7.5 Information Systems Important to Safety

7.5.1 Systems Descriptions

Safety-related display systems are those systems which provide information for the safe operation of the plant during normal operation, anticipated operational occurrences, and accidents. The information systems important to safety include those systems which provide information for manual initiation and control of safety systems, to indicate that plant safety functions are being accomplished and to provide information from which appropriate actions can be taken to mitigate the consequences of anticipated operational occurrences and accidents. The Safety Parameter Display System (SPDS), information systems associated with the emergency response facilities and nuclear data link are information systems important to safety.

7.5.1.1 Post Accident Monitoring System

(1) Variable Types

Regulatory Guide 1.97 defines five "types" and three "categories" of plant variables for accident monitoring instrumentation. A discussion of these classifications is provided below. Each variable has been defined as to both type and category. Plant variables are divided into types according to the purpose of the indication to the plant operator. Any one variable may belong to more than one type.

(a) Type A

Type A are those variables to be monitored that provide the primary information required to permit the control room operators to take the specified manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design basis accident events.

Primary information is information that is essential for the direct accomplishment of the specified safety function. It does not include those variables that are associated with contingency (or backup) action that may also be identified in written procedures or guidelines.

Type A variables are limited to those variables which are necessary (primary) to alert the control room operator of the need to perform preplanned manual actions for safety systems to perform their safety functions, such as, initiating containment spray to permit the systems to perform safety functions for which no automatic system controls are provided. Variables that require actions specified by the Emergency Procedure Guidelines (EPGs) in response to specific operating limits have also been considered in performing the assessment documented in this chapter.

Type A variables do not include variables (1) which may indicate whether a specific safety function is being accomplished (Type B), or (2) which may indicate the need for contingency or corrective actions, resulting from the failure of the plant (Type C) or system(s) (Type D) to respond correctly when needed, or (3) which may indicate to the operator that it is desirable to change or modify the operation/alignment of systems important to safety to maintain the plant in a safe condition after plant safety has been achieved. Subsection 7.5.2.1(1) discusses the selection of specific Type A variables for the ABWR.

(b) Type B

Type B are those variables that provide information to the control room operators to indicate whether plant safety functions are being accomplished, including reactivity control, core cooling, maintaining reactor coolant system integrity, and maintaining containment integrity.

(c) Type C

Type C are those variables that provide information to the control room operators to indicate that barriers to fission product release have the potential for being breached or have been breached. These barriers are the fuel cladding, primary coolant pressure boundary, and primary containment.

The sources of potential breach are limited to the energy sources within the cladding, coolant boundary, or containment.

(d) Type D

Type D are those variables that provide information to the control room operators to indicate the successful operation of individual safety systems or other systems important to safety.

Type D variables should provide information to permit the control room operators to ascertain the operating status of each individual safety system and other systems important to safety to that extent necessary to determine if each system is operating or can be placed in operation to help mitigate the consequences of an accident.

(e) Type E

Type E are those variables monitored to determine the magnitude of release of radioactive materials and to assess the continuation of such releases.These variables should permit the control room operators to monitor the effluent discharge paths and environs within the site boundary to ascertain if there have been significant releases (planned or unplanned) of radioactive materials and to continually assess such releases.

In particular, Type E variables monitor:

(i) The planned paths for effluent release

(ii) Plant areas inside buildings where access is required to service equipment necessary to mitigate the consequences of an accident

(iii) Onsite location where unplanned releases of radioactive materials are detected

(2) Categories of Variables

The design and qualification criteria for the instrumentation used to measure the various variables are divided into three categories that provide a graded approach to instrumentation criteria, depending on importance to safety of the variables.

In general, Category 1 provides for full qualification, redundancy, and continuous real-time display and requires onsite (standby) power. Category 2 provides for qualification but is less stringent in that it does not (of itself) include seismic qualification, redundancy, or continuous display and requires only a high-reliability power source (not necessarily standby power). Category 3 is the least stringent. It provides for high-quality commercial-grade equipment that requires only offsite power.

(a) Category 1 represents the most stringent criteria and is used for key variables. Key variables are those parameters that most directly indicate the accomplishment of a safety function. All Type A variables are considered to be Category 1. For Types B and C, the key variables are Category 1, while backup variables are generally Category 3.

(b) Category 2 provides less stringent criteria and generally applies to instrumentation designated for indication of system operating status. Most Type D variables are classified as Category 2.

(c) Category 3 provides criteria for high quality backup and diagnostic instrumentation or for other instrumentation where the state-of-the-art will not support requirements for higher qualified instrumentation.

(3) Design and Qualification Criteria

The detailed Design and Qualification Criteria for Category 1, 2 and 3 variables are provided in Reg. Guide 1.97 for:

(a) Equipment Qualification

(b) Redundancy

(c) Power Sources

(d) Channel Availability

(e) Quality Assurance

(f) Display and Recording

(g) Range

(h) Equipment Identification

(i) Interfaces

(j) Servicing, Testing, and Calibration

(k) Human Factors

(l) Direct Measurement

A detailed listing of the design and qualification criteria for Categories 1, 2 and 3 is provided in Table 7.5-1.

In addition to design and qualification criteria, Regulatory Guide 1.97 provides a comprehensive listing of "BWR variables" which address accident monitoring requirements. Table 7.5-2 was developed using Table 2 of Regulatory Guide 1.97 as a guide. Design and qualification criteria are addressed as category designations per the discussion above. Variables listed in Table 7.5-2 without comment meet the design and qualification requirements of Regulatory Guide 1.97. Any exceptions taken are noted in the comment column.

