy operational requirements).

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*Safety analyses and consequences related to unforeseeable events or hypothetical accidents should be included under Section XIV - Safety Analyses.

(2) The extent to which the function of the reactor coolant system (or an auxiliary system) may depend upon the operability of an interconnected system (e.g., the heat exchange system associated with a reactor shutdown cooling system).

(3) The extent to which any auxiliary or emergency system or group of systems is relied upon to assure the integrity of the reactor coolant system (e.g., the reliance placed upon the overpressure protection system safety valves in relation to that placed upon other pressure-limiting systems, such as emergency cooling systems and safety system scram circuits).

(4) The overpressure protection provided for each vessel, group of vessels, and the entire system, including:

   (a) The principal assumptions and governing conditions which dictated the maximum pressure-relieving requirements of the system;

   (b) The emergency dissipation or flow capacities of each pressure relief valve as related to the thermal and overall system characteristics under the conditions which governed the sizing of the pressure relief valves, and

   (c) The pressure setting or bursting pressures of pressure-relieving devices.

(5) The extent of independent multiplicity or duality of function of systems or their components which may contribute to operational dependability when called upon to function (e.g., the duplication or sharing of pumps among auxiliary systems connected to the reactor coolant system).

(6) The potential of the system as cause of accidents, either through failure of components or control over the coolant; e.g., loss of coolant flow, pressure relief valve sticking open, etc.

(7) Those operational limits and safety conditions selected for inclusion in "technical specifications" such as:

   (a) System heatup and cooldown rate
   (b) Stress limits dictated by NDT considerations
   (c) Coolant chemistry control
   (d) Coolant radioactivity limits
   (e) Maximum pressures and temperatures.

D. <u>Tests and Inspections</u>. Identification of principal tests and inspections planned for the surveillance of the reactor coolant system to assure continued performance and integrity of the system boundary as well as continued functional performance required for safe operations; e.g.,

(1) A summary of test and inspection programs to be followed with respect to the reactor coolant system, including in-service tests and inspections requiring dismantling and reassembly of components.

(2) Those tests and inspections of such importance to safety as to be included in the technical specifications of the operating license. An example might be the requirement for a material irradiation surveillance program to monitor the physical properties of the reactor vessel to ensure compliance with NDT limitations that have been specified. In any event, an evaluation should be presented of each test and inspection selected as a surveillance requirement for inclusion in the technical specifications. This evaluation should include the identification of the test or inspection, the frequency, acceptance criteria, and action to be taken in the event such criteria are not met. An explanation as to the reasons for each selection should be included.

SECTION V - CONTAINMENT SYSTEM

This section of the Safety Analysis Report should provide information concerning the facility containment system. For the purpose of this Report, the containment system may be considered as composed of the containment structure and the directly associated systems upon which the containment function depends (e.g., the system of isolation valves installed to maintain or re-establish containment system integrity when required). Engineered safeguards which may be called upon to operate in conjunction with the containment function in the event of an accident should be reserved for discussion in Section VI.

In the design of nuclear power plants, the containment system which encompasses the reactor and other portions of the plant (which vary depending on reactor type and plant) constitutes a design feature provided primarily for the protection of public health and safety. Being a standby safety system, it may never be called upon to function, but as a safeguard must be maintained in a state of readiness. The ability to perform its intended role, if called upon, of acting to confine the potentially hazardous consequences of a gross accident, depends upon maintaining tightness within specified bounds throughout operating lifetime.

The Report is expected to provide the Commission with information that shows the containment system has been evaluated for assurance that:

a. The containment will fulfill its intended objectives, and
b. Such objectives are consistent with protection of the public safety.

Information provided should permit a determination of the adequacy of the evaluations; that is, assurance that the evaluations included are correct and complete and all the evaluations needed have been performed. Evaluations in other sections having a bearing on the adequacy of the containment system should be referenced.

More specifically, in recognition of the safety role assigned to the containment system, it is expected that the evaluations should be directed toward:

(1) The bases upon which the containment system requirements were established and, in particular, the identification and explanation for the choice of values of the principal design parameters; i.e., the design pressure and the allowable leakage rate.

(2) The major components and associated systems provided to fulfill the required containment function and the extent of the assurance that the proposed designs will perform their intended function reliably.

(3) The extent to which the containment system's effectiveness and functional dependability will be maintained and verified by testing throughout the plant's operating lifetime.

(4) The capability of the containment system to continue to function in accordance with design specifications when subjected to environmental forces such as, winds, floods, and seismic activity associated with the site location.

(5) The designed-in margin available in the containment performance capability beyond that required to handle the accident postulated for defining upper limits on required performance.

(6) The extent to which the operation of any engineered safeguards (see Section VI) is relied upon to attenuate the postaccident conditions imposed upon the containment system.

Particular emphasis should be placed upon the evaluation of design features, operational reliability, and testability on the assumption that the containment system will not normally operate. It is through a critical evaluation of its design features and testing schedule that assurance is obtained that the system will function properly if called upon.

The following are illustrative of evaluations and supporting information that should be included in this section:

A. Containment System Structure

(1) Design Bases. The bases upon which the design of the containment system structure was established, including, for example:

(a) The postulated accident conditions and the extent of simultaneous occurrences which determined the containment design requirements.

(b) The sources and amounts of energy and material which might be released into the containment structure, and the postaccident time-dependency associated with these releases.

(c) The contribution of any engineered safeguard system in limiting the maximum value of the energy released in the containment structure in the event of an accident.

(2) Containment System Structure Design. The design features of the containment system structure and the explanation* for their selections, including, for example:

(a) Design internal pressure, temperature, and volume

(b) The design leakage rate.

(c) Design external loadings imposed by barometric pressure changes, wind, snow or ice, floods or inundations, and earthquakes.

(d) The code and vessel classification applicable to the design, fabrication, inspection, and testing of the structure.

(e) Plans and elevations showing principal dimensions.

*Where explanation is given in other section, only cross referencing is necessary.

(f) The estimated number and types (preferably supported by typical details) of penetrations, equipment access doors, emergency escape openings, and air locks.

(g) Missile protection features.

(h) Protection provided against combustible, explosive, or reactive materials being released inside the containment structure.

(i) The corrosion protection or material allowances provided.

(j) The extent of thermal or weather insulation provided.

(k) The extent to which shielding requirements have been incorporated.

(l) The provisions or system provided for vacuum relief.

B. Containment Isolation System. The system of isolation valves* applied to fluid lines penetrating the containment barrier to maintain or re-establish containment system integrity during normal operating periods, or emergency and postaccident periods, should be considered as part of the containment system.

(1) Design Bases. The bases established for the design of the isolation valving required for fluid lines, including, for example:

(a) The governing conditions under which containment isolation becomes mandatory.

(b) The criteria applied with respect to the number and location (inside or outside of containment) of independent isolation valves provided for each fluid system penetrating the containment and the basis thereof.

(2) System Design. The design features of the isolation valve system, including, for example:

(a) A piping and instrumentation diagram of the isolation valve system indicating the location with respect to the containment barrier of all isolation valves in fluid systems penetrating the containment wall, or systems communicating directly with the outside atmosphere, (e.g., vacuum relief valves).

(b) A summary of the types of isolation valves applied and their open or closed status under normal operating conditions, shutdown, or accident situations.

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* Isolation valves applied to systems not related to the containment function should be excluded from this section but should be included under the appropriate section of the Report relating to the respective systems.

(c) The primary and secondary modes of actuation provided for the isolation valves, (e.g., valve operators, manual remote or automatic).

(d) The number of parameters sensed and their values which are required to effect closure of isolation valves.

(e) The closure time and sequence of timing for the principal isolation valves to secure containment isolation.

(f) Protection of isolation valves, actuators, and controls against danger from missiles.

(g) Provisions to ensure operability of isolation valve system under accident environments (i.e., imposed pressures and temperatures of the steam-laden atmosphere in the event of an accident).

C. Containment Ventilation System. The system for ventilation of the containment system and for other air purification or cleanup facilities servicing the containment system under normal and emergency conditions should be considered as part of the containment system (if components of this system are considered as pertaining to an engineered safeguard, appropriate information should be included in Section VI).

