VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

June 16, 2000

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No.	00-303
SPS-LIC/BCB	R0
Docket Nos.	50-280/-281
License Nos.	DPR-32/-37

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
PROPOSED TECHNICAL SPECIFICATIONS AND BASES CHANGE -
RPS AND ESFAS ANALOG INSTRUMENTATION SURVEILLANCE FREQUENCY CHANGE FROM MONTHLY TO QUARTERLY

Pursuant to 10 CFR 50.90, Virginia Electric and Power Company requests amendments, in the form of revisions to the Technical Specifications (TS) to Facility Operating License Numbers DPR-32 and DPR-37 for Surry Power Station Units 1 and 2. The proposed amendments revise TS 3.7 and TS Tables 3.7-1, 3.7-2, 3.7-3, and 4.1-1. The changes a) revise the surveillance frequency for Reactor Protection System and Engineered Safety Features Actuation System analog channels from monthly to quarterly, b) increase the frequency for most permissives to a refueling interval, c) increase the time allowed to perform maintenance on an inoperable instrument channel, and d) revise associated action statements consistent with NUREG-1431. In addition, the TS 4.1 Basis is being revised to reflect the subject TS changes. A discussion of the proposed TS and Basis changes is provided in Attachment 1. The mark-up and typed pages are provided in Attachments 2 and 3, respectively.

The proposed amendment will result in a small increase in core damage frequency (CDF). This increase in CDF is bounded by the NRC's generic analyses of WCAP 10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," its supplements, and WCAP 14333, "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times" (see Attachment 1 for a complete discussion).

We have evaluated the significant hazards considerations associated with the proposed license amendment, as required by 10 CFR 50.92, and have determined that there are none (see Attachment 4 for a complete discussion). We have also determined that operation with the proposed changes will not result in any significant increases in the amounts of any effluents that may be released offsite and in any significant increases in individual or cumulative occupational radiation exposure. Therefore, the proposed

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amendment is eligible for categorical exclusion as set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed changes.

Due to the number of program and procedure changes necessary to implement these changes, we request ninety days from the issuance date of the amendments to implement the Technical Specifications changes. Should you have any questions or require additional information, please contact us.

Very truly yours,

W R Matthews

William R. Matthews
Vice President – Nuclear Operations

Attachments:
1. Discussion of Changes
2. Mark-up of Technical Specifications and Bases
3. Proposed Technical Specifications and Bases Changes
4. Significant Hazards Consideration Determination

Commitments made in this letter: None.

cc: U.S. Nuclear Regulatory Commission
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COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by William R. Matthews, who is Vice President - Nuclear Operations, of Virginia Electric and Power Company. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 16th day of June, 2000.

My Commission Expires: 3/31/2004.

Maggie McClure
Notary Public

(SEAL)

Attachment 1

Discussion of Changes

Surry Power Station
Units 1 and 2
Virginia Electric and Power Company

Discussion of Change

Introduction

In the early 1980s, in response to growing concerns of the impact of current testing and maintenance requirements on plant operation, particularly as related to instrumentation systems, the Westinghouse Owners Group (WOG) initiated a program to develop a justification to be used to revise generic and plant specific instrumentation Technical Specifications. Operating plants experienced many inadvertent reactor trips and safeguards actuations during performance of instrumentation surveillance, causing unnecessary transients and challenges to safety systems. Significant time and effort on the part of the operating staff was devoted to performing, reviewing, documenting and tracking the various surveillance activities, which in many instances seemed unwarranted based on the high reliability of the equipment. Significant benefits for operating plants appeared to be achievable through revision of instrumentation test and maintenance requirements. The results of these studies and the recommended changes to the testing of reactor protection and engineered safeguards instrumentation are documented in WCAP-10271, Supplement 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," and WCAP-10271, Supplement 2, "Evaluation of Surveillance Frequencies and Out of Service Times for the Engineered Safety Features Actuation System. "

The NRC completed an evaluation of surveillance testing at power, which indicated that testing in many areas could be reduced without any significant decrease in safety. These findings and recommendations are documented in NUREG-1366, "Improvement to Technical Specifications Surveillance Requirements," and Generic Letter 93-05, "Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation." Reduced surveillance testing of the Reactor Protection System (RPS) and Engineered Safety Features Actuation System (ESFAS) analog instrumentation was recommended. Recently, the NRC has completed a review of the Westinghouse Owners Group Topical Report WCAP-14333P-A, "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times." The NRC approved additional time for testing and extended allowed outage time for the RPS and ESFAS on July 15, 1998.

Consistent with the WCAPs, the associated NRC Safety Evaluations, and Generic Letter 93-05, we are proposing changes to the surveillance frequency of the Reactor Protection System and Engineered Safeguard Features analog instrumentation which will reduce testing at power. These proposed changes in the surveillance frequency reduce the number of high-risk surveillances performed at power.

In WCAP-10271 and its supplements, the WOG evaluated the impact of the proposed surveillance test interval and allowed outage times and changes on core damage frequency and public risk. The NRC staff concluded in its evaluation of the WOG evaluation that an overall upper bound increase of the core damage frequency due to the proposed surveillance test interval and allowed outage time changes is less than 6 percent for Westinghouse Pressurized Water Reactor plants. The NRC Staff also concluded that actual core damage frequency increases for individual plants are expected to be substantially less than 6 percent. The NRC Staff considered this core damage frequency increase to be small compared to the range of uncertainty in the core damage frequency analyses and therefore acceptable.

The WOG evaluated the impact of the additional relaxation of allowed outage times and completion times, and action statements on core damage frequency in WCAP-14333. The increase in core damage frequency (CDF) is 3.1% for those plants with two out of three logic schemes that have not implemented the proposed surveillance test interval, allowed outage times, and completion times evaluated in WCAP-10271 and its supplements. Surry has previously implemented the allowed outage times portion of WCAP-10271 and its supplements. The WCAP-14333 analysis calculates a significantly lower increase in core damage frequency than the WCAP-10271 analysis calculated. This can be attributed to more realistic maintenance intervals used in the current analysis and crediting the AMSAC system as an alternative method of initiating the auxiliary feedwater pumps.

The NRC performed an independent evaluation of the impact on core damage frequency and large early release fraction (LERF). The results of the staff's review indicate that the increase in core damage frequency is small (approximately 3.2%) and the large early release fraction would increase by only 4 percent for 2 out of 3 logic schemes that have not implemented the proposed surveillance test interval, allowed outage times, and completion times evaluated in WCAP-10271 and its supplements.

Background

WCAP-10271 and its supplements document the justification for plant specific Technical Specifications changes. The justification consists of a risk-informed evaluation of the effects of particular Technical Specification changes with consideration given to such things as safety, equipment requirements, human factors, and operational impact. The objective was to reach a balance in which safety and operability are ensured. The proposed Technical Specification revisions provide for increased test and maintenance times, less frequent surveillances, and testing in bypass for the RPS and ESFAS analog instrumentation channels, logic trains and permissives. In addition, the NRC determined that the requirement to routinely verify permissive status is a different consideration than the availability of trip or actuation channels which are required to change state on the occurrence of an event and for which the function availability is more dependent on the surveillance interval.

In February 1985, the NRC issued the SER (letter to J. J. Sheppard from Cecil O. Thomas dated February 21, 1985, Reference 6) for WCAP-10271 and Supplement 1. This SER approved quarterly testing, six hours to place a failed channel in a tripped mode, increased Allowed Outage Times (AOT) for test and maintenance, and testing in bypass for analog channels of the Reactor Protection System (RPS). The quarterly testing had to be conducted on a staggered basis.