7.5.2 Systems Analysis

7.5.2.1 Post Accident Monitoring System

(1) Type A Variables

Type A variables are fundamentally plant parameters needed to alert the control room operators to take safety actions by manually initiating a system or function which otherwise would not be automatically initiated in the course of an event. The Regulatory Guide 1.97 does not specify Type A variables; rather, it requires that each plant develop its own list of Type A variables from a review of each plant design.

For this assessment, the list of Type A variables was identified from a review of accidents described in Chapter 15 and a review of the Emergency Procedure Guidelines (EPGs). The event descriptions of Chapter 15 and the Plant Nuclear

Safety Operational Analysis (NSOA)of Appendix 15A were reviewed to determine the ABWR plant systems which would require manual initiation and the key variables associated with manual initiation of those systems. The Emergency Procedure Guidelines (EPGs) included in Chapter 18, Human Factors Evaluation, Appendix 18A, were also reviewed to identify any other variables requiring safety action. A summary of the Type A variables identified through this process are shown in Table 7.5-3. Details of the Type A variable assessment are provided in the following portion of this section.

(a) Type A Variable Evaluation and Analysis

Chapter 15 contains discussions of numerous events, not all of which are design basis accidents. Appendix 15A is a plant Nuclear Safety Operational Analysis (NSOA) which addresses these events in the following categories:

(i)	Normal operations	
(ii)	Moderate Frequency Accidents (Anticipated Operational Transients)	(Table 7.5-4)
(iii)	Infrequent Accidents (Abnormal Operational Transients)	(Table 7.5-5)
(iv)	Limiting Faults (Design Basis Accidents)	(Table 7.5-6)
(v)	Special Events	(Table 7.5-7)

Variables associated with normal operations are excluded from further investigation because those activities are planned actions which would not normally be expected to cause a threat to the general public.

Because probabilistic risk assessments show that the risk to the general public is dominated by transients rather than design basis accidents, all of the above categories (except normal operations) were considered to determine what parameters required operator action. Tables 7.5-4 through 7.5-7 list the events considered and the primary variables associated with called-for manual action. The manual action variables are taken from either the NSOA or the Chapter 15 event descriptions. The required manual actions are summarized in Table 7.5-8 along with the associated variables.

The EPGs were also reviewed to determine if there are other variables not specifically identified by Chapter 15 which are associated with required operator actions. Table 7.5-8 includes these additional variables and actions which result from a review of the following guidelines included in Appendix 18A:

(i) RPV Control

(ii) Primary Containment Control

(iii) Radioactivity Release Control

(iv) Secondary Containment Control

Some of these variables, especially those related to emergency action, are considered beyond the scope of the regulatory guide by virtue of requiring “contingency actions that are identified in written procedures.”

The final list of Type A variables was derived from the variables indicated on Table 7.5-8 and is summarized on Table 7.5-3.

For the ABWR, actions to isolate systems will be accomplished automatically by the LDS on high area temperatures (T_{2C}). Thus, this parameter is excluded from the Type A variable list. Other secondary containment area parameters (R_{2C} L_{2C}) were deleted because they represent early actions which could be taken to reduce the amount of plant effluent release beyond those values used as a basis for the plant safety analysis. Thus, these parameters were considered to be contingency actions and not required to be Type A.

The offsite release rate (R_E) was also not included with the Type A Variable List because the emergency action (emergency depressurization) specified in the radioactivity release control guidelines would, in all events, have been previously initiated in response to other variables (e.g., RPV Water Level). This conclusion is reached because the source terms required to reach release rates associated with a general emergency (the point at which the emergency action is required by the EPG) can only occur following a release of a substantial proportion of the fuel noble gas inventory. Prevention of such a release is a primary goal of the RPV control guideline. Also, the other operator action (isolate lines discharging outside the primary and secondary containment) is intended to be taken at levels low enough as to not pose a significant risk for the general public. The primary lines which communicate with the RPV are automatically isolated which satisfies the intent of the EPG action for these lines. Other lines which pass outside of the primary and secondary containment but which do not communicate directly with the RPV also receive automatic isolation signals. Thus, response to the radioactivity release control guideline is considered to be a contingency action and is not required to be Type A.

(2) General Variable Assessments

This section summarizes the results of the individual variable assessments concentrating on deviations identified between the existing design of the ABWR and the implementation position for Regulatory Guide 1.97 regarding the need for unambiguous indication.

Strict compliance with the regulatory guide is not provided in all cases. In some cases, an acceptable alternate has been proposed which meets the intent to have meaningful post-accident indications. For some parameters, this can be met by alternate variables to those specified in the Regulatory Guide 1.97 or by specifying combinations of other variables. Another approach chosen is to take exception to the guide where a reasonable justification can be provided.

(a) Drywell Pressure

Requirements for monitoring of drywell pressure are specified for both narrow range (from about -34.32 kPaG to + 34.32 kPaG) and wide range (from 0 to 110% of design pressure). The narrow range monitoring requirement is satisfied in the existing safety-related design by the four divisions of drywell pressure instruments which provide inputs to the initiation of the reactor protection (trip) system (RPS) and the emergency core cooling systems (ECCS). The requirement for unambiguous wide range drywell pressure monitoring are satisfied with two channels of drywell pressure instrumentation integrated with two channels of wetwell pressure instrumentation. Given the existence of (1) the normal pressure suppression vent path between the drywell and wetwell and (2) the wetwell to drywell vacuum breakers, the long-term pressure within the drywell and wetwell will be approximately the same. Therefore, if the two wide range drywell pressure indications disagreed, the operator could refer to the wetwell containment pressure indications to determine which of the two drywell pressure indications is correct. In order to provide full range pressure comparisons between the drywell wide range and wetwell pressure instruments, the drywell pressure instrument range is 689.4 kPa. This value exceeds the required value of 110% of design pressure. Drywell pressure is a Type A variable because it is used to initiate drywell spray to maintain the Reinforced Concrete Containment Vessel (RCCV) below temperature limits in LOCA.