(1) Design Bases. The design bases for the ventilation and air purification system under either normal operating conditions or accident situations, including:

(a) The conditions which establish the need for ventilation of the containment structure.

(b) The bases employed for sizing the ventilation and air purification facilities.

(2) System Design. The design features of the system and the bases thereof, including, for example:

(a) A piping and instrumentation diagram of the ventilation and/or other air purification facilities, delineating the extent of system located within the containment structure.

(b) Performance objectives (e.g., ventilation flow rates, temperature, humidity, or limits of radioactivity levels to be maintained in the containment structure).

(c) Provisions to exhaust, monitor, and filter the ventilation air and the provisions for safe disposal of the effluent to the outside atmosphere (e.g., systems discharging the effluent through stacks).

(d) A summary of the type of isolation valves applied to the ventilation lines of the system penetrating the containment wall, including similar information specified in subsection B.(1)(b) - B.(2)(g), inclusive.

D. Containment Leakage Monitoring Systems. The systems intended to monitor the development of gross leakages or measurement of leakages within allowable limits in the containment system (leakage pumpback systems which monitor containment barrier leakages may be included under this category) should be considered as part of the containment system.

(1) Design Bases. The design bases of the containment monitoring system and any operational restrictions imposed on the containment system from measurements of leakages as derived by operation of the monitoring system (e.g., leakage limits specified which may require plant shutdown and repairs).

(2) System Design. The design features of the system and the bases thereof, including, for example:

(a) A piping and instrumentation diagram of the system delineating the extent of system located inside or outside of the containment structure.

(b) Performance objectives (e.g., operational parameters, leakage rates to be measured).

(c) Provisions for isolating the systems in the event of accident to preclude leakage through any components of the monitoring system.

(d) Features considered for system components located outside of the containment structure which in effect extend the containment boundary by recirculation of containment atmosphere (e.g., pumpback compressor systems).

E. System Design Evaluation. *Evaluations of operational systems associated with the containment which serve to indicate or maintain the state of readiness of the containment within a specified leakage rate limit during operating periods when containment integrity is required; including:

(1) The extent to which the measure of containment leak tightness at any time depends upon the operation of a system, such as a continuous leakage monitoring system, a continuous leakage surveillance system for containment penetrations and seals or a pumpback compressor system which maintains a negative pressure between dual barriers of a containment.

*Safety analyses and discussion of the consequences of accidents under which the containment function becomes essential should be included under Section XIV, "Safety Analyses".

(2) An analysis of the capability of these operational systems to perform their functions reliably and accurately during operating periods, and under conditions of operating interruptions (e.g., the performance margin, if any, in a pumpback compressor system that might allow it to sustain an operational failure and still function adequately).

F. Tests and Inspections. This section should provide information about the program of testing and inspection applicable to:

(1) Preoperational testing of the containment system.

(2) Postoperational surveillance to assure a continued state of readiness to perform.

Emphasis should be given to those tests and inspections considered essential to a determination that performance objectives have been achieved and a performance capability maintained throughout lifetime above some pre-established limit. Such tests might include, for example, integrated leak rate tests of the containment structure, local leak detection tests of penetrations and valves and operability tests of fail-safe features of isolation valves.

The information should include such things as:

(1) What tests have been planned and why.

(2) Considerations that led to test periodicity.

(3) Test methods to be used.

(4) Requirements set for acceptability of observed performance and the reasons therefor.

(5) Action to be taken in the event acceptability requirements are not met.

Evaluations made elsewhere in the Report which explain the bases for tests planned need not be repeated but only cross referenced.

Particular emphasis should be given to those surveillance type tests that are of such importance to safety that they may become a part of the technical specifications of an operating license. The bases for such surveillance requirements should be developed as a part of the Safety Report.

SECTION VI - ENGINEERED SAFEGUARDS

The control over fission products, whether they result either from normal operation or from accidents, represents the central objective in safe reactor design and operation. At the present time, assurances of public safety through design stem from three approaches toward this objective; namely,

a. Design to preclude accidents resulting in release of fission products from the fuel;

b. Location to restrict the dispersal of fission products if by some unforeseen circumstance release beyond the coolant boundary takes place, and

c. Location to minimize exposure risk to large numbers of people.

Those features expressly provided as precautionary safeguards or as backup protection to the various aspects of design to prevent the occurrence of accidents have been grouped under the term "engineered safeguards". These are systems that are intended specifically to function following a major accident and whose objective is to limit the consequences of the accident. In functional terms, they can be broadly categorized as (1) containment, (2) driving force reduction, and (3) fission product inventory reduction.

This section should provide information on engineered safeguards other than the containment system covered in Section V. Information submitted should be directed toward showing that:

(1) The concept upon which the operation of the system is predicated has been or will be proven sufficiently. Experience, full-scale tests under accident conditions, or conservative extrapolations from present knowledge are considerations (e.g., containment suppression concepts, containment venting concepts, filtration mechanisms, heat transfer mechanisms, etc.).

(2) The system will function during the period required and will actually accomplish its intended purpose (e.g., filtration efficiency, pressure reduction, etc.).

(3) The system will function when required and will continue to function for the period required (e.g., consideration of component reliability, system interdependency, redundancy and dispersion of components or portions of system, etc.).

(4) Provisions have been made and tests will be performed to ensure that the system will be dependable and effective.

The engineered safeguards included in reactor plant designs vary from facility to facility. The engineered safeguards explicitly discussed here are those that are commonly used to limit the consequences of an accidental rupture

in the main coolant system of a light water-cooled reactor. As such, these should be treated as illustrative of engineered safeguards that should be treated in this section of the Report, and of the kind of informative material that is needed. Where different types of engineered safeguards may be used in water-cooled or other kinds of reactors, they should be covered in a similar manner.

Most light water-cooled licensed power reactors have containment systems. Guidance with respect to containment systems has been treated separately as Section V. In addition to containment, engineered safeguards that have been included in light water-cooled power reactors are:

(1) Core Spray and Safety Injection Systems.
(2) Containment Spray Systems.
(3) Atmosphere Recirculation Cooling and Filtration Systems.

Only the Core Spray and Safety Injection Systems will be discussed in the paragraphs that follow. The guidance provided on those systems is meant to be illustrative of the kind of information that should be provided on all engineered safeguards.

The depth of information associated with the analyses, evaluations, and the supporting data submitted should take into account the reliance placed on the system for safe operation and overall plant acceptability.

CORE SPRAY AND SAFETY INJECTION SYSTEMS

Loss of the coolant of a reactor after a period of sustained operation could result in core meltdown from the fission product decay heat. In the case of water-moderated reactors a core spray or safety injection system is often included in a facility to prevent, or limit, core meltdown in the event of an accidental main coolant system rupture by supplying coolant to the core to remove the fission product heat produced. A system using spray rings to distribute the coolant over the core is called a core spray system; others are safety injection systems.

The systems generally consist of storage facilities, piping, pumps, valves, heat exchangers, spray nozzles, and instrumentation. The included functions will vary from system to system. Some are once through systems--when the coolant in the storage facility expires, the systems ceases to operate. Some systems combine the storage facility with a recirculation system in which the water accumulating in the containment sump is passed through heat exchangers, cooled, and returned to the core. Some systems include a poison in the coolant for cold shutdown.

The specific functional requirements of a core injection system will depend upon the reactor design. Such matters as the time available following coolant loss, cooling capacity required, and the length of time cooling must be sustained vary. The functional requirements for the system and an explanation of why these were established should be a fundmental part of the Report.

When discussing the factors of dependability and effectiveness, specific attention should be directed to such things as system starting, adequate coolant delivery, availability of coolant, period of time the system must operate, the environment in which the system will operate, the effect of external forces, the state of the art and proposed research and development to ensure proper flow distribution to adequately cool the core, the testing program to ensure dependable operating, and the reliance placed on the system for overall plant safety.

The ability of the system to start and deliver the required cooling capacity is fundamental. Considerations should include the design, operation, and testing that are associated with system dependability from the sensing of an accident, through the availability of emergency power, to the assurance of adequate coolant flow at the core.

