POLICY ISSUE
(Information)

June 11, 1992

SECY-92-214

For: The Commissioners

From: James M. Taylor
Executive Director for Operations

Subject: DEVELOPMENT OF INSPECTIONS, TESTS, ANALYSES, AND ACCEPTANCE CRITERIA (ITAAC) FOR DESIGN CERTIFICATIONS

Purpose: To inform the Commission of the status of the development of ITAAC for certified standard designs. This paper addresses issues identified in SECY-91-178, "Inspections, Tests, Analyses, and Acceptance Criteria for Design Certifications and Combined Licenses" (WITS 9100215 and 9100217), and SECY-91-210, "Inspections, Tests, Analyses, and Acceptance Criteria Requirements for Design Review and Issuance of a Final Design Approval."

Summary: The staff and industry are developing ITAAC for certifying standard designs and senior management is participating directly in many aspects of the development process. The GE Nuclear Energy (GE) Advanced Boiling Water Reactor (ABWR) is the lead plant for developing ITAAC for the evolutionary designs. A complete, high quality ITAAC submittal for the ABWR has not yet been received, and this may impact the current schedule for the FDA. The submission and review of ITAAC for the CE 80+ design may also have an impact on the CE 80+ review schedule. In continuing to develop and review ITAAC, the staff is implementing several aspects of 10 CFR Part 52.

NOTE: TO BE MADE PUBLICLY AVAILABLE IN 3 WORKING DAYS FROM THE DATE OF THIS PAPER

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Background: The requirement to provide ITAAC for a design certification application is contained in 10 CFR 52.47(a)(1)(vi). The staff has discussed various issues associated with ITAAC in two previous Commission papers. In SECY-91-178, the staff discussed the form and content of ITAAC. In SECY-91-210, the staff discussed the relationship between the development of ITAAC and the issuance of a final design approval (FDA).

Discussion: In the staff requirements memorandum (SRM) for SECY-91-178, the Commission directed that the staff keep the Commission informed on the continuing interaction with industry on ITAAC and address a number of other related items. The following discussion responds to that SRM. One item, related to the sign-as-you-go process, has been addressed in a separate paper, SECY-92-134, "NRC Construction Inspection Program for Evolutionary and Advanced Reactors Under 10 CFR Part 52."

STATUS OF THE ITAAC REVIEW

The GE ABWR is the lead plant in the development of the first ITAAC for a standardized design. The NRC staff and GE are developing ITAAC in conjunction with the design certification review for the ABWR. GE is submitting the ITAAC to the staff for review in phases. Senior management meetings with GE are being held at intervals of approximately six to eight weeks in order to resolve difficult issues. In 1991, the staff held numerous meetings with GE and the Nuclear Management and Resources Council (NUMARC) on this subject, and GE initially submitted the first phase in September 1991. This consisted of a set of nine "pilot" system ITAAC for the staff to review. After extensive interaction between the staff and GE, general agreement on the pilot ITAAC was reached in January 1992. The staff and GE met in February, and again in March, to further discuss the details of ITAAC, future submittals, and interfaces for the design. GE submitted the second phase ITAAC on April 8, 1992, which consists of approximately one-half of the ITAAC. As a part of its review of the ITAAC submitted by GE to date, the staff has found that there are several problems that have affected the staff's review. The staff has identified some significant inconsistencies between the SSAR, the Tier 1 design descriptions, and the ITAAC. GE has informed the staff that this is due to the iterative nature of the ITAAC development process and GE's limited resources. This constrained GE involvement has contributed to technical errors occurring in material provided to the staff and has resulted in interim milestone schedular delays. The staff has continued to inform GE of these discrepancies, and their potential negative impact on the current schedule, in formal comments and at senior management meetings.

GE stated that a complete ITAAC submittal would be provided in the third stage ITAAC submittal due on May 31, 1992. However, the area of Human Factors and Control Room Design was not included when the ITAAC were actually submitted on June 1, 1992. GE has stated that this area will be submitted in June 1992. In accordance with SECY-91-161, "Schedules for the Advanced Reactor Reviews and Regulatory Guidance Revisions," the staff is scheduled to issue a final safety evaluation report (FSER) in August 1992. As indicated in the Quarterly Status Report of Advanced Light Water Reactor Reviews (December 1991 - February 1992), dated April 9, 1992, due to the need for additional information from GE, the staff intends to supplement the FSER following its August 1992, issuance to address several issues including GE's June 1, 1992, ITAAC submittal. The August 1992 FSER issuance is expected to provide the Commission and the Advisory Committee on Reactor Safeguards (ACRS) with the bulk of the staff's safety findings including an evaluation of the ITAAC submitted by GE through phase 2. However, GE will need to provide all remaining design information, including ITAAC-related information, on a schedule which will support the issuance of the planned FSER supplement and the subsequent FDA.