(b) Containment Pressure (Wetwell Pressure)

Requirements for monitoring of wetwell containment pressure specify the monitored range to be –34.32 kPaG to three times the design pressure for concrete containments. For the ABWR, 3 times the design pressure is about 931.6 kPaG. The ABWR primary containment has diaphragm safety devices which release wetwell atmosphere at about 617.8 kPaG. Therefore, it is not credible for containment pressure to achieve this value. For this reason and for better resolution of the measurements, the top of the instrument range for containment pressure is 689.4 kPaG. Two channels of instrumentation covering this full pressure range provide adequate post accident monitoring (PAM) indication of primary containment pressure since any disagreement between the output of the two channels could be resolved by the operator's reference to the drywell pressure indicators as discussed above. Since wetwell pressure is the parameter used by the control room operator to manually initiate wetwell spray, wetwell pressure is considered a Type A variable.

(c) Coolant Level in the Reactor

The RPV water level is the primary variable indicating the availability of adequate core cooling. Indication of water level by the differential pressure method is considered acceptable, (without diverse methods of sensing and indication), provided adequate redundancy for qualification of unambiguity is provided over the entire range of interest which extends from the bottom of the core support plate to the center line of the main steamlines.

In the ABWR design, the RPV water level wide range instruments and fuel zone instruments are utilized to provide this Post Accident Monitoring (PAM) indication. The four divisions of wide range instruments cover the range from above the core to the main steamlines. The two channels of fuel zone instruments cover the range from below the core to the top of the steam separator shroud.

Evaluation has concluded that two channels of fuel zone level instrumentation provide adequate post accident monitoring capability. Post-accident operator actions will be in accordance with detailed procedures developed based upon the Emergency Procedure Guidelines (EPG). In the event the vessel water level is below the range of the wide range level (WRL) sensors (i.e., the water level is in the fuel zone range) and the two channels of fuel zone level instrumentation disagree, the EOPs instruct the operator to use the lower of the two and return the water level back up into the range of the WRL instrumentation. Using the four divisions of WRL instruments, an unambiguous indication of vessel water level can be determined, despite a postulated failure of a single instrument channel or division, and the operator could safely continue the execution of appropriate accident instigation activities as defined by the EOPS.

(d) BWR Core Temperature

Regulatory Guide 1.97 requires BWR core temperature (thermocouples) as a diverse indication of adequate core cooling. The BWR Owners' Group has taken exception to this requirement for diverse indication based upon studies regarding the relationship between reactor water level and adequate core cooling. No instrumentation other than RPV water level indication is required to assure indication of adequate core cooling.

(e) Drywell Sump Level

An exception is made to Regulatory Guide 1.97 as written for the design category for the equipment drain sump level. Rather than Category 1, the Category 3 design requirements are more appropriate for the following reason: Indication of drywell floor drain sump level provides monitoring of leakage to the drywell and will be an early indication of a very small reactor coolant system leak/break for those events for which the drywell cooling system remains operable. However, it is primarily a backup variable to other indications of reactor coolant system leaks/breaks such as drywell pressure or drywell radiation level. In addition, containment water level is provided as a Type D, Category 2 variable. A lower design classification for drywell sump level is therefore appropriate and triplicated instrument channels are not necessary.

(f) Containment Area Radiation

The Containment Atmospheric Monitoring System (CAMS) consists of two independent and redundant radiation monitoring channels which provide indication of wetwell and drywell radiation levels. Emergency response actions regarding this variable are consistently directed toward minimizing the magnitude of this parameter. This two channel CAMS design provides adequate PAM indication since, in the event that the two channels of information disagree, the operator can determine a correct and safe action based upon the higher of the two (in-range) indicators.

(g) Primary Containment Isolation Valve Position

The primary containment isolation valve position information provides indication to the operator regarding the successful completion of the primary containment isolation safety function. Following the requirements of 10CFR50 Appendix A, General Design Criteria 54, 55, 56 and 57, lines which penetrate the primary reactor containment are provided with varying degrees of redundant manual, check and automatically initiated isolation valves. Indication of the successful completion of the primary containment isolation safety function is provided by valve closed/not closed indicators for individual power operated valves. This arrangement, which provides redundant isolation valves and independent indication of valve position, is considered sufficient to satisfy the intent of Regulatory Guide 1.97 without requiring the use of triplicated instrument channels.

(h) Coolant Radiation

A continuous post-accident monitor for this parameter is not necessary and is not included in the design. This is consistent with BTP HICB-10, Table 1.

(i) Suppression Pool Water Temperature

The ABWR Suppression Pool Temperature Monitoring (SPTM) System design requirements satisfy the Regulatory Guide 1.97 requirements regarding redundancy. The SPTM System is composed of four separate and independent instrument divisions. Each division has associated with it multiple thermocouples which are spatially distributed around the suppression pool. With this configuration, the bulk average suppression pool temperature can be determined even in the event of the loss of an entire division of instrumentation, since thermocouple sensors of each division will be located in close proximity to facilitate direct comparison. Although the ABWR design initiates reactor scram and suppression pool cooling automatically on high pool temperature, suppression pool water temperature variable is considered a Type A variable since the operator uses it for manual RPV depressurization.

(j) Drywell Atmosphere Temperature

Surveillance monitoring of the temperatures in the drywell is provided by multiple temperature sensors distributed throughout the drywell to detect local area “hot-spots” and to monitor the operability of the drywell cooling system. With this drywell air temperature monitoring system supplied by multiple temperature sensors throughout the drywell, the Regulatory Guide 1.97 requirements for monitoring of drywell air temperature are met and provides the ability to determine drywell bulk average temperature.