On March 20, 1986, the WOG submitted WCAP-10271, Supplement 2 "Evaluation of Surveillance Frequencies and Out of Service Times for the Engineered Safety Systems Actuation System." On May 12, 1987 the WOG submitted WCAP-10271, Supplement 2, Revision 1 (References 7 and 8). Supplement 2 and Supplement 2, Revision 1 specifically demonstrated the applicability of the justification contained in WCAP-10271 to the ESFAS for two, three, and four loop plants with either relay or solid state systems.

In February 1989, the NRC issued the SER (Letter to Roger A. Newton from Charles E. Rossi dated February 22, 1989, Reference 11) for WCAP-10271, Supplement 2 and Supplement 2, Revision 1. The SER approved quarterly testing, six hours to place a failed channel in a tripped mode, increased allowed outage times for test and maintenance, and testing in bypass for analog channels of the ESFAS. The ESFAS functions approved in the SER were those presented in Appendix A1 of the referenced WCAPs. These functions are included in the Westinghouse Standard Technical Specifications. Staggered testing was not required for ESFAS analog channels and the requirement was removed from the RPS analog channels.

In their letter dated April 30, 1990 (Reference 12), the NRC issued the Supplemental SER (SSER) for WCAP-10271, Supplement 2 and Supplement 2, Revision 1. The SSER approved the surveillance test interval (STI) and allowed outage time (AOT) extensions for the ESFAS functions that were included in Appendix A2 of WCAP-10271, Supplement 2, Revision 1. The functions approved are associated with the Safety Injection, Steam Line Isolation, Main Feedwater Isolation, and Auxiliary Feedwater Pump Start signals. The configurations contained in the Appendix A2 are those that are not contained in the Westinghouse Standard Technical Specifications. The SSER also approved the extended AOTs for the Reactor Protection System actuation logic that were requested in WCAP-10271, Supplement 2, Revision 1, Appendix D.

The NRC has more recently completed a review of the Westinghouse Owners Group Topical Report WCAP-14333P, "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times" (Reference 14). Additional time for testing and extended allowed outage time for the RPS and ESFAS were approved by the NRC on July 15, 1998 (Reference 16).

<u>**Current Licensing Basis**</u>

The current surveillance interval for Reactor Protection and Engineered Safety Systems instrumentation, including analog channels, actuation logic and actuation relays is monthly. The Allowed Outage Time for the analog instrument channels is six hours, consistent with WCAP-10271. The minimum testing frequency for those instrument channels is based on an average unsafe failure rate of 2.5 E-6 failure/hr per channel. This failure rate is based on operating experience at conventional and nuclear units through the 1960s.

For the specified one month test interval, the average unprotected time is 360 hours in case of a failure occurring between test intervals. Thus, the probability of failure of one channel between test intervals is 360 x 2.5 E-6 or 0.9 E-3. Since two channels must fail in order to negate the safety function, the probability of simultaneous failure of two-out-of-three channels is $3(.9\ E\text{-}3)^2 = 2.4\ E\text{-}6$. This represents the fraction of time in which each three-channel system would have one operable and two inoperable channels and equals 2.4 E-6 x 8760 hours per year, or (approximately) 1 minute/year.

Current Design Basis

The reactor protection system provides the means for controlling the reactor in response to various measured primary and secondary variables associated with power, temperature, pressure, level, flow, and the availability of electric power. If the combination of monitored variables indicates an approach to unsafe conditions, the reactor protection system will initiate the appropriate protective action, e.g., load runback, prevention of rod withdrawal, or reactor trip (opening the reactor trip breakers).

The reactor protection system and the engineered safeguards are designed in accordance with IEEE-279, "Nuclear Power Plant Protection Systems, " August 1968. The reactor protection system is designed so that the most probable modes of failure in each channel result in a reactor trip signal. The protection system design combines redundant sensors and channel independence with coincident trip philosophy so that a safe and reliable system is provided in which a single failure will not defeat the channel function, cause a spurious trip, or violate reactor protection criteria.

Reactor Protection System channels are designed with sufficient redundancy for individual channel calibration and testing to be performed during power operation without degrading reactor protection. Exceptions are the backup channels such as reactor coolant pump breakers. Removal of one trip channel is accomplished by placing that channel in a partial-trip mode. For example, a two-out-of-three channel becomes a one-out-of-two channel. Testing will not cause a trip unless a trip condition exists concurrently in another channel. During such operation the active parts of the system continue to meet the single-failure criterion, since the channel under test is either tripped or makes use of superimposed test signals that do not negate the process signal. "One-out-of-two" systems are permitted to violate the single-failure criterion during channel bypass provided that acceptable reliability of operation can be otherwise demonstrated and the bypass time interval is short.

Discussion of Changes

With the issuance of the SER and SSER for the WCAP-10271, Supplement 2, the relaxations for the analog channels of the RPS and ESF are now the same and the special conditions applied to shared analog channels are no longer applicable. The AOTs for testing and maintenance of RPS and ESF Actuation Logic are also now the same.

Four specific changes were approved by the Nuclear Regulatory Commission for the Reactor Protection and Engineered Safeguards Analog Channels.

- The surveillance or test frequency may be changed from monthly to quarterly. Surveillance testing of most permissives may be extended to a refueling interval.

- The time allowed for a channel to be inoperable or out of service in an untripped condition may be changed from one hour to six hours. (WCAP-14333 further justified up to 72 hours.)

- The time a channel in a functional group may be bypassed to perform testing may be increased from two to four hours. (WCAP-14333 further justified up to 12 hours for bypassing a channel for surveillance testing.)

- Routine channel testing may be performed in the bypassed condition instead of the tripped condition, if bypass hardware is installed.

With the exception of the quarterly testing, these changes were approved and issued by the NRC on December 31, 1991, as Amendments 165 and 164 for Surry Units 1 and 2, respectively.

The NRC imposed five conditions on utilities seeking to implement the technical specification changes approved generically as a result of their review of WCAP-10271 and WCAP-10271, Supplement 1, and two conditions as a result of their review of WCAP-10271, Supplement 2 and Supplement 2, Revision 1. Two of the conditions imposed in the Reactor Protection System SER are no longer applicable due to approvals given in the ESF SER.

- The first condition in the Reactor Protection System SER requires the use of a staggered test plan for the RPS channels changed to the quarterly test frequency.

  In addition, the NRC Staff concluded that a staggered test strategy need not be implemented for ESFAS analog channel testing and is no longer required for RPS analog channel testing. This conclusion was based on the small relative contribution of the analog channels to RPS/ESFAS unavailability, process parameter signal diversity, and normal operational testing sequencing. Thus, the NRC removed the requirement for the RPS channels.

- The second condition in the Reactor Protection System SER requires that plant procedures require a common cause evaluation for failures in the Reactor Protection System analog channels changed to the quarterly testing frequency and additional testing for plausible common cause failures.

  From a practical standpoint, there are several types of failures which the staff does not regard as common cause failures. Failures such as instrument drift and power supply failures to a single channel are not considered common cause failures. Additional testing is not necessary for these failures or other failures if the cause of those other failures can be evaluated and shown not to affect multiple channels.

  Other failures of the type that are "announced" through control room alarms or annunciation or through other readily observed means need not be considered to be plausible common cause problems and do not require additional testing. Plausible common cause problems should be identified only where failure can be attributed to processes which are common to redundant equipment. That is, the underlying failure mechanism must be shown to have had the potential for causing failures in redundant channels. The failure may also be considered a plausible common cause problem if the failure was attributed to an improper test method that damaged components and that damage can only be found by subsequent testing. In this case, a review of the test records may indicate other channels subjected to the improper test method that warrant additional testing.