The Combustion Engineering, Inc. (ABB-CE) submittal came after the GE ABWR ITAAC. The staff and ABB-CE met in March and May 1992 to discuss issues associated with the CE System 80+ ITAAC. ABB-CE submitted a pilot set of ten ITAAC on April 30, 1992. Based upon the staff's review of ABB-CE's initial submittal, the staff believes significant review resources will be required to complete the System 80+ ITAAC review. The full ITAAC submittal for the System 80+ design is expected in the fall of 1992. Because of the interaction between ITAAC and the technical review of the SSAR, it is not practical to do these reviews in series. The staff will advise the Commission of the impact on the FSER schedule after it receives and initially reviews the complete CE System 80+ ITAAC submittal.

TYPES OF ITAAC

GE has developed two types of ITAAC for the ABWR design. The ABWR ITAAC consist of "systems" ITAAC for the systems of the design, and "generic" ITAAC for generic requirements which extend across multiple systems. The generic ITAAC will be referenced by the systems ITAAC where applicable for any particular system.

GE is developing ITAAC for approximately 85 systems of the design. These ITAAC are evolving through intense interactions between the staff and GE, and examples of system and generic ITAAC are presented in this paper for illustrative purposes only. Enclosure 1 contains an example of a system ITAAC for the standby liquid control system (SLCS), and also includes the SLCS Tier 1 design description which will be certified in the rule. Enclosure 2 contains an example of a generic ITAAC for environmental qualification. An illustration of the relationship between the system and the generic ITAAC is shown in the sixth "certified design commitment" of the SLCS system ITAAC, which invokes the environmental qualification generic ITAAC. GE is continuing to develop the system and generic ITAAC and provided a partial submittal to the staff in the phase 2 ITAAC.

Since the ITAAC are primarily system oriented, the staff is considering additional ITAAC to be developed during the combined license (COL) review and proceeding which, in combination with the design certification ITAAC, would meet the necessary and sufficient standard to provide reasonable assurance that the facility has been constructed and will operate in conformity with the license, the provisions of the Atomic Energy Act, and the Commission's regulations that are necessary to support fuel loading. These COL ITAAC will verify implementation of both hardware (e.g., site-specific design features) and licensee specific "soft" procedural requirements (e.g., training, quality assurance, etc.). The staff will use the results of its construction and preoperational inspection programs to independently verify that both design certification and COL ITAAC have been met. The staff will work with the Office of the General Counsel on the development of both design certification and COL ITAAC.

An area of the review which requires unique treatment in ITAAC are the interface requirements between the plant design specified in the certification and the site-dependent design features which will be developed at the combined license application stage. Interface requirements for the design are required by 10 CFR 52.47(a)(1), which includes a stipulation that the method to be used for verification of the interface requirements be included in the ITAAC submittal. GE has identified several design elements for the site-specific portion of a facility, and is developing the information to meet the interface requirements. These non-certified portions of the design include site-specific elements such as the service water system, offsite power system, and the ultimate heat sink.

Another area of the review which requires unique treatment in ITAAC are design acceptance criteria (DAC). GE is developing DAC for certain areas of the design because they are areas of rapidly changing technology or because they require as-built or as-procured information. These areas include pipe stress analyses, radiation shielding and airborne concentrations, instrumentation and control systems, and control room design details. The use of DAC were described in SECY-92-053, "Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews," and SECY-92-196, "Development of DAC for the ABWR." The staff is interacting extensively with GE in the development of DAC, and GE has stated that it will provide all DAC submittals to the staff for review in the third phase of ITAAC submittals. The staff will provide examples of DAC for the ACRS and Commission review when they are submitted by GE.

REVIEW ISSUES

The process for developing the Tier 1 design descriptions for the systems and the ITAAC is implementing many aspects of 10 CFR Part 52. The staff and industry are working to appropriately implement the following issues. The staff is also preparing a Commission paper that discusses the format and content of a design certification rule and various issues associated with implementing 10 CFR Part 52. This paper is expected to be forwarded to the Commission in June 1992.

TREATMENT OF NON-TRADITIONAL ITEMS IN ITAAC

The applicant for design certification will incorporate into the SSAR (standard safety analysis report) any insights into the design that were obtained from non-traditional items such as probabilistic risk analysis (PRA) and severe accident issue resolutions. The staff has followed the Commission's guidance in its review of the evolutionary designs for the resolution of non-traditional issues such as those discussed in SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationships to Current Regulatory Requirements." In 10 CFR 52.47, the Commission specified that the ITAAC must provide reasonable assurance that "a plant which references the design is built and will operate in accordance with the design certi-fication." Therefore, by verifying key aspects and features of the design, the ITAAC implicitly confirm the implementation of these non-traditional items and the safety findings contained in the safety evaluation report.