(k) Drywell/Wetwell Hydrogen/Oxygen Concentration

The Containment Atmospheric Monitoring System (CAMS) consists of two independent and redundant nonsafety-related drywell/containment oxygen and hydrogen concentration monitoring channels. Emergency response actions regarding these variables are consistently directed toward minimizing the magnitude of these parameters (i.e., there are no safety actions which must be taken to increase the hydrogen/oxygen levels if they are low). Minimizing drywell/wetwell oxygen and hydrogen concentrations is accomplished by manual operator actions using containment venting and purging or using containment spray. Consequently, the two channel CAMS design provides adequate PAM indication, since, in the event that the two channels of information disagree, the operator can determine a correct and safe action based upon the higher of the two (in-range) indications.

(l) Wetwell Atmosphere Air Temperature

Surveillance monitoring of temperatures in the wetwell is provided by multiple temperature sensors dispersed throughout the wetwell, therefore, the required indication of bulk average wetwell atmosphere temperature is satisfied.

(m) Standby Liquid Control System Flow

No flow indication is provided for the ABWR design. The positive displacement SLCS pumps are designed for constant flow. Any flow blockage or line break would be indicated by abnormal system pressure (high or low as compared to RCS pressure) following SLCS initiation. Changing neutron flux, SLCS pressure and SLCS tank level are substituted for SLCS flow and are considered adequate to verify proper system function. One channel of SLCS discharge pressure is provided in addition to the monitoring of neutron flux.

(n) Suppression Pool/Wetwell Water Level

Regulatory Guide 1.97 suggests two ranges for suppression pool water level (i.e., bottom of ECCS suction to 1.5m above normal water level and top of vent to top of weir wall [BWR 6, Mark III Containment]). The ABWR provides:

(i) Four (4) divisions of narrow range suppression pool water (e.g., approximately 0.5 meters above and below normal water level) for control of normal water level and automatic transfer of RCIC and HPCF suctions.

(ii) Two (2) wide range suppression pool/wetwell water level instruments from approximately the centerline of the ECCS suction piping to the wetwell spray spargers. This range allows for control of suppression pool/wetwell water level in the vicinity of the spray spargers at the high end and the ECCS pumps (vortex limits) at the low end.

Two (2) wide range wetwell level instruments are sufficient to control water level at the high level and at the low level by using the highest reading and the lowest reading instruments, respectively, should the instruments disagree. In addition, The low end measurement to the centerline of the ECCS suction piping is considered sufficient since this level measurement is low enough to allow control of the pump vortex limits.

(Note: See drywell water level for instrument range overlap).

(o) Drywell Water Level

The lower drywell water level measurement below the RPV (other than sump level) is not warranted because of its inability to survive a severe accident (core melt) and because of the following: When the suppression pool level is increased to accommodate severe accident drywell flooding (per the ABWR EPGs), suppression pool level will stop increasing while the water spills into the lower drywell through the vents. Once drywell and wetwell water levels equalize, the increase in drywell level will be monitored by the wetwell water level monitors up to the bottom of the RPV. (See also upper drywell water level monitoring for instrument overlap.)

In addition to the above discussion of lower drywell water level monitoring, the ABWR design provides for two (2) upper drywell water level monitors. The range of these instruments in from approximately 0.5 meters below the RPV (lower drywell and above wetwell to lower drywell vents) to the maximum primary containment water level limit (MPCWLL) (upper drywell and approximately five (5) meters above TAF.). This lower range provides an approximately 0.5 meter instrument overlap with the wetwell water level instruments and therefore provides four (4) instruments for monitoring water immediately below the RPV during severe accident conditions.

Two (2) wide range upper drywell level measurements are sufficient, since there is sufficient margin between the TAF and MPCWLL to allow controlling water with the highest level measurement, should the instruments disagree, and still assure containment integrity and core coverage for containment flooding with no severe accident condition.

(p) Standby Liquid Control System Tank Level

As SLCS storage tank level is a backup variable to SLCS discharge pressure as described in the previous section (m), Category 3 qualification is appropriate instead of Category 2 suggested by Regulatory Guide 1.97.

(q) Drywell Spray Flow and Wetwell Spray Flow

The ABWR design does not provide direct Drywell Spray Flow indication. Regulatory Guide 1.97 suggests this as a Type D Variable for the purpose of monitoring drywell spray operation. As allowed by BTP HICB-10, RHR flow, drywell temperature and drywell pressure indications are provided as acceptable alternatives. RHR provides water to the drywell spray headers. Following a postulated accident, presence of drywell spray flow results in drywell pressure and temperature reduction. The operator confirms drywell spray operation by observing that there is RHR flow present and that the drywell pressure and temperature is within expected limits. Operator use of these variables allows accurate and reliable measurement of the effectiveness of the drywell spray in a timely manner. In addition, the position of the spray throttling valves can be monitored and the sprays adequately controlled from the control room using these alternative variables.