Not only the sufficiency of the coolant for the period required but also availability in the proper form (e.g., not frozen and with proper poison concentration) is important. The environment in which the system will be required to function should be considered, in particular, if components are located within the containment system when the conditions might change following the accident.

Possibilities for the system's experiencing external forces such as missiles and forces causing movement or vibration should be evaluated; e.g., since the core spray or safety injection system is connected to the main coolant system, it is conceivable that an accidental rupture of the main coolant system could cause movement that would break the piping and negate or reduce the effectiveness of the core spray or safety injection system.

Evaluations to show that there will be adequate and proper flow distribution through the core are important. There has been little experience in this area. Such complications as the large number of channels, the effect of channel length, the phase change of the cooling water, potential metal-water reactions, and the lag time associated with system operation should be considered.

Since the system does not operate except following an accident, a measure of its dependability may be ensured through testing. Information concerning the initial tests and subsequent periodic tests and inspections should be included.

To provide guidance the following illustrate information that should be included in this section:

A. <u>Design Bases</u>. The design of core spray or safety injection systems is based upon the assumption of the accidental rupture of the main coolant system and how this might affect (1) the core, and (2) the environment in which the system will operate. Some systems are designed to prevent core meltdown in the event of a main coolant pipe rupture with discharge from both ends; others have been designed to prevent meltdown for a smaller size rupture and limit meltdown for the larger rupture. The ability of a system to satisfactorily accommodate a break of a certain size does not necessarily mean it can accommodate all breaks. Whatever the bases for setting the functional requirements for the system, these should be identified and explained. To illustrate, the

design bases would be expanded to include:

(1) The range of main coolant system ruptures and coolant leaks that the core spray or safety infection system was design to accommodate and the analyses* supporting the selection.

(2) The fission product decay heat that the core spray or safety injection system was designed to remove and the analyses* supporting this selection.

(3) The reactivity required for cold shutdown for which the core spray or safety injection system was designed and the analyses* supporting this selection.

(4) Environmental conditions for which the system was designed.

B. System Design. This section should provide the understanding of how the system has been constituted in keeping with the functional requirements established from safety analyses. Information on a safety injection system would be expected to include such things as:

(1) A schematic piping and instrumentation diagram of the system showing the location of all components, piping, storage facilities, points where connecting systems tie in, and instrumentation and control associated with pump actuation.

(2) Codes and classifications applicable in system design.

(3) Materials used to establish compatibility.

(4) Design pressure and temperature of components for various portions of the system and an explanation of their selection.

(5) Capacity of each of the coolant storage facilities.

(6) Pump characteristic curves and pump power requirements.

(7) Heat exchanger characteristics including design flow rates, inlet and outlet temperatures for the cooling fluid and the fluid being cooled, the overall heat transfer coefficient and the heat transfer area.

(8) Relief valve settings or venting provisions included in the system.

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*Where these analyses have been made in other sections, such as Safety Analyses, Section XIV, only cross referencing is necessary.

(9) Reliability considerations incorporated in the design to ensure the system will start when needed and will deliver the required quantity of coolant (e.g., redundancy and dispersion of components, transmission lines, and power sources). A distinction should be made between true redundancy incorporated in a system and multiple components (e.g., a system that is designed to perform its function with only one of two pumps operating has increased reliability by redundancy; whereas, a system that has two pumps and requires that both operate to perform its function does not have redundancy).

(10) Provisions to protect the system (including connections to the primary system or other connecting systems) against damage that might result from movement (between components within the system or between the system and connecting systems) or missiles.

(11) Provisions for various methods of actuation (e.g., automatic, manual, different locations). The conditions requiring system actuation together with the bases for the selection (e.g., during periods when the system is to be available, whenever the main coolant system pressure is less than X psig, the core spray system will be actuated automatically).

(12) Special provisions included to ensure proper operation in all environments (normal and accident) of the mechanical instrumentation, and electrical portions of the system.

(3) Provisions for facilitating performance testing of components (e.g., bypasses around pumps, sampling lines, etc.).

C. Design Evaluation. The functional requirements established for engineered safeguard systems generally come from safety analyses which consider predicted effects of some specific assumed accident as a frame of reference for design. It is expected that such analyses would be included in Section XIV, Safety Analyses. However, having set certain functional performance as the objectives of a safeguard design, this section of the Report is expected to include those system evaluations whereby it has been concluded that functional requirements have been met with an adequate margin for contingencies. Such evaluations are expected also to provide the bases for any operational restrictions such as minimum functional capacity or testing requirements that might be appropriate as a condition of a license in the interests of assuring public safety. For example, in the case of the safety injection system, the following evaluations should be included:

(1) Analyses and tests performed to ensure that there will be adequate flow and proper flow distribution to prevent, or limit, core meltdown as intended.

(2) Analyses and tests performed to determine the nuclear and chemical effects of system operation on the core.

(3) The extent to which components or portions of the core spray or safety injection system are required for operation of other systems and the extent to which components or portions of other systems are required for operation of the core spray or safety injection system. An analysis of how these dependent systems would function should include system priority (which system takes preference); conditions when various components or portions of one system function as part of another system (e.g., when the water level in the reactor is below X feet, the feed pumps will supply water to the safety injection system and not the containment spray system); and any limitations included to ensure minimum capability (e.g., storage facility common to both core spray and containment spray systems shall have provisions whereby the quantity available for core spray will not be less than Y gallons).

(4) The range of acceptable lag times associated with system operation; that is, the period between the time an accident has occurred requiring the operation of the system and the time spray is discharged into the core. Analysis supporting the selection should include valve opening time, pump starting time, etc.

(5) Thermal shock considerations, both in terms of effect on operability of the core spray or safety injection system and the effect on connecting systems.

(6) Bounds within which key system parameters must be maintained in the interests of constant standby readiness; e.g., such things as:

(a) Minimum poison concentrations in coolant.
(b) Minimum coolant reserves in storage volumes.
(c) Minimum inoperable components.

(7) Interdependency of the system with other engineered safeguard systems.

D. Tests and Inspections. The core spray and safety injection systems are standby systems, not normally operating. They operate only following an accident; consequently, a measure of system readiness must be achieved via tests and inspections. The periodic tests and inspections planned should be identified and reasons explained as to why the program of testing planned is believed to be appropriate.

The information should include such things as:

(1) What tests have been planned and why.

(2) Considerations that led to test periodicity.

(3) Test methods to be used.

(4) Requirements set for acceptability of observed performance and the reasons therefor.

Evaluations made elsewhere in the Report which explain the bases for tests planned need not be repeated but only cross referenced.

Particular emphasis should be given to those surveillance type tests that are of such importance to safety that they may become a part of the technical specifications of an operating license. The bases for such surveillance requirements should be developed as a part of this Safety Analysis Report.

SECTION VII - INSTRUMENTATION AND CONTROL

Nuclear plant instrumentation senses the results of the nuclear power generating process and makes it possible to actuate appropriate mechanisms to keep the process within safe and economical bounds. The information sensed by the instrumentation is used to perform regulating and protective functions. Regulating systems take the plant from shutdown to power and then monitor and maintain key plant variables, such as reactor power, flow, temperature and radioactivity levels within predetermined limits at steady state and during normal plant transients. Protective systems are utilized to shut down the reactor to protect the core and the integrity of the coolant boundary against effects of abnormalities such as equipment malfunctions, component failures, and operator errors. Protective systems are also provided to signal for containment isolation and to bring other engineered safeguards into action in the event of a major reactor accident.

Information about instrumentation and controls provided in this section should place emphasis upon the protective functions served. To stress this outlook, it is desired that instrumentation and control grouped functionally as Protective Systems and Regulating Systems be discussed first. This then should be followed by such detail on individual instrument channels as may be necessary to explain what parameters are sensed and how the intelligence obtained is translated by instrumentation into protective actions.