The staff has requested GE to develop a cross reference of key aspects, analyses, and features of the design from the SSAR to the ITAAC in order to document how these issues have been incorporated into the ITAAC. Specifically, the cross reference will show how key aspects of the accident analyses, PRA, and severe accident issue resolutions are included in ITAAC. Enclosure 3 contains a preliminary version of this cross reference for containment performance analyses. For example, the PRA has shown that the vacuum breakers are a key component in analyzing the behavior of the containment for severe accidents. Thus, the ITAAC for the containment system will include the key facets of the vacuum breakers which are significant contributors to risk for those analyses.

RELATIONSHIP OF THE DESIGN DESCRIPTION TO THE ITAAC

In 10 CFR 52.47(a)(1)(vi), the Commission stated that ITAAC must be provided which are "necessary and sufficient" to provide reasonable assurance that a facility which references the design is built and will operate in accordance with the design certification. The two-tiered design descriptions will constitute the portion of the design that will be incorporated into the design certification rule. The certified design description will control proposed changes to that portion of the facility throughout its lifetime in accordance with Section 52.63. The ITAAC will only be used for the fuel load decision and to aid in determining if subsequent modifications to the facility have caused a change in the certified design.

The staff has reviewed the ITAAC submitted to date in order to determine whether or not all elements of the design description will have corresponding ITAAC elements. The nine pilot ITAAC all correspond to Tier 1 design descriptions, although not all parts of the Tier 1 design descriptions are covered by an ITAAC. For example, the SLCS design description identifies boron as the poison to shut down the reactor. There is no explicit ITAAC to verify the use of boron. However, the staff is working with GE to ensure that its use is verified in the analysis to meet the 850 ppm poison concentration stated in the acceptance criteria for the first certified design commitment in the SLCS ITAAC. The staff is continuing its evaluation of whether or not all elements of the design description will have corresponding ITAAC elements.

Also, GE has proposed that certain systems could have Tier 1 design descriptions, but may not require any corresponding ITAAC to verify the design for those systems. Examples of these systems are the fuel service equipment, the internal pump maintenance facility, and the fuel cask cleaning facility. The staff will review this proposal as part of the review of the phase 3 ITAAC.

The staff has incorporated algorithms into the ITAAC and specified appropriate numerical bounding values on the acceptance criteria, where appropriate. This is illustrated in Enclosure 1, in the first certified design commitment for the SLCS system, where the staff specified the boron requirements to shut down the reactor.

REGULATORY REQUIREMENTS NOT IN ITAAC

Some regulatory requirements must be met after fuel loading, but prior to operation. These requirements will not be covered by ITAAC because the results of ITAAC must be determined prior to fuel loading. Examples of these requirements are those associated with the Initial Test Program for start-up testing, low power testing, and power ascension testing. The test programs involve verification of fuel, control rod, and core characteristics, as well as system and integrated plant operating characteristics.

TREATMENT OF REGULATORY REQUIREMENTS NOT IN ITAAC

Section 52.103(c) does not address requirements such as those identified in the preceding paragraph that are not covered in ITAAC. However, treatment of these issues will be similar to their treatment at facilities licensed under Part 50, where the test programs occurred after the issuance of an operating license (OL). Under Part 50, these test programs were reviewed and approved as part of the application for an OL. Verification of the satisfactory completion of these requirements was a condition of the license. Any changes to the requirements in the OL required a license amendment.

Under Part 52, the process will operate in a corresponding manner, by incorporation of a requirement for successful completion of the test programs into the COL as a condition of the license. Section 52.97(b) states that "any modification to, addition to, or deletion from the terms of a combined license,...is a proposed amendment to such license." Thus, in the same manner as for facilities licensed under

Part 50, the public will be afforded the opportunity to participate in any proposed changes to the conditions in a combined license.

ITAAC REQUIREMENTS FOR FDA

In SECY-91-210, the staff discussed the ITAAC review and approval requirements for a certification FDA. The staff and GE are continuing to define the information required in the certified design information, and this is resulting in changes to the standard safety analysis report (SSAR). For example, the SSAR may have referenced a single value for a parameter in an analysis. However, the acceptance criteria in the ITAAC may require a range of values for that parameter in order to provide necessary flexibility in the design certification. This will require SSAR amendments in order to provide analyses to support the range of acceptance criteria in the ITAAC.