Table 7.5-1 Design and Qualification Criteria for Instrumentation

Category 1	Category 2	Category 3
1. Equipment Qualification		
The instrumentation is qualified in accordance with Regulatory Guide 1.89, “Qualification of Class 1E Equipment for Nuclear Power Plants”, and the methodology described in NUREG-0588, “Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment”.	Same as Category 1	No specific provision
(For equipment located in a mild environment, no specific environmental qualification is required except as required by General Design Criterion 4 of 10CFR50.)	(Same as Category 1)	(No specific provision)
Instrumentation whose ranges are required to extend beyond those ranges calculated in the most severe DBA event for a given variable are qualified using the guidance provided in Paragraph 63.6 of ANS-4.5.	Same as Category 1	No specific provision
Qualification applies to the complete instrumentation channel from sensor to display where the display is a direct-indicating meter or recording device. If the instrumentation channel signal is used in a computer-based display, recording, or diagnostic program, qualification applies from the sensor up to and including the channel isolation device.	Same as Category 1	No specific provision
The seismic portion of qualification is in accordance with Regulatory Guide 1.100, “Seismic Qualification of Electric Equipment for Nuclear Power Plants.” Instrumentation should continue to read within the required accuracy following, but not necessarily during, a safe shutdown earthquake.	No specific provision	No specific provision

Table 7.5-1 Design and Qualification Criteria for Instrumentation (Continued)

Category 1	Category 2	Category 3
2. Redundancy		
No single failure within either the accident-monitoring instrumentation, its auxiliary supporting features, or its power sources concurrent with the failures that are a condition or result of a specific accident should prevent the operators from being presented the information necessary for them to determine the safety status of the plant and to bring the plant to and maintain it in a safe condition following that accident. Where failure of one accident-monitoring channel results in information ambiguity (that is, the redundant displays disagree) that could lead operators to defeat or fail to accomplish a required safety function, additional information should be provided to allow the operators to deduce the actual conditions in the plant. This is accomplished by providing additional independent channels of information of the same variable (addition of an identical channel) or by providing an independent channel to monitor a different variable that bears a known relationship to the multiple channels (addition of a diverse channel). Redundant or diverse channels are electrically independent and physically separated from each other and from equipment not classified important to safety in accordance with Regulatory Guide 1.75, “Physical Independence of Electric Systems,” up to and including any isolation device. Within each redundant division of a safety system, redundant monitoring channels are not needed except for steam generator level instrumentation in two-loop plants.	No specific provision	No specific provision

Table 7.5-1 Design and Qualification Criteria for Instrumentation (Continued)

Category 1	Category 2	Category 3
3. Power Source		
The instrumentation is energized from station standby power sources as provided in Regulatory Guide 1.32, “Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants,” and is backed up by batteries where momentary interruption is not tolerable.	The instrumentation is energized from a high-reliability power source, not necessarily standby power, and backed up by batteries where momentary interruption is not tolerable.	No specific provision
4. Channel Availability		
The instrumentation channel is available prior to an accident noted in the Exception to Paragraph 6.7, "Maintenance Bypass," in IEEE-603, "Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” or as specified in the technical specifications.	The out-of-service interval is based on normal technical specification requirements on out-of-service for the system it serves where applicable or where specified by other requirements.	No specific provision
5. Quality Assurance		
The recommendations of the following regulatory guides pertaining to quality assurance are followed:	Same as Category 1 as modified by the following:	The instrumentation is of high-quality commercial grade and is selected to withstand the specific service environment.
Regulatory Guide 1.28 “Quality Assurance Program Requirements Design and Construction” Regulatory Guide 1.30 (Safety Guide 30) “Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment”	Since some instrumentation is less important to safety than other instrumentation, it is not necessary to apply the same quality assurance measures to all instrumentation. The quality assurance requirements that are implemented provide control over activities affecting quality to an extent consistent with the importance to safety of the instrumentation.	

Table 7.5-1 Design and Qualification Criteria for Instrumentation (Continued)

Category 1	Category 2	Category 3
Regulatory Guide 1.38 "Quality Assurance Requirements for Packaging, Shipping Receiving, Storage and Handling of Items for Water-Cooled Nuclear Power Plants"		
Regulatory Guide 1.58 "Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel"		
Regulatory Guide 1.64 "Quality Assurance Requirements for the Design of Nuclear Power Plants"		
Regulatory Guide 1.74 "Quality Assurance Terms and Definitions"		
Regulatory Guide 1.88 "Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records"		
Regulatory Guide 1.123 "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants"		
Regulatory Guide 1.144 "Auditing of Quality Assurance Programs for Nuclear Power Plants"		
Regulatory Guide 1.146 "Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants"		
6. Display and Recording		
Continuous real-time display is provided. The indication is on a dial, digital display, CRT, or strip-chart recorder.	The instrumentation signal is displayed on an individual instrument or it is processed for display on demand.	Same as Category 2
Recording of instrumentation readout information is provided for at least one redundant channel.	Signals from effluent radioactivity monitors and area monitors are recorded.	Signals from effluent radioactivity monitors, and meteorology monitors are recorded.

Table 7.5-1 Design and Qualification Criteria for Instrumentation (Continued)

Category 1	Category 2	Category 3
If direct and immediate trend or transient information is essential for operator information or action, the recording is continuously available on redundant dedicated recorders. Otherwise, it is continuously updated, stored in computer memory, and displayed on demand. Intermittent displays such as data loggers and scanning recorders are used if no significant transient response information is likely to be lost by such devices.	Same as Category 1	Same as Category 1
7. Range		
If two or more instruments are needed to cover a particular range, overlapping of instrument span is provided. If the required range of monitoring instrumentation results in a loss of instrumentation sensitivity in the normal operating range, separate instruments are used.	Same as Category 1	Same as Category 1
8. Equipment Identification		
[See also item 11]		
Types A, B, and C instruments designated as Categories 1 and 2 are specifically identified with a common designation on the control panels so that the operator can easily discern that they are intended for use under accident conditions.	Same as Category 1	No specific provision
9. Interfaces		
The transmission of signals for other use is through isolation devices that are designated as part of the monitoring instrumentation and that meet the provisions of this document.	Same as Category 1	No specific provision
10. Servicing, Testing, and Calibration		
Servicing, testing, and calibration programs are specified to maintain the capability of the monitoring instrumentation. If the required interval between testing is less than the normal time interval between plant shutdowns, a capability for testing during power operation is provided.	Same as Category 1	Same as Category 1
Whenever means for removing channels from service are included in the design, the design facilitates administrative control of the access to such removal means.	Same as Category 1	Same as Category 1