  Surry's existing programs and/or procedures will be reviewed and revised, if necessary, to ensure plausible RPS/ESFAS analog channel common cause failures are evaluated. Plant procedures will ensure that appropriate remedial action be taken, such as additional testing of the other channels in that function if a common cause failure is identified.

  Plausible common cause failures are identified only where failure can be attributed to the processes which are common to redundant equipment. The root cause mechanism must be shown to have the potential for causing failures in redundant channels. Failures that are not considered the results of a common cause include, but are not limited to, the following:

  - Instrument drift, normal drift correctable by calibration.
  - Loss of power to single channel.
  - Simple transistors/component failures that have no distinguishing characteristics may be considered a random failure.
  - Failures that are announced through control room alarms or through other readily observed means.

Failures that may be considered as plausible common cause failures include, but are not limited to the following:

- Failures cased by improper test method that damages components during the testing process.
- Test equipment out-of-calibration and/or tolerance used to calibrate redundant equipment.
- Incorrect steps in procedures used to calibrate redundant modules.
- Incorrect engineering calculations and/or setpoints used to calibrate redundant modules.
- Incorrect part installed in similar loops during module repair.
- Design changes resulting in change of component function.
- Manufacturing changes resulting in changes in module performance.

• The third condition in the Reactor Protection System SER requires installed hardware capability for testing in the bypass mode. As stated by the NRC in the safety evaluation for WCAP-10271:

"Testing of the RPS analog channels in the bypassed condition by use of temporary jumpers or by lifting leads is not acceptable. The chance of personnel errors leaving a number of channels in the bypassed condition would be too large for the routine use of such methods. Therefore, licensees choosing this option to perform routine channel testing in the bypass mode should ensure that the plant design allows testing in bypass without lifting leads or installing temporary jumpers."

Surry does not have the hardware capability for full bypass testing of each RPS and ESF instrument channel. Only those instrument channels that have hardware installed to permit testing in bypass without lifting leads or installing jumpers will be routinely tested in bypass. Analog channel testing with an inoperable channel will be completed with the channel being tested in trip. However, the inoperable channel may be bypassed to perform surveillance testing on another channel of the same functional unit as permitted by Technical Specifications.

• The fourth condition in the RPS SER involves channels that provide input to both the RPS and the ESF. As stated by the NRC in the safety evaluation for WCAP-10271:

"Now that the ESF SER has been issued and all of the relaxations for the RPS analog channels are applicable to the ESF analog channels, this condition does not apply."

- The fifth condition in the RPS SER, and second in the ESF SER, addresses setpoint drift. As stated by the NRC in the safety evaluation for WCAP-10271:

  "The bistable equipment is inherently stable and should show good performance. In many cases, channels are not recalibrated for many months. A review of the "as found" and "as left" data over a twelve-month period should provide sufficient information to address the adequacy of the existing setpoints and allowable values."

  A review of the RPS and ESF instrument loops was completed for Surry. In every case, the review established at a 95% confidence level that each instrument loop was found to exhibit drift well below the assumed value of 1 percent per month. In addition, the review evaluated the impact of drift on control parameters and confirmed that the instrument drift remains acceptable for plant control systems.

- This first condition in the ESF SER requires that the plant specific applications must confirm the applicability of the generic analyses to the plant.

  The Reactor Protection and Engineered Safety Features protection functions and the associated logic schemes evaluated in WCAP-10271, Supplements 1 and 2 are representative of the three loop protective functions and logic schemes at Surry.

In addition, the NRC determined that the requirement to routinely verify permissive status is a different consideration than the availability of trip or actuation channels which are required to change state on the occurrence of an event and for which the function availability is more dependent on the surveillance interval. The definition of the channel check includes comparison of the channel status with other channels for the same parameter. For the RPS and ESFAS interlocks, the NRC justified and accepted the change from the monthly surveillance requirement to at least once every 18 months.

Subsequently, the Nuclear Regulatory Commission approved WCAP-14333, which justified additional allowed outage times and bypass time changes for the RPS and ESFAS.

- For analog channels, the allowed outage time was increased from 6 hours to 72 hours, and the allowed testing bypass time was increased from 4 to 12 hours.

- For logic cabinets and master and slave relays, the allowed outage time was increased from 6 to 24 hours.

In order to implement the extended test times and allowed outage times, the staff has imposed two conditions on each licensee.

- The first condition was to confirm the applicability of the WCAP-14333 analyses for their plants.

  Attached Tables 1, 2, and 3 establish the applicability of Surry's RPS and ESFAS to the WCAP-14333 analysis.

- The second condition was to address the Tier 2 and 3 risk analyses, including the configuration risk management program (CRMP) insights which confirm that these insights are incorporated into their decision making process before taking equipment out of service.

  Consistent with the Safety Evaluation Report for WCAP-14333, Virginia Power has examined the impact of outages of the analog instruments and the RPS and ESFAS logic trains in order to identify potential limitations of concurrent equipment outages. When compared to the base case with all risk significant equipment available, there were no significant increases in component importance due to the unavailability of any of the channels or trains identified above. As a result, there is no need for special restrictions to avoid risk significant configurations.

  Instantaneous risk is already evaluated and controlled by the Virginia Power Maintenance Rule Program, consistent with 10 CFR 50.65(a)(4). Risk evaluations for surveillance, post-maintenance testing, and corrective and preventive maintenance consider the need to preclude potentially high-risk configurations. In addition, these evaluations address additional equipment outages that occur during the allowed outage time period and ensure that equipment removed from service immediately prior to or during the proposed AOT will be appropriately assessed from a risk perspective. Planned configurations of high risk are avoided and in the case of emergent work, high-risk configurations are avoided or minimized consistent with 10 CFR 50.65(a)(4).

Specific Changes

The following describes the proposed changes to the RPS and ESF instrumentation operability and surveillance requirements. In addition to the changes to implement the NRC approved recommendation of WCAPs-10271 and 14333, other changes consistent with the accident analysis and NUREG-1431, Revision 1, "Standard Technical Specifications for Westinghouse Plants," are being proposed.

Changes to TS Page 3.7-1

- Delete TS 3.7.A - The statement to permit testing and continued operation with an inoperable channel/train is now included in the appropriate Action Statement.

Changes to Table 3.7-1, Reactor Trip Instrument Operating Conditions and Actions

- Revise Actions 2, 6, 7, 9 and 12 to increase the time that an inoperable analog instrument channel may be maintained in an untripped condition from 6 hours to 72 hours consistent with the WCAPs and NRC SERs.

- Revise Action 11 to increase the time that the logic cabinets, master, or slave relays may be inoperable to 24 hours consistent with the WCAPs and NRC SERs. Reduce the time required to be in Hot Shutdown from 8 hours to 6 hours. These changes are consistent with Standard Technical Specifications for Westinghouse Pressurized Water Reactors (STS) NUREG-1431, Rev. 1.

- Revise Actions 2, 6, and 7 to increase the time that an inoperable channel may be bypassed to allow testing from 4 to 12 hours.

- Revise Action 7 for Functional Units 7, 9, 10, 11 a and b, 13, and 14 ("at-power" trips) to only require a power reduction to less than the P-7 setpoint (10%) rather than Hot Shutdown and subsequently then cooled down to Cold Shutdown. Action 7 is consistent with NUREG-1431, Rev. 1. The Chapter 14 accident analyses do not assume operability of these "at-power trips" below 10% power.

- Revise the required action for Functional Units 12 and 17 from Action 7 to 6. These functions are not "at-power trips." This change is necessary due to changing action statement 7 to only require a power reduction to less that 10% (at-power trips).