Also, the development of the DAC areas have required changes to both the certified design descriptions and the SSAR. In addition, rulemaking for design certification, which follows the certification FDA issuance, cannot be initiated until the proposed design description is completed. Accordingly, the staff's experience, to date, in carrying out both the review of the SSAR and the ITAAC has reinforced the conclusions stated in SECY-91-210, that ITAAC should be reviewed and approved before issuance of a certification FDA. The design and ITAAC evaluations are fundamentally linked and should be completed prior to FDA issuance. This conclusion is further supported by the Commission in the SRM dated April 21, 1992, related to SECY-92-037, "Need for NRC-Sponsored Confirmatory Interal System Testing of the Westinghouse AP600 Design," which indicated that following FDA issuance, the staff will be bound by the safety decisions that are rendered in the FDA.

Coordination: The Office of the General Counsel has reviewed this paper and has no legal objection to its contents.

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Enclosures:
1. SLCS System Design Description and ITAAC
2. Environmental Equipment Generic ITAAC
3. Safety Analysis Verification using ITAAC

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2.2.4 Standby Liquid Control System

The standby liquid control system (SLCS) is designed to inject neutron absorbing poison using a boron solution into the reactor and thus provide back-up reactor shutdown capability independent of the normal reactivity control system based on insertion of control rods into the core. The system is capable of operation over a wide range of reactor pressure conditions up to and including the elevated pressures associated with an anticipated plant transient coupled with a failure to scram (ATWS).

The standby liquid control system (SLCS) is designed to provide the capability of bringing the reactor, at any time in a cycle, from full power and at all conditions to a subcritical condition with the reactor in the most reactive xenon-free state without control rod movement. The system will complete the injection of the boron solution in 50 to 150 minutes.

The SLCS consists of a boron solution storage tank, two positive displacement pumps, two motor operated injection valves which are provided in parallel for redundancy and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged through the 'B' high pressure core flooder (HPCF) subsystem sparger. Figure 2.2.4 shows major system components. Key equipment performance requirements are:

a.	Pump flow (minimum)	100 gpm with both pumps running
b.	Maximum reactor pressure (for injection)	1250 psig
c.	Pumpable volume in storage tank (minimum)	6100 U.S. gal

The required volume of solution contained in the storage tank is dependent upon the solution concentration and this concentration can vary during reactor operations. A required boron solution volume/concentration relationship is used to define acceptable SLCS storage tank conditions during plant operation.

The SLCS is automatically initiated during an ATWS. An ATWS condition exists when either of the following occurs:

a. High RPV pressure (1125 psig) and Average Power Range Monitor (APRM) not down scale for 3 minutes, or

b. Low RPV level (Level 2) and APRM not down scale for 3 minutes.

When the SLCS is automatically initiated to inject a liquid neutron absorber into the reactor, the following devices are actuated:

a. the two injection valves are opened;

b. the two storage tank discharge valves are opened;

c. the two injection pumps are started; and

d. the reactor water cleanup isolation valves are closed.

The SLCS can also be manually initiated from the main control room. When the SLCS is manually initiated to inject a liquid neutron absorber into the reactor, the following devices are actuated by each switch:

a. one of the two injection valves is opened;

b. one of the two storage tank discharge valves is opened;

c. one of the two injection pumps is started; and

d. one of the reactor water cleanup isolation valves is closed.

The SLCS provides borated water to the reactor core to compensate for the various reactivity effects during the required conditions. These effects include xenon decay, elimination of steam voids, changing water density due to the reduction in water temperature, Doppler effect in uranium, changes in neutron leakage and changes in control rod worth as boron affects neutron migration length. To meet this objective, it is necessary to inject a quantity of boron which produces a minimum concentration of 850 ppm of natural boron in the reactor core at 70°F. To allow for potential leakage and imperfect mixing in the reactor system, an additional 25% (220) is added to the above requirement. The required concentration is achieved accounting for dilution in the RPV with normal water level and including the volume in the residual heat removal shutdown cooling piping. This quantity of boron solution is the amount which is above the pump suction shutoff level in the tank thus allowing for the portion of the tank volume which cannot be injected.

The pumps are capable of producing discharge pressure to inject the solution into the reactor when the reactor is at high pressure conditions corresponding to the system relief valve actuation (1560 psig) which is above peak ATWS pressure.

The SLCS includes sufficient Control Room indication to allow for the necessary monitoring and control during design basis operational conditions. This includes pump discharge pressure, storage tank liquid level and temperature as well as valve open/close and pump on/off indication for those components shown on Figure 2.2.4 (with the exception of the simple check valves).

The SLCS uses a dissolved solution of sodium pentaborate as the neutron-absorbing poison. This solution is held in a storage tank which has a heater to maintain solution temperature above the saturation temperature. The heater is capable of automatic operation and automatic shutoff to maintain an acceptable solution temperature. The SLCS solution tank, a test water tank, the two positive displacement pumps, and associated valving is located in the secondary containment on the floor elevation below the operating floor. This is a Seismic Category I structure, and the SLCS equipment is protected from phenomena such as earthquakes, tornados, hurricanes and floods as well as from internal postulated accident phenomena. In this area, the SLCS is not subject to conditions such as missiles, pipe whip, and discharging fluids.