Table 7.5-1 Design and Qualification Criteria for Instrumentation (Continued)

Category 1	Category 2	Category 3
The design facilitates administrative control of the access to all setpoint adjustments, module calibration adjustments, and test points.	Same as Category 1	Same as Category 1
Periodic checking, testing, calibration and calibration verification are in accordance with the applicable portions of Regulatory Guide 1.118, “Periodic Testing of Electric Power and Protection Systems,” pertaining to testing of instrument channels. (Note: Response time testing not usually needed.)	Same as Category 1	Same as Category 1
The location of the isolation device is such that it would be accessible for maintenance during accident conditions.	Same as Category 1	No specific provision
11. Human Factors		
[See also item 8]		
The instrumentation is designed to facilitate the recognition, location, replacement, repair, or adjustment of malfunctioning components or modules.	Same as Category 1	Same as Category 1
The monitoring instrumentation design minimizes the development of conditions that would cause meters, annunciators, recorders, alarms, etc., to give anomalous indications potentially confusing to the operator. Human factors analysis is used in determining type and location of displays (see Chapter 18).	Same as Category 1	Same as Category 1
To the extent practicable, the same instruments are used for accident monitoring as are used for the normal operations of the plant to enable the operators to use, during accident situations, instruments with which they are most familiar.	Same as Category 1	Same as Category 1
12. Direct Measurement		

Table 7.5-1 Design and Qualification Criteria for Instrumentation (Continued)

Category 1	Category 2	Category 3
To the extent practicable, monitoring instrumentation inputs are from sensors that directly measure the desired variables. An indirect measurement is made only when it can be shown by analysis to provide unambiguous information.	Same as Category 1	Same as Category 1

Table 7.5-2 ABWR PAM Variable List

Variable	Range Required	Type	Cate-gory	Discussion Section
Neutron Flux	10^{-6}% to 100% full power	B	1	
Control Rod Position	Full in or not full in	B	3	
Boron Concentration	0–1000 ppm	B	3	
BWR Core Temperature	93.3°C to 1260°C			Subsection 7.5.2.1(2)(d)
Reactor Coolant System Pressure	0 to10.35 MPaG	B,C,D	1	
Drywell Pressure	-0.034 MPaG to +0.034 MPaG (narrow range) 0–110% design pressure (wide range)	A,B,C,D	1	Subsection 7.5.2.1(2)(a)
Drywell Sump Level	Top to Bottom	B,C	3	Subsection 7.5.2.1(2)(e)
Coolant Level in Reactor	Bottom of core plate to main steamline	B,C	1	Subsection 7.5.2.1(2)(c)
Suppression Pool Water Level	Bottom of ECCS suction line to 1.5 meters above normal water line	C	1	Subsection 7.5.2.1(2)(n)
	Top of vent to top of weir wall	D	2	Subsection 7.5.2.1(2)(n)
Drywell Water Level	(None specified)	D	2	Subsection 7.5.2.1(2)(o)
Containment Area Radiation	10^{-2} Sv/h to 10^{5} Sv/h	C,E	1	Subsection 7.5.2.1(2)(f)
Wetwell Pressure	– 0.034 MPaG to 3 times design pressure	A,B,C	1	Subsection 7.5.2.1(2)(b)
Primary Containment Isolation Valve Position	Closed – not closed	B	1	Subsection 7.5.2.1(2)(g)
Coolant Gamma	370 µBq to 370Bq/ml or TID-14844 Source Term in Coolant Volume	C	3	
RHR Flow	0–110% Design Flow	D	2	
HPCF Flow	0–110% Design Flow	D	2	
RHR Heat Exchanger Outlet Temperature	4.4°C to 176.7°C	D	2	
RCIC Flow	0–110% Design Flow	D	2	
Standby Liquid Control System Flow	0–110% Design Flow	D	2	Subsection 7.5.2.1(2)(m)

Table 7.5-2 ABWR PAM Variable List (Continued)

Variable	Range Required	Type	Cate-gory	Discussion Section
SLCS Storage Tank Level	Top to Bottom	D	3	Subsection 7.5.2.1(2)(o)
SRV Position	Closed – Not Closed	D	2	
Feedwater Flow	0–110% Design Flow	D	3	
High Radioactivity Liquid Tank Level	Top to Bottom	D	3	
Standby Energy Status	Plant Specific	D	2	
Suppression Pool Water Temperature	4.4°C to 140°C	A, D	1	Subsection 7.5.2.1(2)(i)
Drywell Atmosphere Temperature	4.4°C to 226.7°C	D	1	Subsection 7.5.2.1(2)(j)
Drywell/Wetwell Hydrogen Concentration	0–30 Volume%	C	3	Subsection 7.5.2.1(2)(k)
Drywell/Wetwell Oxygen Concentration	0–10 Volume%	C	2	Subsection 7.5.2.1(2)(k)
Wetwell Atmosphere Temperature	4.4°C to 226.7°C	D	1	Subsection 7.5.2.1(2)(l)
Secondary Containment Airspace (effluent) Radiation Noble Gas	37 pBq/cm^3 to 37MBq/cm^3	C	2	
Containment Effluent Radioactivity—Noble Gas	37 pBq/cm^3 to 0.37μBq/cm^3	C	3	
Condensate Storage Tank Level	Top to Bottom	D	3	
Cooling Water Temperature to ESF System Components	4.4°C to 93.3°C	D	2	
Cooling Water Flow to ESF System Components	0–110% Design Flow	D	2	
Emergency Ventilation Damper Position	Open – Closed Status	D	2	
Service Area Radiation Exposure Rate	10^{-3} Sv/h to 10^{2} Sv/h	E	3	
Purge Flows—Noble Gases and Vent Flow Rate	37 PBq/cm^3 to 0.37 Bq/cm^3 0–110% Vent Design Flow	E	2	
Identified Release Points—Particulates and Halogens	37 nBq/cm^3 to 3.7 mBq/cm^3 0–110% Vent Design Flow	E	3	