The following is intended to provide guidance toward such an approach:

A. Protective Systems. The protective systems, which consist of the reactor protective circuitry and the instrumentation and control for engineered safeguards, normally perform the most important of the instrumentation and control safety functions. They provide protection against unanticipated release of fission products from the fuel, from the coolant boundary and from the containment. For the purpose of the Safety Analysis Report, any equipment supplying signals to any of these protective systems should be considered a part of that protective system. This includes equipment which might be physically a part of the nuclear instrumentation, nonnuclear process instrumentation, radiation instrumentation, or other plant instrumentation. Any actuator or other device controlled by any of the protective systems such as safety rod scram magnets, scram solenoid valves, or ventilation valve control solenoids should also be considered a part of the protective system. Information presented should be organized along lines such as the following:

(1) Design Bases. The bases upon which the design of the protective systems was established, including such matters as:

(a) Vital protective functions required of the instrumentation; e.g., those requirements derived from analysis of plant disturbances or incidents discussed in Section XIV, Safety Analyses, such as conditions which must initiate scram and containment isolation.

(b) Principles of design that dictated the general system concepts such as:*

1. Number of independent instrument channels to provide vital protective actions.

2. Relationship of protective and regulating instrumentation.

3. Testability as a function of operational mode.

(c) Functional requirements to be satisfied by protective actions such as rod cutback, rod stops, low count rate interlocks.

(d) Operating environment in which equipment must function; e.g., the most severe environment for the control room or the containment under which reactor operation might be allowed to continue.

The applicant should explain and substantiate his selection from the viewpoint of safety.

(2) Systems Designs

(a) A description of the systems, supported by block diagrams showing each channel of protection from detector (or other input device) to final actuating device, should be provided. Sufficient information should be included on the sampling systems, sensors, amplifiers, trip elements, actuators, and other control component to explain the principal features of their design and to show how the design requirements will be met; e.g.: features aimed at reliability, testability and serviceability such as redundancy of channels, coincident arrangements of circuitry, on-line testing capability, mechanical and electrical interlocking, rod withdrawal stops or cutbacks (rather than scram). Features to provide reactor shutdown to protect plant equipment other than the core should be clearly identified to prevent their being evaluated on the same basis as core protection.

(b) A summary chart which tabulates the process variables sensed for protective reasons, the normal steady state values of these variables and the signal levels at which protective system actions are initiated. Discussion of the data tabulated should include an explanation of the relationship of these values to safety limits that have been derived from safety analyses such as those made under Section XIV.

*Note: Examples are illustrative and not all inclusive. "The Technology of Nuclear Reactors", Vol. 1, edited by T. J. Thompson and J. Beckerley, MIT Press, pp.293-294, further illustrates such requirements.

(3) Systems Evaluation - Systems evaluation should include:

(a) Analysis of the capability of the protective systems to meet their functional requirements. (The safety analyses prepared under Section XIV may be referenced to the extent that the evaluations therein show functional capability of the system.)

(b) Analysis of developmental and other preoperational tests for suitability as evidence that the systems will meet their functional requirements under all environmental conditions under which they might be required to operate.

(c) An evaluation of the system's susceptibility to component failures or faults and the effect of such failures on ability to perform the vital protective functions. The evaluation should be supported by schematic diagrams of those parts of the circuit where redundant channels are brought together (e.g., in logic circuits and final actuating devices).

(d) An evaluation of the need for and the adequacy of the planned testing and surveillance to assure that the protective systems, particularly those which are normally in a passive state, are in a constant state of readiness. The nature of the tests should be explained, including such matters as the frequency for performing each test and performance required as indication of acceptability.

B. Regulating Systems. These systems provide the ability to regulate reactor power via such process variables as pressure, temperature, flow, and radiation level, either automatically or manually. From a safety viewpoint, such control systems are of significance because:

a. Characteristics of the regulating circuitry influence performance requirements imposed upon the protective instrumentation.

b. Component failures or malfunctioning of the regulating systems could create a demand for protective action.

The principal objective of presenting information with respect to regulating systems should, therefore, be to show how such factors have been dealt with in design and planned modes of operation. Information provided should include:

(1) Design Bases

(a) The bases upon which the design of the regulating systems was established, including such matters as regulating functions required of the instrumentation (e.g.: reactivity effects from such phenomena as xenon buildup, fuel depletion, and core poisoning) or for which regulating systems are designed to compensate.

(b) Design requirements dictated by safety considerations such as: the potential for a system malfunction resulting in a reactor accident and the requirement to maintain a reactor plant variable under steady state within limits from which the protective systems could function effectively.

(c) Requirements established from startup considerations; e.g., the ability to override xenon, maintain continued surveillance over count rate, or limit reactivity insertion rates.

(d) Requirements imposed on process instrumentation by core design criteria; e.g., flow control as a function of power.

(2) System Designs and Evaluation

(a) Design

A description of the system provided for regulating reactivity, with an explanation of principal characteristics as they relate to the functional requirements they satisfy, including items such as automatic and manual control schemes, rod programming and sequencing, system inputs (reference can be made to the appropriate process variable instrument system for discussion of the systems) and interlocking.

(b) Evaluation

1. Evaluation of potential system failures such as electrical faults, phase reversal, service failures, and switch and relay failures which could result in a reactivity excursion. Such evaluations should be supported by schematic diagrams showing circuitry from power source to rod drives including manual switching and automatic controls.

2. An evaluation of the effectiveness of interlocking to prevent undesired reactivity excursions. Such evaluations should include schematic diagrams of the interlocking.

3. An evaluation of the capability of the controls to maintain operation in a region where the reactor protective system is effective.

4. Analysis of system stability when disturbed by occurrences such as a sudden loss of load.

5. Analyses that show the relationship between steady state values of variables used for regulating functions and safety limits established from safety considerations such as those discussed under Safety Analyses, Section XIV.

C. Instrumentation Systems Design and Evaluation. The inputs to protective and regulating systems are provided by a diversity of instruments. These sense a number of plant process variables such as neutron flux, temperature pressure and coolant flow. The previous paragraphs have attempted to focus emphasis upon the "system engineering" which explains why and how outputs from the individual instrument channels are brought together to accomplish the broad functions of plant protection and power regulation. In this paragraph, information should be provided about the individual instrument channels. For purposes of the Report, such instrumentation can be subdivided into nuclear, nonnuclear process, and radiation instrumentation. Information along the following lines should be provided:

(1) Nuclear Instrumentation

(a) Design

1. A general description of the system including a block diagram indicating all channels and functions and principal characteristics of the design; e.g., neutron flux range covered by each channel, range switching sensitivities, readouts, signals to other systems, and test provisions.

2. A general description of major component parts of the system such as detectors, preamplifiers, log and linear amplifiers, rate circuits, and trip and logic devices. The description should be supported by schematic diagrams of any novel circuits.

3. A drawing showing the physical location of detectors, control rods, sources, and reflectors.

(b) Evaluation

1. An evaluation of the adequacy of the provisions for monitoring reactivity during subcritical operation, startup, refueling, etc.

2. An evaluation of the effect of loss of power.

3. Analysis from both a functional and reliability viewpoint including considerations such as susceptibility to and effects of component failure and malfunctions, environmental conditions (including a description of environmental tests), redundancy achieved, independence of redundant channels, and testability.

4. Analysis to assure that requirements imposed by the necessity to maintain operation within safe limits are met. Specific reference should be made to the safety analyses, under Section XIV, for these requirements.

5. An evaluation of the adequacy of neutron sources for initial and subsequent startups.

(2) <u>Nonnuclear Process Instrumentation</u>

(a) <u>Design</u>

A general description of the design of this category of instrumentation including an instrumentation flow plan according to Instrument Society of America Recommended Practice ISA-RP5.1 (or equivalent) for the primary plant and any part of the secondary plant having safety significance. A brief description of the major components should be included.

(b) <u>Evaluation</u>

1. Analysis from both a functional and reliability viewpoint, including considerations such as susceptibility to and effects of component failures and malfunctions, environmental conditions (including a description of environmental tests), diversity and redundancy and testability.

2. Analysis to assure that requirements such as sensing of variables and the accuracy and response times imposed by the necessity to maintain operation within safe limits are met. Cross referencing should be made to the Safety Analyses, Section XIV, as appropriate, for these requirements.