The pumps, heater, valves and controls are powered from the standby power supply or normal offsite power. The pumps and valves are powered and controlled from separate buses and circuits so that single active failure will not prevent system operation. The power supplied to one motor operated injection valve, storage tank discharge valve, and injection pump is powered from Division I, 480 VAC. The power supply to the other motor-operated injection valve, storage tank outlet valve, and injection pump is powered from Division II, 480 VAC. The power supply to the tank heaters and heater controls is connectable to a standby power source. The standby power source is Class 1E from an on-site source and is independent of the off-site power.

Components of the system which are required for injection of the neutron absorber into the reactor are classified Seismic Category I. The major mechanical components are designed to meet ASME Code requirements as shown below.

Component	ASME Code Class	Design Conditions Pressure	Temperature
Storage Tank	2	Static Head	150°F
Pump	2	1560 psig	150°F
Injection Valves	1	1560 psig	150°F
Piping Inboard of Injection Valves	1	1250 psig	575°F

Piping and components not required for the injection of the neutron absorber (e.g. test tank, sampling system line and storage tank vent) are classified NNS.

Design provisions to permit system testing include a test tank and associated piping and valves. The tank can be supplied with demineralized water which can be pumped in a closed loop through either pump or injected into the reactor.

The SLCS is separated both physically and electrically from the control rod drive system.

Inspection, Test, Analyses and Acceptance Criteria

Table 2.2.4 provides a definition of the inspections, tests, and/or analyses together with associated acceptance criteria which will be undertaken for the SLCS.

Table 2.2.4: STANDBY LIQUID CONTROL SYSTEM

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The minimum average poison concentration in the reactor after operation of the SLCS shall be equal to or greater than 850 ppm.	1. Construction records, revisions and plant visual examinations will be undertaken to assess as-built parameters listed below for compatibility with SLCS design calculations. If necessary, an as-built SLCS analysis will be conducted to demonstrate the acceptance criteria is met. **Critical Parameters:** a. Storage tank pumpable volume b. RPV water inventory at 70°F c. RHR shutdown cooling system water inventory at 70°F	1. It must be shown the SLCS can achieve a poison concentration of 850 ppm or greater assuming a 25% dilution due to non-uniform mixing in the reactor and accounting for dilution in the RHR shutdown cooling systems. This concentration must be achieved under system design basis conditions. This requires that SLCS meet the following values: Storage tank pumpable volume range 6100-6800 gal. RPV water inventory $\leq 1.00 \times 10^6$lb RHR shutdown cooling system inventory $\leq .287 \times 10^6$ lb
2. A simplified system configuration in shown in Figure 2.2.4.	2. Inspections of installation records together with plant walkdowns will be conducted to confirm that the installed equipment is in compliance with the design configuration defined in Figure 2.2.4.	2. The system configuration is in accordance with Figure 2.2.4.

Table 2.2.4: STANDBY LIQUID CONTROL SYSTEM (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. SLCS shall be capable of delivering 100 gpm of solution with both pumps operating against the elevated pressure conditions which can exist in the reactor during events involving SLCS initiation.	3. System preoperation tests will be conducted to demonstrate acceptable pump and system performance. These tests will involve establishing test conditions that simulate conditions which will exist during an SLCS design basis event. To demonstrate adequate Net Positive Suction Head (NPSH), delivery of rated flow will be confirmed by tests conducted at conditions of low level and maximum temperature in the storage tank, and the water will be injected from the storage tank to the RPV.	3. It must be shown that the SLCS can automatically inject 100 gpm (both pumps running) against a reactor pressure of 1250 psig with simulated ATWS conditions. It must also be shown that the SLCS pumps can pump the entire storage tank pumpable volume.
4. The system is designed to permit in-service functional testing of SLCS.	4. Field tests will be conducted after system installation to confirm, in-service system testing can be performed.	4. Using normally installed controls, power supplies and other auxiliaries, the system has the capability to: a. Pump tests in a closed loop on the test tank and b. Reactor pressure vessel injection tests using demineralized water from the test tank.
5. The pump, heater, valves and controls can be powered from the standby AC power supply as described in Section 2.2.4.	5. System tests will be conducted after installation to confirm that the electrical power supply configurations are in compliance with design commitments.	5. The installed equipment can be powered from the standby AC power supply.
6. SLCS components which are required for the injection of the neutron absorber into the reactor are classified Seismic Category I and qualified for appropriate environment for locations where installed.	6. See Generic Equipment Qualification verification activities (ITA).	6. See Generic Equipment Qualification Acceptance Criteria (AC).