Table 7.5-2 ABWR PAM Variable List (Continued)

Variable	Range Required	Type	Cate-gory	Discussion Section
Airborn Radiohalogens and Particulates	37 µBq/cm^3 to 37Bq/cm^3	E	3	
Plant and Environs Radiation/Radioactivity (Portable Instruments)	10^{-5} Sv/h to10^{2} Sv/h photons 10^{-5} Sv/h to 10^{2} Sv/h, beta and low energy photons	E	3	Portable Instruments *
Meteorological Data (Wind Speed, Wind Direction, and Atmospheric Stability)	0–360°, 0–22 m/s, -5°C to 10°C	E	3	
On Site Analysis Capability (Primary Coolant, Sump and Space Containment Air Grab Sampling)	Refer to Regulatory Guide 1.97	E	3	
Secondary Containment Area Radiation	10^{-3} Sv/h to 10^{2} Sv/h	E	2	
Suppression Chamber Spray (Wetwell) Flow	0-110% Design Flow	D	2	Subsection 7.3.1.1.4

* Out of ABWR Standard Plant Scope

Table 7.5-3 ABWR Type A Variables

Suppression Pool Water Temperature
Drywell Pressure

Table 7.5-4 Moderate Frequency Incidents (Anticipated Operational Transients)

Event Description	NSOA Event Figure No	Tier 2 Section No.	Manual Action Variables*
Manual or Inadvertent SCRAM	15A-12	15A.6.3.3 Event 7	P_{RPV}, L_{RPV}
Loss of Plant Instrument or Service Air System	15A-13	15A.6.3.3 Event 8	T_{SP}, P_{RPV}, L_{RPV}
Recirculation Flow Control Failure—One RIP Runout	15A-14	15.4.5	P_{RPV}, L_{RPV}
Recirculation Flow Control Failure—One RIP Runback	15A-15	15.3.2	P_{RPV}, L_{RPV}
Three RIPs Trip	15A-16	15.3.1	P_{RPV}, L_{RPV}
All MSIV Closure	15A-17	15.2.4	T_{SP}, P_{RPV}, L_{RPV}
One MSIV Closure	15A-18	15.2.4	T_{SP}, P_{RPV}, L_{RPV}
Loss of All Feedwater Flow	15A-19	15.2.7	P_{RPV}, L_{RPV}
Loss of a Feedwater Heater	15A-20	15.1.1	ϕ, P_{RPV}, L_{RPV}
Feedwater Controller Failure—Runout of One Feedwater Pump	15A-21	15.1.2	P_{RPV}, L_{RPV}
Pressure Regulator Failure—Opening of One Bypass Valve	15A-22	15.1.3	P_{RPV}, L_{RPV}
Pressure Regulator Failure—Closing of One Control Valve	15A-23	15.2.1	P_{RPV}, L_{RPV}
Main Turbine Trip with Bypass System Operational	15A-24	15.2.3	T_{SP}, P_{RPV}, L_{RPV}
Loss of Main Condenser Vacuum	15A-25	15.2.5	P_{RPV}, L_{RPV}
Generator Load Rejection with Bypass System Operational	15A-26	15.2.2	T_{SP}, P_{RPV}, L_{RPV}
Loss of AC Power (Unit Auxiliary Transformer)	15A-27	15.2.6	T_{SP}, P_{RPV}, L_{RPV}
Inadvertent Startup of HPCF Pump	15A-28	15.5.1	ϕ
Inadvertent Opening of a Safety/Relief Valve	15A-29	15.1.4	T_{SP}, P_{RPV}, L_{RPV}
Main Turbine Trip with One Bypass Valve Failure	15A-31	15.2.3	T_{SP}, P_{RPV}, L_{RPV}
Generator Load Rejection with One Bypass Valve Failure	15A-32	15.2.2	T_{SP}, P_{RPV}, L_{RPV}

Table 7.5-4 Moderate Frequency Incidents (Anticipated Operational Transients) (Continued)

Event Description	NSOA Event Figure No	Tier 2 Section No.	Manual Action Variables*
Abnormal Startup of Idle Reactor Internal Pump	15A-45	15.4.4	P_{RPV}, L_{RPV}
Recirculation Flow Control Failure—All RIPs Runout	15A-46	15.4.5	ϕ, L_{RPV}
Recirculation Flow Control Failure—All RIPs Runback	15A-47	15.3.2	L_{RPV}
RHR Shutdown Cooling Increased Cooling	15A-50	15.1.6	T_{RPV}
Feedwater Controller Failure Runout of All Feedwater Pumps	15A-51	15.1.2	P_{RPV}, L_{RPV}
Pressure Regulator Failure— Opening of all Bypass and Control Valves	15A-52	15.1.3	P_{RPV}, L_{RPV}
Main Turbine Trip with Bypass Failure	15A-55	15.2.3	T_{SP},P_{RPV}, L_{RPV}
Generator Load Rejection with Bypass Failure	15A-56	15.2.2	T_{SP},P_{RPV}, L_{RPV}

* See Table 7.5-9 for Definition of Symbols

Table 7.5-5 Infrequent Incidents (Abnormal Operational Transients)