- Revise Action 9 for Functional Unit 16 to require reducing power to less than 10% power when the conditions of the action are not met. This eliminates plant operation between the P-8 and P-7 setpoints and additional time to place the unit in Hot Shutdown if another channel becomes inoperable. This restriction is required to be consistent with the out of service and shutdown action times assumed in the WCAP's risk analysis performed to support the AOT extension from 1 to 6 and ultimately to 72 hours, and subsequently approved by the NRC.

- Revise Action 1 to require that the Unit be shutdown <u>and</u> the trip breakers opened rather than shutdown <u>and/or</u> open the reactor trip breakers. This action is consistent with NUREG-1431, Revision 1.

- In addition to AOT and testing extensions, revise Action 2.A to include REACTOR CRITICAL as a Mode where plant operation may continue if the conditions are met.

Combined the A and B portions of the Action Statement, eliminating the alpha characters.

- Revise Action 3.b to permit increasing power above 10% power or reducing power below P-6 within 24 hours with an inoperable Intermediate Range channel, consistent with NUREG-1431, Revision 1 and the safety analysis.

- Revise Action 4.a to require suspending reactivity changes that are more positive than necessary to meet the required shutdown margin or refueling boron concentration limits with an inoperable source range below P-6. In addition, establish an action for two inoperable source range channels below P-6. These changes are consistent with NUREG-1431, Revision 1.

- Revise Action 7 to only require the plant to be reduced in power to less than P-7 (10%) if the conditions of the action statement cannot be met for the "at-power trips." These changes are consistent with NUREG-1431, Revision 1 and the safety analysis.

- Revise Actions 6 and 7 from "equal to Minimum Channels Operable" to "less than the Total Number of Channels" for entry into the action statement. This will eliminate unnecessary entry into TS 3.0.1 and unnecessary plant cooldown if a second channel becomes inoperable below 10% power.

- Revise Action 8.A to permit a trip breaker to be bypassed for up to 4 hours for concurrent surveillance testing of the trip breaker and actuation logic provided the other trip breaker is operable consistent with WCAP-14333.

- Action 10 is not necessary; Action 9 has been modified to include the appropriate actions for the associated functional unit. Therefore Action 10 is deleted.

- Revise Action 11 to permit 24 hours to return the logic to operable status prior to requiring a unit shutdown consistent with WCAP-14333.

- Delete Action 12. The revised Action 7 establishes the same requirements for the "at-power trips." Reduce power to less than the P-7 setpoint (10%).

<u>Changes to Table 3.7-2, Engineered Safeguards Action Instrument Operating Conditions and Actions</u>

- Revise Actions 17, 20, 25 26 to increase the time that an inoperable analog instrument channel may be maintained in an untripped condition from 6 hours to 72 hours consistent with the WCAPs and NRC SERs.

- Revise Actions 14 and 22 to increase the time that the logic cabinets, master, or slave relays may be inoperable to 24 hours, consistent with the WCAPs and NRC SERs.

- Revise Actions 17, 20, 25 and 26 to increase the time that an inoperable channel may be bypassed from 4 to 12 hours to allow testing.

- Revise the Action for Functional Units 3d and 3e from 21 to 24, which only requires a unit shutdown and cool down to less than 350°F and 450 psig consistent with NUREG-1431, Revision 1 and the safety analysis.

- Revise the Action for Functional Units 4a and 4b from 20 to the new 26 which will require the emergency diesel generator to be declared inoperable if the loss of power instruments cannot meet the conditions of the actions for an inoperable channel consistent with NUREG-1431, Revision 1.

- In addition to AOT extension, Action 14 is revised to reduce the time permitted to place the unit in Hot Shutdown to 6 hours if the channel is not returned to service within the new 24 hour allowed outage time. This action is consistent with NUREG-1431.

- Establish Action 26 to include the AOT and testing extensions and require that the associated emergency diesel generator (EDG) be declared inoperable and enter the EDG action statement if the required conditions are not satisfied in the time.

Table 3.7-3, Instrument Operating Conditions for Isolation Functions and Actions

- Revise the Action for Functional Unit 1.b.3 from 15 to 21 consistent with NUREG-1431, Revision 1.

- Include a Permissible Bypass condition for Functional Unit 3, Turbine Trip and Feedwater Isolation, to eliminate the need for the protection when all MFRVs and SG FWIVs and associated bypass valves are closed and deactivated or isolated by manual valves, consistent with NUREG-1431, Revision 1.

- In addition to AOT and testing extension, Action 20 is revised to only require a unit shutdown and cool down to less than 350°F and 450 psig rather than to Hot Shutdown and then cooled down to Cold Shutdown. This action is consistent with NUREG-1431, Revision 1 and the safety analysis.

- In addition to AOT extension, Action 22 is revised to reduce the time to place the unit in Hot Shutdown to 6 hours and increase the time to go to 350°F and 450 psig to 12 hours if the channel cannot be returned to service within the new 24 hour allowed outage time. These actions are consistent with NUREG-1431.

Table 4.1, Minimum Frequencies for Checks, Calibrations and Test of Instrument Channels

The surveillance test interval for the RPS and ESF analog instruments is changed from monthly to quarterly. The following summarizes the instruments with surveillances that are changed from monthly to quarterly:

- Nuclear Power Range (Table 4.1-1 Item 1)
- Nuclear Intermediate Range (Table 4.1-1 Item 2) Quarterly testing not required. The requirement for testing Prior to Startup is changed from 7 days to 31 days.
- Reactor Coolant Temperature (Table 4.1-1 Item 4)
- Reactor Coolant Flow (Table 4.1-1 Item 5)
- Pressurizer Water Level (Table 4.1-1 Item 6)
- Pressurizer Pressure (High and Low) (Table 4.1-1 Item 7)
- 4KV Voltage and Frequency (Table 4.1-1 Item 8)
- Steam Generator Level (Table 4.1-1 Item 11)
- Recirculation Mode Transfer (Table 4.1-1 Item 15a)
- Reactor Containment Pressure-CLS (Table 4.1-1 Item 17)
- Steam Line Pressure (Table 4.1-1 Item 22)
- Turbine First Stage Pressure (Table 4.1-1 Item 23)
- Auxiliary Feedwater (Table 4.1-1 Item 32a)
- Auxiliary Feedwater (Table 4.1-1 Item 32b) is changed from NA to R consistent with NUREG-1431, Revision 1.
- Loss of Voltage and Degraded Voltage (Table 4.1-1 Item 33a and b)
- Steam/Feedwater Flow and Low S/G Water Level (Table 4.1-1 Item 39)
- Turbine Trip and Feedwater Isolation S/G Water Level High (Table 4.1-1 Item 41a)

Reactor Trip System Interlocks Table 4.1-1 Item 42 - Change the test frequency of the surveillance test requirement from monthly to refueling consistent with WCAP-10271. The currently specified test interval for interlock channels allows the surveillance requirement to be satisfied by verifying that the permissive logic is in its required state using the annunciator status light. The surveillances, as currently required, only verifies the status of the permissive logic and does not address verification of channel setpoint or operability. The setpoint verification and channel operability are verified after a refueling shutdown. The definition of the channel check includes comparison of the channel status with other channels for the same parameter. The requirement to routinely verify permissive status is a different consideration than the availability of trip or actuation channels which are required to change state on the occurrence of an event and for which the function availability is more dependent on the surveillance interval.

Items 12, 13, 14, 16, 18, 21, 24, 25, 27, 29, and 31 are being deleted. These items do not generate an output signal that is used in the RPS or ESF Systems. Removal of these instruments is consistent with NUREG-1431.