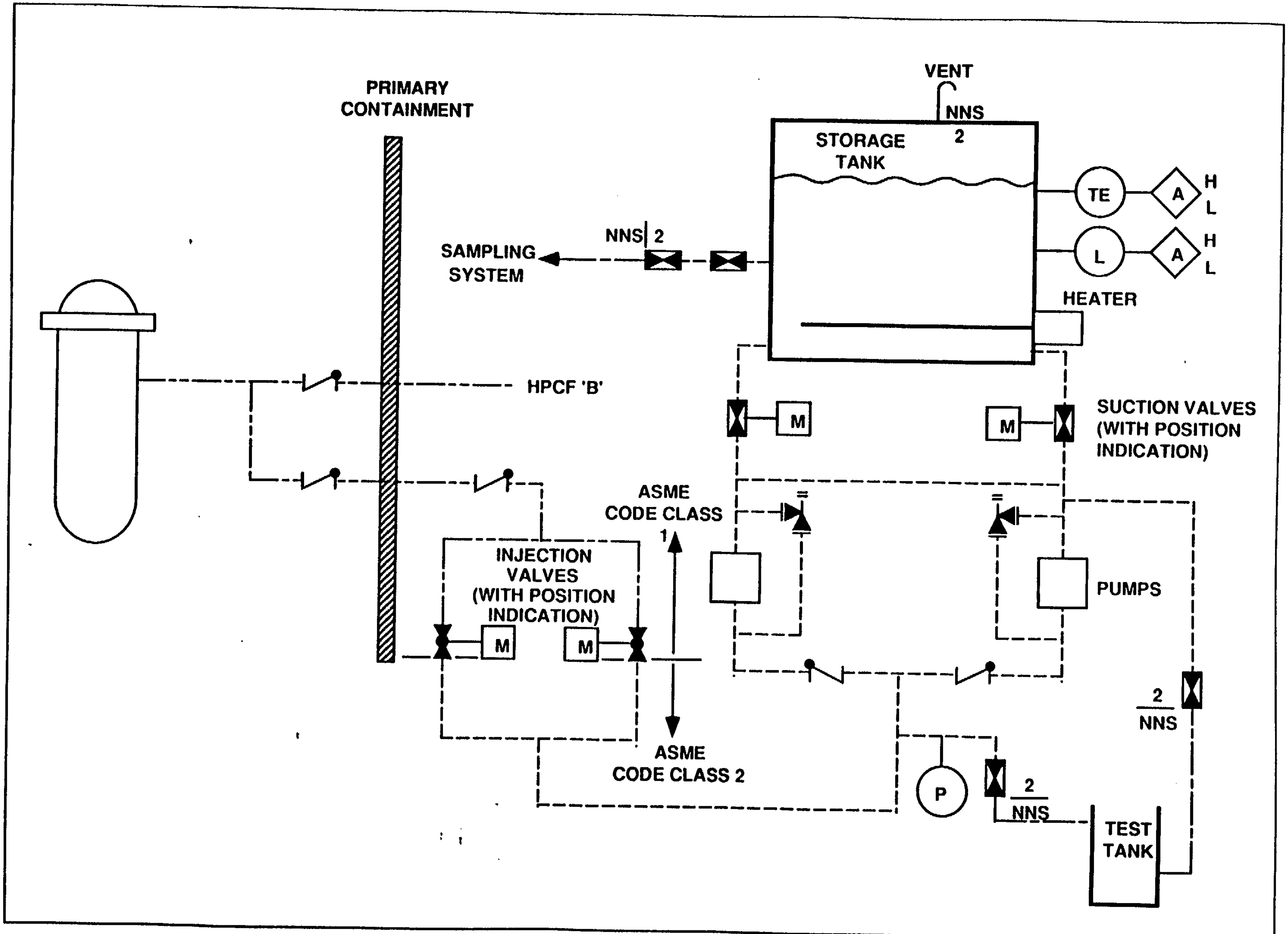

Figure 2.2.4 STANDBY LIQUID CONTROL SYSTEM (STANDBY MODE)

3.1 Equipment Qualification (EQ)

Design Description

Mechanical and electrical equipment that is important to safety is qualified for the full range of environmental conditions that will exist up to and including the time the equipment has finished performing its safety-related function. Equipment used for the certified design will be in full compliance with the regulatory requirements and industrial standards governing qualification methodology to be used for safety equipment in nuclear power plants.

The scope of this generic material is to address the complete spectrum of environmental conditions that may occur in the facility. Not all safety equipment will experience all of these conditions; the intent is that qualification be performed by selecting the conditions applicable to each particular piece of equipment and performing the necessary qualification using acceptable methods.

Inspection, Test, Analyses and Acceptance Criteria

Table 3.1 provides a definition of the inspections, tests, and/or analyses (together with associated acceptance criteria) which will be performed to demonstrate compliance with the equipment qualification commitments for the certified design.