(3) <u>Radioactivity Instrumentation</u>

This category of instrumentation might be subdivided into three groups: (1) instrumentation provided as radiation monitors for the control of routine release of radioactive effluents; (2) instrumentation provided for the protection of operation personnel; and (3) instrumentation that signals for containment isolation and brings other reactor engineered safeguards into action. It is desired that only the latter be discussed in this section. Information of the general type suggested under "Nuclear Instrumentation" on page 40 should be provided.

Process Radiation and Area Monitoring Instrumentation should preferably be discussed in Section XI.

E. <u>Operating Control Stations</u>. A detailed layout of the control stations is not required in the Safety Analysis Report; however, sufficient information should be provided to show what actions can be controlled and monitored from the control stations, particularly those available during accident conditions which allow operators to cope with the emergency. A brief description of the control station covering the following matters should normally be sufficient:

1. General layout.
2. A summary of essential information displayed and recorded.

3. Summary of the functions that are manually controlled from each station.

4. A summary of audible and visual alarms.

5. Principal means of communication between control rooms and other areas of the facility.

6. Occupancy requirements.

7. Dependence, if any, upon auxiliary stations for normal or emergency plant control.

8. Features considered significant to the enhancement of safe operation.

SECTION VIII - ELECTRICAL SYSTEMS

All reactors require coolant pump power when operating and for some time immediately following shutdown. In addition, the functioning of protective circuitry and the safeguard systems brought into action with the sensing of potentially unsafe situations depends upon the availability of electrical power. In this section of the Report, the applicant is expected to discuss the electrical power distribution system of the facility with emphasis upon those features whereby continuity of power sufficient for safety needs is assured.

The following illustrate the types of evaluations and supporting information that should be included in this section.

A. Design Bases. This subsection should identify those functional requirements established to provide for continued power to essential equipment in the facility in the event of failure in any part of the system.

B. Electrical System Design. As in the subsection on reactor design, the presentation which follows combines in the same subsection the description of distribution networks and evaluation from the standpoint of providing continuity of power. Alternate formats are acceptable as long as the information establishes not only what electrical systems are provided but why the particular arrangement is considered suitable.

(1) Network Interconnections

(a) A schematic single line diagram showing tie-in of the nuclear plant to the grid and the availability of incoming lines as alternate sources for station loads should be included.

(b) An evaluation of the reliability of the arrangement in providing continuity of power should be made, including such things as:

1. A resume of credible faults capable of causing the loss of more than one source of power.

2. Effectiveness of the protective features provided to minimize power failure.

3. Local environmental factors that might create unusual demands upon alternate or standby power.

(2) Station Distribution System

(a) Single line diagrams as appropriate for identifying arrangement of station buses, major loads and switching circuitry shall be included.

(b) Evaluation of the electrical layout for vulnerability to physical damage to vital circuits as a result of accidents should be made.

(c) Evaluation of the distribution of loads to the various buses in the interests of maintaining power to vital loads.

(d) Evaluation of possible step or ramp electrical load changes either through operator error, equipment malfunction, or environmental disturbances.

(3) Emergency Power

(a) Line diagrams showing interconnection of locally generated standby and emergency power and the loads to be fed by such sources.

(b) Brief descriptions of emergency power sources with emphasis on those features provided to protect the supplies from faults, overloads or damage under emergency conditions which could interfere with performance.

(c) Evaluation of loads required to be powered in the interests of safety and the relationship of the emergency load to installed capacity.

(d) Performance history of emergency sources of power and the possible effects of failure to deliver should an emergency arise.

D. Tests and Inspections. Identification of principal tests and inspections planned to assure availability of power for safety functions. Emphasis should be given to testability and testing of those systems not normally operating.

SECTION IX - AUXILIARY AND EMERGENCY SYSTEMS

This section of the Safety Analysis Report should provide information concerning the auxiliary and emergency systems included in the facility. Examples of the systems which are considered to be of this category are:

a. Heat removal systems (i.e., reactor shutdown cooling, emergency cooling, component cooling, area cooling, radiation or thermal shield cooling).

b. Chemical injection systems (e.g., nuclear poison fluid injection systems, chemical additions for corrosion control).

c. Reactor coolant purification systems (e.g., reactor coolant cleanup, treatment and degasification).

d. Reactor coolant volume and pressure adjustment systems (e.g., reactor coolant make-up, charging systems).

e. Auxiliary heating systems (e.g., reactor coolant temperature control before plant startup, or after shutdowns).

f. Reactor coolant leakage control systems (e.g., seal water systems, vents, and drains).

g. Reactor coolant sampling system.

h. Equipment and systems decontamination processes.

i. Facility service systems (e.g., fire protection, instrument air, and service water).

j. Reactor component handling systems (e.g., handling of fuel assemblies, control rod assemblies, core structural components, and material irradiation specimens).

Each system might be considered as a supporting system upon which the safe operation or servicing of the reactor coolant system (Section IV) depends to a varying degree. In some cases, the dependable operation of several systems is required to protect the reactor coolant system by controlling system conditions within specified operating limits. Other systems are called upon to operate under emergency conditions.

Information to be presented in this section should emphasize those systems in which component malfunctions, inadvertent interruptions of system operation, or a complete system failure may lead to a hazardous or unsafe condition. The extent of information provided for each system should be weighed by the relative contribution of, or reliance placed upon, each system with respect to the overall plant operational safety. In some cases, auxiliary systems which are not required to operate continuously during normal plant operations (e.g., decontamination systems) may be adequately treated with limited information. On

the other hand, emergency systems which, although operated intermittently, are relied upon to control conditions within specified limits may warrant more extensive treatment (e.g., emergency cooling systems).

The following are illustrative of the types of evaluations and supporting information that should be included for systems discussed in this section:

A. Heat Removal, Auxiliary, and Emergency Systems. Evaluations for the types of systems illustrated by items a. through i., above, including the design bases and schematic flow diagram of each system supplemented by, as applicable:

(1) System performance requirements and the conditions under which the system must operate.

(2) The multiple functions assigned to a system and mode of operation for each function.

(3) The availability and reliability considerations (e.g., heat sink and component redundancy included and sharing among systems allowed).

(4) The codes and classifications (of vessels) applied to components and piping of those systems exposed to the reactor coolant.

(5) The extent of system insolubility (e.g., with respect to the reactor coolant system and the containment boundary).

(6) The provisions for handling radioactively contaminated fluids or gases which may leak into the system from the reactor coolant system.

(7) An analysis of consequences resulting from malfunctions, failure of system components, or loss of cooling source for those systems required to operate in an emergency which may lead to a hazardous condition.

(8) An analysis of limits on variables or conditions of operation that might be required in the interests of safety.

B. Reactor Component Handling System. Evaluations of the reactor component handling system with particular emphasis upon the fuel handling and transfer facilities supplemented by sectional drawings of the system and including, for example:

(1) The design bases and features of the system which assure that unirradiated fuel can be safely received, stored, and transferred into the reactor without inadvertent criticality.

(2) The features of the system design which will permit removal of irradiated fuel from the reactor for storage or shipment under all foreseeable conditions without excessive radiation, undue escape of radioactive materials or inadvertent criticality.

(3) The extent to which the operation of any auxiliary system is necessary in the performance of fuel handling or transfer operations.

(4) The conditions which will be imposed upon the facility with respect to the reactor shutdown security requirements, and the containment integrity when refueling or fuel transfer operations are scheduled.

(5) The extent of visual, mechanical, and electrical means provided to monitor refueling operations, fuel transfer, and fuel storage.

(6) Safety provisions incorporated in the system to prevent the development of hazardous conditions in the event of component malfunctions, accidental damage, or operational and administrative failures during refueling or transfer operations.

(7) Those bounds to be established as limits in the interest of safety with explanations therefor.

C. Tests and Inspections. Identification of principal tests and inspections planned for the surveillance of the auxiliary and emergency systems to assure continued acceptability of performance required for safety:

(1) A summary of test and inspection programs to be followed for each system.