Notes are being added to functional units 8, 32b, 33a, and 33b to identify that setpoint verification is not required during the quarterly test.

The test frequency definitions at the end of Table 4.1-1 are being revised to be consistent with the WCAPS and NUREG-1431 (e.g., "P" – changes from 7 to 31 days, "M" changed to 31 days). Any notes in the specific channel descriptions are also changed for consistency (i. e., Auxiliary Feedwater 32.a and b).

<u>TS 4.1 Basis</u>

The basis of TS 4.1 is being revised to address quarterly testing and the extended allowed outage times, as follows.

- Delete second paragraph in the check section

- Replace the Testing Section of the bases with the following three paragraphs:

  The OPERABILITY of the Reactor Protection System and ESFAS instrumentation systems and interlocks ensures that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic and sufficient redundancy are maintained to permit a channel to be out of service for testing or maintenance consistent with maintaining an appropriate level of reliability of the RPS and ESFAS instrumentation and 3) sufficient system functional capability is available from diverse parameters.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. Specific surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, EVALUATION OF SURVEILLANCE FREQUENCIES AND OUT OF SERVICE TIMES FOR THE REACTOR TRIP INSTRUMENTATION SYSTEM, and supplements to that report, WCAP-10271 Supplement 2, EVALUATION OF SURVEILLANCE FREQUENCIES AND OUT OF SERVICE TIMES FOR THE ENGINEERED SAFETY FEATURES ACTUATION SYSTEM, and supplements to that report, and WCAP-14333P, PROBABILISTIC RISK ANALYSIS OF THE RPS AND ESF TEST TIMES AND COMPLETION TIMES, as approved by the NRC and documented in SERs dated February 21, 1985, February 22, 1989, the SSER dated April 30, 1990 for WCAP-10271 and July 15, 1998 for WCAP-14333P.

Surveillance testing of instrument channels is routinely performed with the channel in the tripped condition. Only those instrument channels with hardware permanently installed that permits bypassing without lifting a lead or installing a jumper are routinely tested in the bypass condition. However, an inoperable channel may be bypassed by lifting a lead or installing a jumper to permit surveillance testing of another instrument channel of the same functional unit.

Risk Assessment

In WCAP-14333, the WOG evaluated the impact of the additional relaxation of allowed outage times and completion times, and action statements on core damage frequency. The change in core damage frequency is 3.1 percent for those plants with two out of three logic schemes that have not implemented the proposed surveillance test interval, allowed outage times, and completion times evaluated in WCAP-10271 and its supplements. This analysis calculated a significantly lower increase in core damage frequency than the WCAP-10271 analysis calculated. This can be attributed to more realistic maintenance intervals used in the current analysis and crediting the AMSAC system as an alternative method of initiating the auxiliary feedwater pumps. The NRC Staff considered this core damage frequency increase to be small compared to the range of uncertainty in the core damage frequency analyses and therefore acceptable.

The NRC performed an independent evaluation of the impact on core damage frequency (CDF) and large early release fraction (LERF). The results of the staff's review indicate that the increase in core damage frequency is small (approximately 3.1%) and the large early release fraction would increase by only 4 percent for 2 out of 3 logic schemes that have not implemented the proposed surveillance test interval, allowed outage times, and completion times evaluated in WCAP-10271 and its supplements. Surry has implemented a portion of WCAP-10271 and therefore, the change in CDF and LERF would be smaller.

The impact on the Probabilistic Risk Assessment (PRA) due to an increase in the RPS and ESFAS surveillance interval from monthly to quarterly is considered minor. The evaluation used the Surry PRA model to estimate an overall change in the CDF of approximately one percent. For configurations involving the instrumentation and protection components such as those addressed by this package, the Large Early Release Frequency (LERF) impact is typically bounded by the CDF impact.

The proposed changes are consistent with the NRC staff's letters dated February 21, 1985, February 22, 1989, April 30, 1990, and July 15, 1998, to the WOG regarding evaluation of WCAP-10271, WCAP-10271 Supplement 1, WCAP-10271 Supplement 2 WCAP-10271 Supplement 2, Revision 1, and WCAP-14333.

Environmental Assessment

The proposed Technical Specifications changes address instrumentation issues and have no environmental impact or increase in the individual or cumulative occupational radiation exposure. The protection circuitry retains sufficient redundancy and diversity to ensure that core protection is maintained and hence, the risk of offsite release is not increased. The Reactor Protection System and Engineered Safety Features Actuation System will continue to be operated and tested in the same manner. No new effluents or effluent release paths are created as a result of the proposed Technical Specifications changes to the Reactor Protection System and Engineered Safety Features Actuation System instrumentation and actuation logic operability and surveillance requirements. The proposed changes will continue to ensure the RPS and ESFAS will be operable as assumed in the safety analysis to mitigate the consequences of the accidents identified above and therefore, there is no environmental impact as a result of the proposed Technical Specifications changes.

References

1. WCAP-10271, "EVALUATION OF SURVEILLANCE FREQUENCIES AND OUT OF SERVICE TIMES FOR THE REACTOR PROTECTION INSTRUMENTATION SYSTEM," dated January 1983.

2. WCAP-10271 Supplement 1, "EVALUATION OF SURVEILLANCE FREQUENCIES AND OUT OF SERVICE TIMES FOR THE REACTOR PROTECTION INSTRUMENTATION SYSTEM," dated July 1983.

3. Letter (OG-86) from J. J. Sheppard (WOG) to H. R. Denton (NRC) dated February 3, 1983 (WCAP-10271 submittal).

4. Letter from C. O. Thomas (NRC) to J. J. Sheppard (WOG) dated July 28, 1983 (NRC Request Number 1 for Additional Information on WCAP-10271).

5. Letter (OG-106) from J. J. Sheppard (WOG) to C. O. Thomas (NRC) dated October 4, 1983 (WCAP-10271 Supplement 1 and question response submittal).

6. Letter from C. O. Thomas (NRC) to J. J. Sheppard (WOG) dated February 21, 1985 (NRC Safety Evaluation Report for WCAP-10271).

7. WCAP-10271 Supplement 2 "EVALUATION OF SURVEILLANCE FREQUENCIES AND OUT OF SERVICE TIMES FOR THE ENGINEERED SAFETY FEATURES ACTUATION SYSTEM," dated February 1986.

8. WCAP-10271-P-A Supplement 2, Revision 1 "EVALUATION OF SURVEILLANCE FREQUENCIES AND OUT OF SERVICE TIMES FOR THE ENGINEERED SAFETY FEATURES ACTUATION SYSTEM," dated June 1990.

9. Letter (OG-87-15) from Roger A. Newton (WOG Chairman) to R. W. Starosticki (NRC) dated May 12, 1987 (WCAP-10271 Supplement 2, Revision 1 submittal).

10. Letter H. R. Denton (NRC) to L. D. Butterfield (WOG Chairman) dated July 24, 1985 (Comments to Guidelines).

11. Letter Charles E. Rossi (NRC) to Roger A. Newton (WOG) dated February 22, 1989 (NRC Safety Evaluation Report for WCAP-10271 Supplement 2 and Supplement 2, Revision 1).

12. Letter Charles E. Rossi (NRC) to Gerard T. Goering (WOG) dated April 30, 1990 (NRC Supplemental Safety Evaluation Report for WCAP-10271 Supplement 2, Revision 1).

References (continued)

13. Letter Charles E. Rossi (NRC) to Robert F. Janacek, Chairman BWR Owners Group dated April 27, 1988, "STAFF GUIDANCE FOR LICENSEE DETERMINATION THAT THE DRIFT CHARACTERISTICS FOR INSTRUMENTATION USED IN RPS CHANNELS ARE BOUNDED BY NEDC-30851P ASSUMPTIONS WHEN THE FUNCTIONAL TEST INTERVAL IS EXTENDED FROM MONTHLY TO QUARTERLY."