Table 3.1: Equipment Qualification

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Mechanical and electrical equipment important to safety will be qualified for the environmental conditions that exist up to and including the time the equipment has finished performing its safety- related function. Conditions that exist during normal, abnormal and design basis accident events will be considered in terms of their cumulative effect on equipment performance. These conditions will be considered for the time period up to the end of components refurbishment interval or end of equipment life. These conditions include number and/or duration of equipment functional and test cycles/ events; process fluid conditions (where applicable); the voltage, frequency, load, and other electrical characteristics of the equipment; the dynamic loads associated with seismic events, containment response to hydrodynamic conditions, system transients, and other vibration inducing events, and the pressure, temperature, humidity, chemical and radiation environments, aging and submergence (if any) that can affect or degrade equipment performance. Other environmental conditions that will be considered are those included within environmental compatibility (EMC). These conditions are electromagnetic interference (EMI), electrostatic discharge (ESD), radio-frequency interference (RFI) and surge withstand capability (SWC).	1. Documentation relating to EQ issues will be completed for all equipment items important to safety and reviewed on a selected basis for compliance with requirements. This documentation will be in the form of the equipment qualification list and the device specific qualification files. The review will include review of specified environmental conditions, qualification methods (e.g., analyses or testing), and documentation of qualification results.	1. It will be confirmed that a comprehensive list of equipment important to safety has been prepared. The following information for this equipment shall be provided in a qualification file and subject to audit: a. The performance specifications under conditions existing during and following design basis accidents. For electrical items, this will include the voltage, frequency, load and other electrical characteristics for which the performance specified above can be ensured. b. The environmental conditions, including temperature, pressure, humidity, radiation, electromagnetic compatibility, chemicals and submergence at the location where the equipment must perform as specified above. This will include environmental conditions defined in 10 CFR 50.49, for electrical items and shall include consideration of synergistic effects and margins for unquantified uncertainty. c. The testing method used to qualify the equipment. Each item of equipment important to safety must be qualified by one (or a combination) of the following methods: 1) Testing an identical item of equipment under identical conditions or under similar conditions with a

Table 3.1: Equipment Qualification (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		1.c (cont.) supporting analysis to show that the equipment to be qualified is acceptable 2). Testing a similar item of equipment with a supporting analysis to show that the equipment to be qualified is acceptable. 3) Experience with identical or similar equipment under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable. 4) Analysis in combination with partial type test data that supports the analytical assumptions and conclusions. e. The results of the qualification have been documented to permit verification that the item of equipment important to safety: 1) Is qualified for its application; and 2) Meets its specified performance requirements when it is subjected to the conditions predicted to be present when it must perform its safety function up to the end of its qualified life.

Table 3.1: Equipment Qualification (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
2. The installed condition of mechanical and electrical equipment important to safety will be compatible with conditions for which it was qualified.	2. An inspection will be performed of installed safety equipment to assess compatibility with the methods and assumptions used to qualify the equipment.	2. The installed configuration is bounded by the test configuration and conditions. No physical interferences exist with adjacent plant feature which have not been addressed by the qualification process.

Table B1b: Safety Analysis Verification Using ITAAC

SSAR Entry	Parameter	Value (1)	Verifying ITAAC	
6.2.1	**Containment Functional Design**			
6.2.1.1.4.1	**Vacuum Breakers**			
	Diameter (inches)	20	2.14.1	Primary Containment System
	Quantity	8	2.14.1	Primary Containment System
Table 6.2-2	**Drywell**			
	Volume (ft^3)	259,563	2.14.1	Primary Containment System
	Leak Rate, Drywell and Wetwell (%/Day)	0.5	2.14.1	Primary Containment System
	Wetwell			
	Volume (ft^3)	210,475	2.14.1	Primary Containment System
	Minimum Suppression Pool Water Volume (ft^3)	126,427	2.14.1	Primary Containment System
	Total Vent Area (ft^2)	125	2.14.1	Primary Containment System
	Vent Centerline Submergence (Low Water Level), (ft):			
	Top Row	11.48	2.14.1	Primary Containment System
	Middle Row	15.98	2.14.1	Primary Containment System
	Bottom Row	20.48	2.14.1	Primary Containment System
Table 6.2.2-a	**RHR System**			
	Pump Capacity (gpm/pump)	4200	2.4.1	Residual Heat Removal System
	Heat Transfer Area (ft^2/unit)		2.4.1	Residual Heat Removal System
	Heat Transfer Coefficient (Btu/sec-F)	195	2.4.1	Residual Heat Removal System
	Service Water Flow (lbm/hr)	2.63×10^6	2.4.1	Residual Heat Removal System
Table 6.2-2d	**Secondary Containment**			
	Free Volume (ft^3)	3.0×10^6	2.15.10	Reactor Building
	Pressure (inch H_2O)	-0.25	2.15.10	Reactor Building
	Leak Rate (%/day)	50	2.15.10	Reactor Building