(2) Identification and explanation of those tests and inspections considered to be of such importance to safety as to be proposed for inclusion in the technical specifications.

SECTION X - STEAM AND POWER CONVERSION SYSTEM

This section of the Safety Analysis Report should provide information concerning the facility steam and power conversion system. For purposes of this section, the steam and power conversion system (heat utilization system) should be considered to comprise:

(a) The steam system and turbine generator units of an indirect-cycle reactor plant, as defined by the secondary coolant system, or

(b) The steam system and turbine generator units in a direct-cycle plant, as defined by the system extending beyond the reactor isolation valves.

There will undoubtedly be many aspects of the steam portion of the facility that have remote or no relationship to public protection against exposure to radiation. The Safety Analysis Report is, therefore, not expected to deal with this part of the facility to the same depth or detail as those features playing a more significant safety role. Enough should be provided in the way of information to allow understanding in broad terms of what the secondary plant (steam and power conversion system) is, but, as stressed in previous sections, emphasis should be on those aspects of design and operation that do or might affect the reactor and its safeguard features or contribute toward the same safety objective of control of radioactivity. The capability of the system to function without compromising directly or indirectly the nuclear safety of the plant under both normal operating or transient situations should be shown by the information provided.

The following illustrates the types of evaluations and supporting information that should be included in this section:

A. Design Bases. The principal design bases for the steam and power conversion system, including, for example:

(1) The operating and performance requirements of the steam and power conversion system under both normal operating and transient conditions.

(2) The characteristics imposed on the steam and power conversion system by the electrical power distribution system (e.g., base loaded or load following operations).

(3) The functional limitations imposed on the steam and power conversion system by the design or operational characteristics of the reactor coolant system. (e.g., rate at which electrical load may be increased or decreased with and without reactor control rod motion or steam bypass.)

(4) The secondary functions assigned to the steam and power conversion system upon which the operability of the system depends (e.g., motive steam to the main air ejectors).

B. System Design and Operation. The principal design features of the system which may be illustrated or supplemented by:

(1) A schematic flow diagram of the system denoting major components; principal pressures, temperatures, and flow rates under normal steady state full power operation; and the volume in system.

(2) A piping and instrumentation diagram of the system including those subsystems which, by virtue of steam or condensate flow, extend the fluid boundary of the steam and power conversion system, such as reheating, superheating, or desuperheating systems; make-up and blowdown systems; steam dump or bypass systems; and air ejector exhaust system; delineating on the diagram:

(a) The extent of any section of the system located outside the containment boundary:

(b) The extent of isolability of any section of the system or its subsystems as provided by the use of isolation or bypass valves; and

(c) The location of pressure-relieving devices in the system and their associated blowdown or heat dissipation system.

(3) The codes and classifications (of vessels) applied in the design, construction, inspection, and testing of the principal components and piping of the steam-power conversion system, and the primary side of the subsystems connected thereto (i.e., subsystems through which radioactively contaminated steam or condensate may flow).

(4) Unusual features of design, construction, or operation which may have a relation to safety; (e.g., radiation monitoring and holdup system of condenser air ejector off-gas).

(5) Shielding (if required) employed for individual components of the system, (e.g., turbine, condenser air ejectors, piping) and the extent of access control which may be exercised based on the shielding provided.

(6) The requirements for additives whose control is intended as a preventative measure in limiting corrosion of the system.

(7) Provisions employed to control impurities (e.g., cleanup systems).

(8) Normal operating concentrations of radioactive contaminants in the system.

C. System Analysis. Evaluations applied to the steam-power conversion system in support of its performance capability and integrity requirements under normal and transient operating conditions, including, for example:

(1) The functions of trips, interlocks, or alarms actuated by potentially harmful deviations of system variables (e.g., condenser vacuum, steam generator feedwater level) to control their influence upon the operation of the reactor coolant system.

(2) The analyses of the effects of pressure, temperature, and flow transients originating in the steam-power conversion system on reactor system behavior. (e.g., loss of load following an emergency shutdown of the turbine generator from full load conditions.)

(3) The analyses of the effects of inadvertent manipulation or malfunction of controls (e.g., steam bypass, dump, or turbine throttle valve rapid closure) of the steam-power conversion system on reactor system transients.

(4) The analyses of the capability of the safety valve system to provide overpressure protection of the system and its components including:

(a) The system conditions under which pressure relief is required and which dictated the maximum pressure-relieving requirements.

(b) The energy dissipation or flow capacity of the pressure-relieving system as related to the thermal characterisitcs of the reactor coolant system.

(c) The pressure setting or bursting pressures of the pressure-relieving devices.

(5) Evaluation of the capability of the system (in direct-cycle plants) to perform as a dependable barrier to the escape (within specified limits) of accumulated radioactivity in the steam flowing through the entire system or extensions thereof (subsystems), during normal or routine operation.

(6) Evaluation of the effect of interactions between the steam-power conversion system and the reactor coolant system (or reactor) in the event of malfunctions, or operational failures of components in the steam-power conversion system, during normal or routine operation. (e.g., turbine trips or feedwater pump failures, condenser failure).

(7) Evaluation of the capability of the steam-power conversion system located outside of the containment structure to limit or control the loss of radioactivity in the event fission products released in the reactor coolant system enter the steam-power conversion system (e.g., steam generator tube failure in indirect-cycle plants).

(8) Evaluation of operations to ascertain the operational restrictions, if any, that should be established as safety limits or safety conditions in the interests of health and safety of both plant personnel and the public. The stress should be on protection of persons, not equipment per se.

D. Tests and Inspections. Identification of principal tests and inspections planned to assure functional performance as required for continued protection of persons; e.g., proper functioning of isolation valves and steam bypass valves.

SECTION XI - RADIOACTIVE WASTES AND RADIATION PROTECTION

10 CFR Part 20 establishes standards for protection against radiation hazards arising out of activities under license by the AEC. The standards set forth maximum permissible limits of exposure of individuals and maximum permissible concentrations of radioactivity that can be released to the air and water environment of a nuclear facility. In this section of the Report, information should be provided that identifies expected sources of radioactive wastes, the aspects of system designs provided to maintain control over them and the monitoring to be done to show that predefined maximum limits on concentrations or radiation levels to which people might be exposed will not be exceeded.

Radioactive Wastes

Information on the sources of radioactivity and the systems designed to process and control them should include such things as the following:

A. Design Bases. The bases upon which process system designs have been established, including, for example:

(1) The estimated maximum and average volume rates of accumulation of the various forms of radioactive wastes, identification of principal sources, estimates of isotopic content and activity concentrations, and the basis for each estimate.

(2) The selected method of disposal for each type of radioactive waste, and identification of the limiting radioactivity concentration or release rate, or radiation level established for each type, and the basis for each such limit.

B. System Design and Evaluation. Information about the systems that will process and control the radioactive materials should identify principal features of design including, for example:

(1) The collection and processing equipment for the solid waste disposal system, including storage and packaging facilities.

(2) A schematic flow diagram of the liquid waste disposal system, denoting all principal components for collection and processing, and all points of discharge to the environment.

(3) A schematic flow diagram of the gaseous waste disposal system, denoting all principal components for collection, processing, and discharge; and all points of release to the atmosphere.

(4) The process radiation monitoring instrumentation with a schematic indication of the types of detectors, their members, relative locations, sensitivities and such other information needed to show what process controls are being maintained.

(5) Evaluation of process systems and controls for capacity and capability to maintain radioactivity limits within 10 CFR Part 20 standards. Evaluation made of systems and precedures for handling wastes would normally be expected to include analyses such as:

(a) Dilution effects of the air or water into which radioactivity is routinely released; e.g.: derivation of dilution factors obtained through diffusion from the stack exit to the site boundary where such natural dilution is required in order to meet off-site radioactivity concentration limits. Supporting data, basic assumptions, and factors of conservatism should be cited. Dilution factors for both instantaneous and continuous releases should be given and the methods of averaging over specific time periods should be presented.

(b) Environmental factors as they might affect operations with respect to efficient control.

(c) Possibilities for equipment malfunction or component failures that might cause releases to exceed Part 20 limits.