14. WCAP-14333P, "PROBABILISTIC RISK ANALYSIS OF THE RPS AND ESFAS TEST TIMES AND COMPLETION TIMES," dated May 1995.

15. WCAP-14334NP, "PROBABILISTIC RISK ANALYSIS OF THE RPS AND ESF TEST TIMES AND COMPLETION TIMES," dated May 1995.

16. Letter Thomas H. Essig, (NRC) to Mr. Louis F. Liberatori Jr., Chairman Westinghouse Owners Group, dated July 15, 1998, "REVIEW OF WESTINGHOUSE OWNERS GROUP TOPICAL REPORTS WCAP 14333P AND WCAP-14334NP, DATED MAY, 1995, PROBABILISTIC RISK ANALYSIS OF THE RPS AND ESFAS TEST TIMES AND COMPLETION TIMES."

Table 1
WCAP-14333 Implementation Guidelines: Applicability of the Analysis General Parameters

Parameter	WCAP-14333 Analysis	Plant Specific Parameter
Logic Cabinet Type (1)	Relay and SSPS	Relay
Component Test Intervals (2)		
Analog channels *(Reference TS Table 4.1-1)*	3 months	3 months (proposed) (11)
.. Logic cabinets (SSPS)	2 months	N/A
.. Logic cabinets (Relay) *(Ref. TS Table 4.1-1 Item #26)*	1 month	1 month
.. Master Relays (SSPS)	2 months	N/A
.. Master Relays (Relay)	1 month	Various ***
.. Slave Relays	3 months	Various ***
.. Reactor trip breakers *(Ref. TS Table 4.1-1 Items #30)*	2 months	1 month ****
Analog Channel Calibrations (3)		
.. Done at-power *(per Periodic Tests)*	Yes	Infrequent
.. Interval *(Ref. TS Table 4.1-1)*	18 months	18 months
Typical At-Power Maintenance Intervals (4)		
.. Analog channels	24 months	Infrequent
.. Logic cabinets (SSPS)	18 months	N/A
.. Logic cabinets (Relay)	12 months	Infrequent
.. Master relays (SSPS)	Infrequent (5)	N/A
.. Master relays (Relay)	Infrequent (5)	Infrequent
.. Slave relays	Infrequent (5)	Infrequent
.. Reactor trip breakers	12 months	Infrequent
AMSAC (6) *(See 11448(11548)-ESK-5K)*	Credited for AFW pump start	Provides AFW pump start
Total Transient Event Frequency (7) *Calc SM-1174, Event trees*	3.6	1.95
ATWS Contribution to CDF (current PRA model) (8) *Calc SM-1174, p. 62*	8.4E-06	4.00E-09
Total CDF from Internal Events (current PRA model) (9) *SM-1174, p. 62*	5.8E-05	2.99E-05 *
Total CDF from Internal Events (IPE) (10) *Calc SM-1174, p. 62*	Not Applicable	7.38E-05 **

Notes for Table 1

1. Indicate type of logic cabinet; SSPS or Relay (both are included in WCAP-14333).
2. Fill in applicable test intervals. If the test intervals are equal to or greater than those used in WCAP-14333, the analysis is applicable to your plant.
3. Indicate if channel calibration is done at-power and, if so, fill in the interval. If channel calibrations are not done at-power or if the calibration interval is equal to or greater than that used in WCAP-14333, the analysis is applicable to your plant.
4. Fill in the applicable typical maintenance intervals or fill in "equal to or greater than" or "less than". If the maintenance intervals are equal to or greater than those used in WCAP-14333, the analysis is applicable to your plant.
5. Only corrective maintenance is done on the master and slave relays. The maintenance interval on typical relays is relatively long, that is, experience has shown they do not typically completely fail. Failure of slave relays usually involves failure of individual contacts. Fill in "infrequent" if this is consistent with your plant experience. If not, fill in the typical maintenance interval. If "infrequent" slave relay failures are the norm, then the WCAP-14333 analysis is applicable to your plant.
6. Indicate if AMSAC will initiate AFW pump start. If yes, then the WCAP-14333 analysis is applicable to your plant.
7. Include total frequency for initiators requiring a reactor trip signal to be generated for event mitigation. This is required to assess the importance of ATWS events to CDF. Do not include events initiated by a reactor trip.
8. Fill in the ATWS contribution to core damage frequency (from at-power, internal events). This is required to determine if the ATWS event is a large contributor to CDF.
9. Fill in the total CDF from internal events (including internal flooding) for the most recent PRA model update. This is required for comparison to the NRC's risk-informed CDF acceptance guidelines. *
10. Fill in the total CDF from internal events from the IPE model (submitted to the NRC in response to Generic Letter 88-20). If this value differs from the most recent PRA model update CDF provide a concise list of reasons, in bulletized form, describing the differences between the models that account for the change in CDF. **
11. If your analog channel test interval is 1 month, the STI increase justified and approved by the NRC in WCAP-10271 has not been implemented in your plant, even so, this analysis still remains applicable.

* This figure does not include the contribution from internal flooding.

** The calculated CDF has decreased due to two primary reasons: (1) The addition of the SBO EDG to the plant and its incorporation into the PRA model, and (2) more extensive modeling of all systems. This latter contributor is too extensive to provide details here.

*** Slave and master relay testing is performed at a variety of intervals for different components in the RPS and ESFAS systems. These intervals are typically comparable to the generic WCAP-14333 assumptions.

**** The reactor trip breakers are tested on a schedule that is comparable to the generic WCAP-14333 analysis. The WOG is presently reviewing a draft WCAP that will extend this interval. It will be evaluated for implementation following NRC approval. In the interim, trip breaker testing is explicitly included in the instantaneous risk analysis per the 10 CFR 50.65 a(4) risk management program at Surry.

Table 2
WCAP-14333 Implementation Guidelines: Applicability of Analysis Reactor Trip Actuation Signals

Event	WCAP-14333 Analysis	Plant Specific Parameter (1)
Large LOCA	Not Required	N/A
Medium LOCA	Not Required	N/A
Small LOCA	Nondiverse (12) w/OA (13)	Agree (TS Table 3.7-2 Functional Unit 1)
Steam Generator Tube Rupture	Nondiverse w/OA	Agree (TS Table 3.7-2 Functional Unit 1)
Interfacing System LOCA	Not Required	N/A
Reactor Vessel Rupture	Not Required	N/A
Secondary Side Breaks	Nondiverse w/OA	Agree (TS Table 3.7-2 Functional Unit 1)
Transient Events, such as:	Diverse (4) w/OA	Agree *
Reactor Trip	Generated by RPS	Agree (TS Table 3.7-1)
Loss of Offsite Power	Not Required by RPS	N/A
Station Blackout	Not Required by RPS	N/A
Loss of Service Water or Component Cooling Water	Nondiverse w/OA	Agree (0-AP-12.00 and 12.01 for SW, 1(2)-AP-15.00 for CC)

** events without automatic protection are addressed by procedures which direct a reactor trip when required to maintain plant control or safety margins.*

Notes for Table 2

12. Fill in "agree" if your plant design and operation is consistent with this analysis, that is, the noted reactor trip signals are available at a minimum. If not, explain the difference. If "agree" is listed for each event, then the WCAP-14333 analysis is applicable to your plant.
13. Nondiverse means that (at least) one signal will be generated to initiate reactor trip for the event.
14. OA indicates that an operator could take action to initiate reactor trip for the event, that is, there is sufficient time for action and procedures are in place that will instruct the operator to take action.
15. Diverse means that (at least) two signals will be generated to initiate reactor trip for the event.