Table B1b: Safety Analysis Verification Using ITAAC (Continued)

SSAR Entry	Parameter	Value (1)	Verifying ITAAC	
Table 6.3-1	**Low Pressure Flooder System**			
	Minimum Vessel Pressure to Initiate Flow (psid)	225	2.4.1	Residual Heat Removal System
	Minimum Rated Flow (gpm/unit) at Vessel Pressure (psid)	4200 40	2.4.1	Residual Heat Removal System
	Initiating Signals Low Water Level (ft above TAF)	<0.6	2.1.2	Nuclear Boiler System
	Maximum Time from Signal to Pumps at Rated Speed (sec)	29	2.4.1	Residual Heat Removal System
	Maximum Time from Low Pressure Permissive signal to Injeciton Valve Fully Open (sec)	36	2.4.1	Residual Heat Removal System
	RCIC System			
	Minimum Vessel Pressure to Initiate Flow (psid)	1177	2.4.4	Reactor Core Isolation Cooling
	Minimum Rated Flow (gpm/unit) at Vessel Pressure (psid)	800 1177-150	2.4.4	Reactor Core Isolation Cooling
	Initiating Signals Low Water Level (ft above TAF)	<8.1	2.1.2	Nuclear Boiler System
	Maximum Time from Signal to Injection Valve Fully Open (sec)	29	2.4.4	Reactor Core Isolation Cooling
	HPCF System			
	Minimum Vessel Pressure to Initiate Flow (psid)	1177	2.4.2	High Pressure Core Flooder System
	Minimum Rated Flow (gpm/unit) at Vessel Pressure (psid)	800-3200 1177-100	2.4.2	High Pressure Core Flooder System
	Initiating Signals Low Water Level (ft above TAF)	<3.4	2.1.2	Nuclear Boiler System
	Maximum Time from Signal to Injection Valve Full Open (sec)	36	2.4.2	High Pressure Core Flooder System
	ADS			
	Minimum Flow Capacity (lbs/hr) at Vessel Pressure (psig)	6.4×10^6 1125	2.1.2	Nuclear Boiler System
	Initiating Signals Low Water Level (ft above TAF)	<0.6	2.1.2	Nuclear Boiler System
	Maximum Time from Signal to Valves Fully Open (sec)	<29	2.1.2	Nuclear Boiler System

Table B1b: Safety Analysis Verification Using ITAAC (Continued)

SSAR Entry	Parameter	Value (1)	Verifying ITAAC	
Table 6.3-4	**LOCA Break Sizing**			
	Steamline (ft^2)	1.06	2.1.1	Reactor Pressure Vessel System
	Feedwater Line (ft^2)	0.903	2.1.1	Reactor Pressure Vessel System
	RHR Shutdown Cooling Suction Line (ft^2)	0.852	2.1.1	Reactor Pressure Vessel System
	RHR Injection Line (ft^2)	0.221	2.1.1	Reactor Pressure Vessel System
	High Pressure Core Flooder (ft^2)	0.099	2.1.1	Reactor Pressure Vessel System
	Bottom Head Drain Line (ft^2)	0.0218	2.1.1	Reactor Pressure Vessel System
Table 6.3-9	**Design Parameters for RHR System Components**			
	Pump Flow Rate (gpm)	4200	2.4.1	Residual Heat Removal
Table 15.0.1	**Input Parameters and Initial Conditions for System Response Analysis Transient**			
	Safety/Relief Valve Capacity at 80.5 kg/cm^2 (%NBR)	91.13	2.1.2	Nuclear Boiler System
	Recirculation Pump Trip Inertia Time Constant (sec)	0.62	2.1.3	Reactor Recirculation System
	Steamline Volume (m^3)	113.2	2.1.2	Nuclear Boiler System
Table 15.0-6	**FMCRD Scram Time**			
	100% Rod Insertion (sec)	3.719	2.2.2	Control Rod Drive System
15.2	Increase in Rx Pressure			
15.2.2.3.1	TCV Full Stroke Closure (sec)	0.15	2.10.8	Turbine Control System
15.2.3.3.1	Turbine Stop Valve Full Stroke Closure (sec)	0.10	2.10.9	Turbine Control System
15.2.4.3.1	MSIV Closure (sec)	3-5	2.1.2	Nuclear Boiler System
15.2.5.3.1	Same as 15.2.3.3.1			
15.4	**Reactivity and Power Distribution Anomalies**			
15.4.1.2.3.2	FMCRD Withdrawal (mm/sec)	30	2.2.2	Control Rod Drive System