(d) The need and basis for safety limits on process variables that should not be exceeded or requisite equipment that should be available in compliance with Part 20.

C. Tests and Inspections. Guidance provided in other subsections relating to tests and inspections is applicable here and will not be repeated.

Radiation Protection

In this section, those features of design and operation of the facility provided primarily for protection of operating personnel should be discussed, including, for example, such matters as shielding, area radiation monitoring and health physics considerations.

The shielding evaluations and supporting information provided should be limited to the principal shielding installations; those installations usually referred to as the primary shield and the secondary, or containment, shield. The miscellaneous shielding provided elsewhere throughout the facility, the shielding provided for the waste disposal and purification systems, and that associated with refueling, for example, should be treated in those sections of the report wherein the particular system or activity is treated.

The Commission recognizes that the shielding effectiveness in reducing radiation in areas to which personnel will have access will be determined under operating conditions by surveys. Accordingly, minimal information with respect to design details or calculational assumptions will be sufficient.

However, certain sections of shielding are often designed and installed to provide radiation protection in the unlikely event of a major accident. For this type of shielding complete testing may be necessary and assurance of adequacy under accident conditions must, in some part, be based on analysis. For such cases, the analysis should be included in the Safety Report.

Some radiation monitoring systems are installed to monitor the operational status of another system rather than to provide information associated with the direct protection of people. The radiation monitors installed in a coolant purification system to monitor demineralizer efficiency is an example of this type of system. A monitoring system used for detection of fuel element failures is another. The common characteristic of these systems is that they are used for operational control, and are not directly associated with establishing whether or not radiation levels and concentrations in a particular area are acceptable from an occupancy viewpoint. The former systems, those used for operational control, should be discussed generally in the section wherein the process system being monitored is treated. <u>The radiation monitoring systems considered in this section should be those directly involved in measurement of levels and concentrations of radioactivity to which people may be exposed</u>.

In addition to information on shielding and radiation monitoring, the principal health physics control features should be explained; e.g., personnel monitoring, protective equipment, laboratory facilities, work areas, and contamination control.

The following are illustrative of the types of evaluations and supporting information that should be included in this section:

A. <u>Primary and Secondary Shielding</u>. Information on the primary and secondary, or containment, shielding, including, for example:

(1) <u>Design Bases</u>

The design bases for the shielding, including those bases associated with normal operation, with shutdown conditions, and with conditions resulting from major credible accidents.

(2) <u>Design</u>

Supporting information for the shield designs, including, for example:

(a) Locations, configurations, principal dimensions, and arrangements with respect to radiation sources.

(b) Principal shield materials.

(c) Methods of support and provisions for cooling, insulating, corrosion protection, and expansion to insure continued effectiveness and structural integrity under all credible conditions.

(3) Evaluation

(a) For those shield sections that cannot be fully tested, a presentation of the shield attenuation analysis, including:

1. Basic calculational methods and models.

2. Assumed radiation sources, source levels, and values of significant parameters, and the basis for each assumption.

3. Calculated surface and positional dose rates during normal operation and shutdown, and in the event of accident.

(b) An evaluation of any significant limits, such as those which might be associated with allowable shield temperatures, or minimum levels of liquid shields, and the bases upon which each limit was selected.

(c) Plans for radiation surveys.

B. Area Radiation Monitoring Systems. A discussion of area radiation monitoring systems, including, for example:

(1) Design Bases

The design bases for the area radiation monitoring systems in terms of the protective functions to be performed.

(2) Design

A layout of the area radiation monitoring systems, including types of detectors and their numbers, locations, sensitivities, calibration, methods of initiating alarms, and locations of alarms and indicators. Particular attention should be given to systems provided primarily or solely for the analysis of conditions following an accident.

(3) Evaluation

An evaluation of any significant operating limits, including, for example, minimum numbers of detectors that must be operable during specific modes of operation, and the maximum time a given system may be out of service during facility operation. The basis upon which each limit was selected should be explained.

C. Health Physics. A discussion of principal health physics control features, including, for example:

(1) The availability of personnel monitoring systems, such as film badges, pocket dosimeters and pocket high radiation alarms and the frequency of determination of personnel radiation exposures.

(2) Personnel protective equipment to be available, including such items as face masks, self-contained air-breathing units, and protective clothing. The intended methods of use and maintenance of the equipment should be summarized.

(3) Change room facilities, personnel and equipment decontamination areas, contaminated work areas, and methods to be used to control access to high radiation areas.

(4) A brief description of health physics laboratory facilities, if any.

(5) Availability of portable radiation survey instruments and other health physics instrumentation such as hand and foot counters, sample counters, and multichannel analyzers, including their numbers, types and methods of use.

(6) Bio-assay and medical examination programs.

D. Tests and Inspections. Guidance provided on other subsections about tests and inspections is equally applicable here and will not be repeated.

SECTION XII - CONDUCT OF OPERATIONS

This section of the Safety Analysis Report should provide information relating to the framework within which operation of the facility will be conducted.

The Commission recognizes that the operation of the facility entails a myriad of instructions and procedures of varying detail for the operating staff. The Commission has no intent to review such procedures in detail and, therefore, they should not be included in the Safety Analysis Report. The Commission is interested, however, in how an applicant generally intends to conduct operations. Part of the Commission's safety review is directed toward assuring that a licensee will maintain a technically competent and safety-oriented staff.

The following illustrate the kinds of information desired:

A. Organization and Responsibility. A description of the applicant's organization for operation of the nuclear reactor facility, including, for example:

(1) A functional description of the structure of both the startup and the regular operating organizations.

(2) Qualifications established as prerequisites for key positions within the organization.

(3) The lines of responsibility and authority for both operation and safety.

(4) The approximate number of personnel to be assigned to the organization as a whole, and to individual operating and safety branches.

B. Training. Identification of staff training programs including, for example:

(1) The program for training of the operating staff for initial startup and routine operations.

(2) The program for training of replacement personnel for the operating organization.

(3) On-the-job training and upgrading operating personnel.

(4) Emergency drills in coordination with local police, fire, rescue teams, and similar groups.

C. Written Procedures.

(1) Plans for having available written detailed procedures for normal plant operations, and expected deviations from normality.

(2) Precautionary planning for the unlikely yet credible accident which might result in radioactivity release to the public.

D. Records. Identification of principal records to be maintained; e.g., daily logs of operation, records of tests, inspections, and measurements, records of radioactivity released to the environs, records of monitored safety limits, records of changes made to the facility and the operating procedures, and records of safety audits of operation.

E. Review and Audit of Operations. A discussion of the methods to be used for review and audit of operations, procedures equipment, changes to procedures, changes to equipment and incidents.

SECTION XIII - INITIAL TESTS AND OPERATION

This section of the Safety Analysis Report should provide information relating to the period of initial operation, with particular emphasis on tests planned to demonstrate the degree to which the facility does, in fact, meet design criteria. Explanations for any special limits, conditions, surveillance requirements, and procedures to be in force during the initial period of operation and until such time as acceptable design performance is demonstrated should be included in this section.

Throughout prior sections of the Report, limits, conditions, surveillance requirements, and procedures for facility operation may have been established. For some facilities, however, these may be made more restrictive during the period of initial operation and relaxed to their final condition only as actual operation demonstrates their acceptance from a safety viewpoint. Such matters should be discussed in this section.

The following guides are illustrative of information that should be included:

A. Tests Prior to Reactor Fueling. A tabulation of significant tests of components and systems to be conducted prior to initial fueling of the reactor. The tests should be tabulated in the sequence in which they are proposed to be performed. A brief statement as to the essential purpose of each test should be included.

B. Initial Criticality. Plans for approach to criticality should be briefly described, including such safety prerequisites for initial loading and a resume of principal procedures and safety precautions to be in effect during these tests. The reasons for these precautions should be explained.

C. Postcriticality Tests. Tests planned following initial criticality should be identified; e.g., zero and low power tests, stability tests, shielding surveys, escalation to full power. A summary in anticipated sequence of their conduct with principal reason for each should be included. Any special safety precautions needed should be identified and explained.