Table 3
WCAP-14333 Implementation Guidelines: Applicability of Analysis (Cont'd)
Engineered Safety Features Actuation Signals

Safety Function	Event	WCAP-14333 Analysis Assumption	Plant Specific Parameter (1)
Safety Injection	Large LOCA	Nondiverse (12)	Agree (TS Table 3.7-2 Functional Unit 1)
	Medium LOCA	Nondiverse, OA (13) by SI switch on main control board	Agree (TS Table 3.7-2 Functional Unit 1)
	Small LOCA	Nondiverse, OA by SI switch on main control board, OA of individual components	Agree (TS Table 3.7-2 Functional Unit 1)
	Interfacing Systems LOCA	Nondiverse, OA by SI switch on main control board, OA of individual components	Agree (TS Table 3.7-2 Functional Unit 1)
	SG Tube Rupture	Nondiverse, OA by SI switch on main control board, OA of individual components	Agree (TS Table 3.7-2 Functional Unit 1)
	Secondary Side	Nondiverse, OA by SI switch on main control	Agree (TS Table 3.7-2 Functional Unit 1)
Auxiliary Feedwater Pump Start	Events generating SI signal	Pump actuation on SI signal	Agree (TS Table 3.7-2 Functional Unit 3)
Main Feedwater Isolation	Secondary Side Breaks	Nondiverse	Agree (TS Table 3.7-3 Functional Unit 3)
Steamline Isolation	Secondary Side Breaks	Nondiverse	Agree (TS Table 3.7-3 Functional Unit 2)
Containment Spray Actuation	All events	Nondiverse signal	Agree (TS Table 3.7-2 Functional Unit 2)
Containment Isolation	All events	From SI signal	Agree (TS Table 3.7-3 Functional Unit 1)
Containment Cooling	All events	From SI signal	Agree (TS Table 3.7-2 Functional Unit 2)

Notes:

16. Fill in "agree" if your plant design and operation is consistent with this analysis, that is, the noted engineered safety features actuation signals are available at a minimum. If not, explain the difference. If "agree" is listed for each event, then the WCAP-14333 analysis is applicable to your plant.
17. Nondiverse means that (at least) one signal will be generated to initiate the engineered safety feature noted for the event.
18. OA indicates that an operator could take action to initiate the engineered safety feature for the event, that is, there is sufficient time for action and procedures are in place that will instruct the operator to take action.

Attachment 2

Mark-up of Unit 1 and Unit 2 Technical Specifications Changes

Surry Power Station
Units 1 and 2
Virginia Electric and Power Company

3.7 INSTRUMENTATION SYSTEMS

Operational Safety Instrumentation

Applicability

Applies to reactor and safety features instrumentation systems.

Objectives

To ensure the automatic initiation of the Reactor Protection System and the Engineered Safety Features in the event that a principal process variable limit is exceeded, and to define the limiting conditions for operation of the plant instrumentation and safety circuits necessary to ensure reactor and plant safety.

Specification

A. ~~During on-line testing or in the event of a subsystem instrumentation channel failure, plant operation at RATED POWER shall be permitted to continue in accordance with Tables 3.7-1 through 3.7-3.~~

A ~~B.~~ The Reactor Protection System instrumentation channels and interlocks shall be OPERABLE as specified in Table 3.7-1.

B ~~C.~~ The Engineered Safeguards Actions and Isolation Function Instrumentation channels and interlocks shall be OPERABLE as specified in Tables 3.7-2 and 3.7-3, respectively.

C ~~D.~~ The Engineered Safety Features Initiation instrumentation setting limits shall be as stated in Table 3.7-4.

D ~~E.~~ The explosive gas monitoring instrumentation channel shown in Table 3.7-5(a) shall be OPERABLE with its alarm setpoint set to ensure that the limits of Specification 3.11.A.1 are not exceeded.

1. With an explosive gas monitoring instrumentation channel alarm setpoint less conservative than required by the above specification, declare the channel inoperable and take the action shown in Table 3.7-5(a).

2. With less than the minimum number of explosive gas monitoring instrumentation channels OPERABLE, take the action shown in Table 3.7-5(a). Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, prepare and submit a Special Report to the Commission (Region II) to explain why this inoperability was not corrected in a timely manner.

E ~~7.~~ The accident monitoring instrumentation listed in Table 3.7-6 shall be OPERABLE in accordance with the following:

1. With the number of OPERABLE accident monitoring instrumentation channels less than the Total Number of Channels shown in Table 3.7-6, items 1 through 9, either restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

2. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum OPERABLE Channels requirement of Table 3.7-6, items 1 through 9, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

F ~~8.~~ The containment hydrogen analyzers and associated support equipment shall be OPERABLE in accordance with the following:

1. Two independent containment hydrogen analyzers shall be OPERABLE during REACTOR CRITICAL or POWER OPERATION.

   a. With one hydrogen analyzer inoperable, restore the inoperable analyzer to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 6 hours.

TABLE 3.7-1

REACTOR TRIP

INSTRUMENT OPERATING CONDITIONS

Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Action
1. Manual	2	2	1		1
2. Nuclear Flux Power Range	4	3	2	Low trip setting at P-10	2
3. Nuclear Flux Intermediate Range	2	2	1	P-10	3
4. Nuclear Flux Source Range				P-6	
a. Below P-6 - Note A	2	2	1		4
b. Shutdown - Note B	2	1	0		5
5. Overtemperature ΔT	3	2	2		6
6. Overpower ΔT	3	2	2		6
7. Low Pressurizer Pressure	3	2	2	P-7	7
8. Hi Pressurizer Pressure	3	2	2		~~7~~ 6

Note A - With the reactor trip breakers closed and the control rod drive system capable of rod withdrawal.

Note B - With the reactor trip breakers open.

TABLE 3.7-1

REACTOR TRIP

INSTRUMENT OPERATING CONDITIONS

	Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Action
9.	Pressurizer-Hi Water Level	3	2	2	P-7	~~6~~ 7
10.	Low Flow	3/loop	2/loop in each operating loop	2/loop in any operating loop	P-8	~~6~~ 7
				2/loop in any 2 operating loops	P-7	7
11.	Turbine Trip					
	a. Stop valve closure	4	1	4	P-7	~~12~~ 7
	b. Low fluid oil pressure	3	2	2	P-7	~~6~~ 7
12.	Lo-Lo Steam Generator Water Level	3/loop	2/loop in each operating loop	2/loop in any operating loops		~~7~~ 6
13.	Underfrequency 4KV Bus	3-1/bus	2	2	P-7	~~8~~ 7
14.	Undervoltage 4KV Bus	3-1/bus	2	2	P-7	7
15.	Safety Injection (SI) Input From ESF	2	2	1		~~8A~~ 11
16.	Reactor Coolant Pump Breaker Position	1/breaker	1/breaker per operating loop	1 2	P-8 P-7	9 ~~10~~ 9

Amendment Nos. ~~180 and 190~~

~~07-08-99~~

TABLE 3.7-1

REACTOR TRIP

INSTRUMENT OPERATING CONDITIONS

	Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Action
17.	Low steam generator water level with steam/feedwater flow mismatch	2/loop-level and 2/loop-flow mismatch	1/loop-level and 2/loop-flow mismatch or 2/loop-level and 1/loop-flow mismatch	1/loop-level coincident with 1/loop-flow mismatch in same loop		~~7~~ 6
18.	a. Reactor Trip Breakers	2	2	1		8
	b. Reactor Trip Bypass Breakers - Note C	2	1	1		
19.	Automatic Trip Logic	2	2	1		11
20.	Reactor Trip System Interlocks - Note D					
	a. Intermediate range neutron flux, P-6	2	2	1		13
	b. Low power reactor trips block, P-7					
	Power range neutron flux, P-10 and	4	3	2		13
	Turbine impulse pressure	2	2	1		13
	c. Power range neutron flux, P-8	4	3	2		13
	d. Power range neutron flux, P-10	4	3	2		13
	e. Turbine impulse pressure	2	2	1		13

Note C - With the Reactor Trip Breaker open for surveillance testing in accordance with Specification Table 4.1-1 (Item 30)
Note D - Reactor Trip System Interlocks are described in Table 4.1-A

TABLE 3.7-1 (Continued)

TABLE NOTATION

ACTION STATEMENTS

ACTION 1. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 6 hours ~~and/or~~ open the reactor trip breakers.