Event Description	NSOA Event Figure No	Tier 2 Section No.	Manual Action Variables*
Control Rod Withdrawal Error—Startup and Refueling Operations	15A-30	15.4.1	ϕ
Misplaced/Misoriented Fuel Bundle Accident	15A-57	15.4.7 15.4.8	None

* See Table 7.5-9 for Definition of Symbols

Table 7.5-6 Limiting Faults (Design Basis Accidents)

Event Description	NSOA Event Figure No	Tier 2 Section No.	Manual Action Variables*
Control Rod Ejection Accident	15A-33	15.4.9	None†
Control Rod Drop Accident	15A-34	15.4.10	None†
Control Rod Withdrawal Error—Power Operation	15A-35	15.4.2	None†
Fuel-Handling Accident	15A-36	15.7.4	R_{2C}
Loss-of-Coolant Accident Resulting from Spectrum of Postulated Piping Breaks Within the RCPB Inside Containment	15A-37 and 15A-38	15.6.5	$L_{RPV,}$ $L_{SP,}$ P_{RPV}, $P_{DW,}$ ϕ
Small, Large, Steam and Liquid Piping Breaks Outside Containment	15A-39 and 15A-40	15.6.2 15.6.4 15.6.6	T_{SP},P_{RPV}, L_{RPV}
Trip of All RIPs	15A-48	15.3.1	P_{RPV}, L_{RPV}
Loss of RHR Shutdown Cooling	15A-49	15.2-9	T_{RPV}
Pressure Regulator Failure— Closure of all Bypass and Control Valves	15A-53	15.2.1	T_{SP},P_{RPV}, L_{RPV}
Reactor Internal Pump Seizure	15A-58	15.3.3	P_{RPV}, L_{RPV}
Reactor Internal Pump Shaft Break	15A-59	15.3.4	P_{RPV}, L_{RPV}

* See Table 7.5-9 for Definition of Symbols

† Analysis indicates not plausible

Table 7.5-7 Special Events

Event Description	NSOA Event Figure No	Tier 2 Section No.	Manual Action Variables*
Shipping Cask Drop Spent Fuel	15A-60	15.7.5	None
Reactor Shutdown From Anticipated Transient Without SCRAM (ATWS)	15A-61	15.8	T_{SP},P_{RPV}, L_{RPV}, P_{DW}
Reactor Shutdown From Outside Control Room	15A-62	15A.6.6.3	T_{SP}, L_{SP},L_{RPV}, P_{RPV} Event 55
Reactor Shutdown Without Control Rods	15A-63	15A.6.6.3	T_{SP},ϕ, L_{RPV}, P_{RPV} Event 56

* See Table 7.5-9 for Definition of Symbols

Table 7.5-8 Summary of Manual Actions

Manual Action	Variable*	Source†
Decrease Reactor Power	ϕ	T
Initiation of Suppression Pool Cooling	T_{SP}	T
Initiation of Shutdown Cooling	P_{RPV},L_{RPV}	T
Manual Depressurization	P_{RPV},L_{RPV}	T
Initiation of N_2 Make Up and Purge	H_{2C},O_2	T
Initiation of Leakage Control Systems	N/A for ABWR	N/A for ABWR
Initiate Standby Liquid Control	ϕ,T_{SP}	T
Lowering Power by Lowering Water Level (ATWS)	ϕ,L_{RPV}	E
Emergency Action‡ If Exceed: Heat Capacity Temperature Limit Heat Capacity Level Limit Suppression Pool Load Limit Reference Leg Boiling Limit SRV Tailpipe Level Limit Maximum Primary Containment Water Level Limit Maximum Drywell Temperature Maximum Containment Temperature Maximum Containment Pressure Pressure Suppression Limit Maximum Secondary Containment Operating Valves Offsite Release Rate	 T_{SP},P_{RPV} T_{SP},L_{SP} L_{SP},P_{RPV} T_{DW},T_{RPV} (or P_{RPV}) L_{SP},P_{RPV} Lc ,P_{WW} T_{DW} P_{WW},L_{SP} P_{WW},L_{SP} P_{WW},L_{SP} T_{2C},R_{2C},L_{2C} R_E	E
Initiation of Drywell/Wetwell Sprays	T_{DW},T_{WW},P_{DW},L_{SP}	E
Initiation of Containment Flooding	P_{RPV},L_{RPV}	
Initiation of RPV Venting	P_{RPV},L_{RPV}	
Terminate Containment Flooding	Rc, L_{RPV}, Lc	

* See Table 7.5-9 for Definition of Symbols.

† E = EPG; T = Tier 2

‡ Scram, Emergency RPV Depressurization, RPV Flooding and/or Drywell Cooling.

Table 7.5-9 Definition of Symbols for Tables 7.5-4 Through 7.5-8

T_{SP}	—	Suppression Pool Temperature
T_{DW}	—	Drywell Temperature
T_{RPV}	—	Reactor Water Temperature
P_{RPV}	—	RPV Pressure
P_{WW}	—	Wetwell Pressure
L_{RPV}	—	RPV Level
L_{SP}	—	Suppression Pool Level
Ø	—	Neutron Flux
H_{2C}	—	Drywell/Wetwell Hydrogen Concentration
O_{2C}	—	Drywell/Wetwell Oxygen Concentration
P_{DW}	—	Drywell Atmospheric Pressure
T_{2C}	—	Temperature—Secondary Containment
R_{2C}	—	Radiation Level—Secondary Containment
L_{2C}	—	Sump Level—Secondary Containment
R_E	—	Exhaust Vent Radiation Level
L_C	—	Drywell Level
R_C	—	Radiation Level-Primary Containment