D. Operating Restrictions.

(a) A summary of safety precautions, which require more conservative setting on process variables or the availability of equipment for initial tests than is planned as a normal mode or operation, should be included in tabular form, together with the normal value or condition. Information provided in this section should explain the reasons for the differences.

(b) A similar summary should be included of special procedures to be in effect during initial operation.

SECTION XIV - SAFETY ANALYSES

Safety of a reactor plant as achieved through design is commonly shown by studies made of the response of the plant to disturbances in process variables and to postulated malfunctions or failures of equipment. Such safety analyses provide a significant contribution to the design specifications for components and systems and subsequently serve importantly in showing that a design consistent with public safety has been achieved. These analyses are a focal point of Commission prelicensing review of a reactor facility.

In previous sections of this guide, it has been stated that the individual system and component designs should be evaluated for effects of anticipated process disturbances and for susceptibility to component malfunction or failure. In this section, it is expected that the consequences of those failures or abnormal situations will be developed and the capacity built into the plant to control or accommodate such situations shown (or the limitations of expected performance revealed).

It is recognized that situations analyzed may range from a fairly common disturbance such as load loss from a line fault to the remote possibility of loss of integrity of a major component.

Treatment of this range of situations in two parts is suggested as follows:

Part I:

Analysis of those situations that bring into play the safety features of design required to protect the core and coolant boundary.

Part II:

Analysis of those situations that require operation of standby safeguards because of postulated failures which create demands beyond the capacity of core and boundary protective systems.

Part I. Core and Coolant Boundary Protection Analysis

The reactor control and protective systems are intended to provide a high degree of assurance that combinations of power, coolant flow, temperature and pressure will not cause thermal limits set for the fuel to be exceeded. Exceeding such limits means generally that some fuel may fail. This in itself may not represent a safety problem. It is common practice, however, to treat evidence of fuel failure beyond expectation from steady state normal operation as evidence of an undesirable increase in vulnerability to an unsafe situation. As such, both design and operation are aimed, in the interests of safety, at a capability for handling a variety of deviations from normal steady state conditions without significant fuel failures.

The abnormal situations that were analyzed and the conclusions that were reached with respect to plant response should be shown through information such as the following:

A. Abnormalities. A summary in tabular form of deviant events analyzed should be provided; e.g., such situations as:

(1) Anticipated variations in the reactivity load of the reactor, to be compensated by means of action such as buildup and burnout of xenon poisoning, fuel burnup, on-line refueling, fuel followers, temperature, moderator and void coefficients.

(2) Reactivity excursions from such causes as uncontrolled rod withdrawals, mechanical content rod failure, cold coolant insertion, loss of coolant.

(3) Anticipated electrical load disturbances, with effects to be controlled by action such as a turbine trip.

(4) Failure of the regulating instrumentation, causing for example, a power-coolant mismatch.

(5) Situations created by malfunction of components other than instrumentation such as loss of a coolant pump.

(6) Possibilities for equipment failures involving loss of component integrity which shifts safety action of instrumentation from one of prevention to one of initiating protective safeguards against the release and dispersal of radioactivity.

(7) External causes such as storms or earthquakes.

B. Identification of Causes. For each situation evaluated there should be included a description of the events that must occur, the order of occurrence and analysis of effects and consequences and the basis upon which credibility or probability of occurrence is determined.

The evaluation should show the extent to which reactor protective systems must function, the effect of failure of protective functions, the credit taken for designed-in safety features, reactor protective characteristics, or the performance of backup protective systems, during the entire course of the situation analyzed.

C. Analysis of Effects and Consequences. The analysis of effects and the attendant consequences should be supported by sufficient information, including, for example:

(1) The methods, assumptions, conditions, employed in estimating the course of events and the consequences.

(2) The mathematical or physical model employed denoting any simplification or approximations introduced to render the analyses tractable.

(3) Identification of any digital computer program or analog simulation used in the analysis with principal emphasis upon the input data and the extent or range of variables investigated.

(4) The results and consequences derived from each analysis and the margin of protection provided by either a backup or protective system which is depended upon to limit the extent or magnitude of the consequences.

(5) The considerations of uncertainties in calculational methods, in equipment performance, in instrumentation response characteristics, or other indeterminate effects taken into account in the evaluation of the results.

Part II. Standby Safeguards Analysis

The analysis provided in Part I should reveal the capacity of the facility for handling abnormal events short of loss of integrity of the coolant boundary. They also should show the limitations of the accident-preventative features of design. Part II should include analyses of those situations requiring operation of standby or engineered safeguards because of postulated equipment failures which would create demands beyond the capacity of the core and coolant boundary protective systems.

A. Situations Analyzed. A listing of those events analyzed such as loss of coolant, rod ejection, spent fuel handling, and inadvertent criticality should be provided.

B. Identification of Causes. For each situation evaluated, there should be included explanatory material on how such an event might be initiated and what sequentially must take place to require the standby safeguards to function.

C. Accident Analysis. Analysis should show the extent to which protective systems must function. Where standby systems provide redundant or backup protective functions, the analysis should show not only the effects if expected performance is achieved, but also effects if a degraded performance is experienced. The purpose of such an exercise is to show the sensitivity of protective results to system effectiveness. This section should include such things as:

(1) Explanation for conditions and assumptions associated with the accidents analyzed, including any reference to published data or research and development investigations in substantiation of the assumed or calculated conditions.

(2) The time-dependent characteristics, activity, and release rate of the fission products, or other transmissible radioactive materials within the containment system which could escape to the environment via leakages in the containment boundaries.

(3) The extent of systems interdependency (containment system and other engineered safeguards) in contributing directly or indirectly to controlling or limiting leakages from the containment system, such as the contributions of:

  (a) Containment water spray systems
  (b) Containment air cooling systems
  (c) Containment atmosphere filtration systems
  (d) Reactor core spray or safety injection systems
  (e) Postaccident heat removal systems.

(4) The physical or mathematical models assumed in the analyses and the bases thereof with specific reference to:

  (a) Distribution and fractions of fission product inventory assumed to be released from the fuel.

  (b) The concentrations of radioactive or fission product inventory airborne in the containment atmosphere during the postaccident time intervals analyzed.

  (c) The mass transport phenomena applied to the radioactively contaminated containment atmosphere and its time-dependent leakage and dispersion to the environment.

  (d) The conditions of meteorology, topography or other circumstances, and combinations of adverse conditions, considered in the analyses.

(5) Summary of the calculations of estimated dose rates and integrated doses from exposure to radiation as a function of distance from the containment system and time after the accident, including:

  (a) The time-dependent effects of radiation intensity from a contained source (within containment system).

  (b) The time-dependent effects of source intensities associated with the plume transported downwind from the containment building.

  (c) The time-dependent effects of source strength from ground deposition contributed by impaction, absorption, or adsorption after vertical transport from plume or by rainout.

  (d) The time-dependent whole body doses to potential receptors on the ground.

  (e) The time-dependent doses from inhalation or ingestion by various body organs.

SECTION XV - TECHNICAL SPECIFICATIONS

A principal function of the Safety Analysis Report is to provide sufficient technical information to demonstrate to the Commission that the nuclear facility has been designed and constructed, and is to be operated in such manner that no undue risk to the health and safety of the public will result. This finding of safety is a prerequisite to the granting of an operating license.

In accordance with the Atomic Energy Act, each operating license issued by the Atomic Energy Commission must contain technical specifications. In general, technical specifications include those technical operating limits, conditions, and requirements imposed upon facility operation in the interests of the health and safety of the public. The applicant for an operating license proposes technical specifications for his facility which are reviewed by the AEC and modified as necessary before becoming a part of the operating license.

Throughout the previous sections of this guide, the necessity for identification of safety limits, conditions and surveillance requirements has been indicated. It is from such information that technical specifications and supporting analyses are developed. The technical specifications proposed by an applicant for his facility should be included as an appendix to the Safety Analysis Report.

It is beyond the scope of this guide to provide detailed guidance on the contents of technical specifications. For such purpose, reference should be made to 10 CFR Part 50, § 50.36.

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