REACTOR CRITICAL and

ACTION 2.~~A~~ With the number of OPERABLE channels ~~equal to~~ the Minimum OPERABLE Channels requirement POWER OPERATION may proceed provided the following conditions are satisfied:

1. The inoperable channel is placed in the tripped condition within ~~6~~ 72 hours.

2. The Minimum OPERABLE Channels requirement is met; however, the inoperable channel may be bypassed for up to ~~4~~ 12 hours for surveillance testing of the redundant channel(s) per Specification 4.1.

3. Either, THERMAL POWER is restricted to ≤ 75% of RATED POWER and the Power Range, Neutron Flux trip setpoint is reduced to ≤ 85% of RATED POWER within ~~4~~ 78 hours; or, the QUADRANT POWER TILT is monitored at least once per 12 hours.

TABLE 3.7-1 (Continued)

4. The QUADRANT POWER TILT shall be determined to be within the limit when above 75 percent of RATED POWER with one Power Range Channel Inoperable by using the moveable incore detectors to confirm that the normalized symmetric power distribution, obtained from 2 sets of 4 symmetric thimble locations or a full-core flux map, is consistent with the indicated QUADRANT POWER TILT at least once per 12 hours.

~~2.B.~~ With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, be in at least HOT SHUTDOWN within 6 hours

ACTION 3. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement and with the THERMAL POWER level:

a. Below the P-6 (Block of Source Range Reactor Trip) setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint.

b. Above the P-6 (Block of Source Range Reactor Trip) setpoint, but below 10% of RATED POWER, decrease power below P-6 or restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED POWER, within 24 hours.

c. Above 10% of RATED POWER, POWER OPERATION may continue.

TABLE 3.7-1 (Continued)

ACTION 4. With the number of channels OPERABLE one less than required by the Minimum OPERABLE Channels requirement and with the THERMAL POWER level:

immediately suspend reactivity changes that are more positive than necessary to meet the required shutdown margin or refueling boron concentration limit and

a. Below P-6, (Block of Source Range Reactor Trip) setpoint, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour. Two Source Range channels must be OPERABLE prior to increasing THERMAL POWER above the P-6 setpoint.

with two Source Range channels inoperable, open the reactor trip breakers immediately

b. Above P-6, operation may continue.

ACTION 5. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, verify compliance with the Shutdown Margin requirements within 1 hour and at least once per 12 hours thereafter.

ACTION 6.~~A~~ With the number of OPERABLE channels ~~equal to the Minimum OPERABLE Channels requirement,~~ less than the Total Number of Channels REACTOR CRITICAL and POWER OPERATION may proceed provided the following conditions are satisfied:

1. The inoperable channel is placed in the tripped condition within ~~6~~ 72 hours.

2. The Minimum OPERABLE Channels requirement is met; however, the inoperable channel may be bypassed for up to ~~4~~ 12 hours for surveillance testing of other channels per Specification 4.1.

6.~~B~~. If the conditions are not satisfied in the time permitted, ~~With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement,~~ be ~~in~~ at least HOT SHUTDOWN within 6 hours.

TABLE 3.7-1 (Continued)

ACTION 7. With the number of OPERABLE channels less than the Total Number of Channels, ~~equal to the Minimum OPERABLE Channels~~, REACTOR CRITICAL and POWER OPERATION may proceed provided the following conditions are satisfied:

1. The inoperable channel is placed in the tripped condition within ~~6~~ 72 hours.

2. The Minimum OPERABLE Channels requirement is met; however, the inoperable channel may be bypassed for up to ~~4~~ 12 hours for surveillance testing per Specification 4.1.

#If the conditions are not satisfied in the time permitted, reduce power to less than the P-7 setpoint within the next 6 hours

ACTION 8.A. ~~With the number of OPERABLE channels~~ one less than the Minimum OPERABLE Channels requirement, be in at least HOT SHUTDOWN within 6 hours. In conditions of operation other than REACTOR CRITICAL or POWER OPERATIONS, with the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour. However, one channel may be bypassed for up to 2 hours for surveillance testing per ~~Specification 4.1~~, provided the other channel is OPERABLE, or one reactor trip breaker may be bypassed for up to 4 hours for concurrent surveillance testing of the reactor trip breaker and automatic trip logic provided the other train is OPERABLE.

8.B. With one of the diverse trip features (undervoltage or shunt trip device) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply Action 8.A. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

TABLE 3.7-1 (Continued)

ACTION 9. With one channel inoperable, 72 restore the inoperable channel to OPERABLE status within ~~8~~ hours or reduce THERMAL POWER to below the P-~~8~~ 9 (Block of Low Reactor Coolant Pump Flow and Reactor Coolant Pump Breaker Position) setpoint within the next ~~2~~ 6 hours. ~~Operation below P-8 may continue pursuant to ACTION 10.~~

ACTION 10. Deleted ~~With the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, operation may continue provided the inoperable channel is placed in the tripped condition within 6 hours.~~

ACTION 11. With the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within ~~8~~ 6 hours. In conditions of operation other than REACTOR CRITICAL or POWER OPERATIONS, with the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour. However, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.1 provided the other channel is OPERABLE.

ACTION 12. Deleted ~~With the number of OPERABLE channels less than the Total Number of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within 6 hours.~~

ACTION 13. With the number of OPERABLE channels less than the Minimum OPERABLE Channels requirement, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or be in at least HOT SHUTDOWN within the next 6 hours.

TABLE 3.7-2 (Continued)

ENGINEERED SAFEGUARDS ACTION
INSTRUMENT OPERATING CONDITIONS

Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Actions
2. CONTAINMENT SPRAY					
a. Manual	1 set	1 set	1 set*		15
b. High containment pressure (Hi-Hi)	4	3	3		17
c. Automatic actuation logic	2	2	1		14
3. AUXILIARY FEEDWATER					
a. Steam generator water level low-low					
1) Start motor driven pumps	3/steam generator	2/steam generator	2/steam generator any 1 generator		20
2) Starts turbine driven pump	3/steam generator	2/steam generator	2/steam generator any 2 generators		20
b. RCP undervoltage starts turbine driven pump	3	2	2		20
c. Safety injection - start motor driven pumps	See #1 above (all SI initiating functions and requirements)				
d. Station blackout - start motor driven pumps	1/bus 2 transfer buses/unit	1/bus 2 transfer buses/unit	2		

* Must actuate 2 switches simultaneously

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TABLE 3.7-2 (Continued)
ENGINEERED SAFEGUARDS ACTION
INSTRUMENT OPERATING CONDITIONS

Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Actions
3. AUXILIARY FEEDWATER (continued)					
e. Trip of main feedwater pumps - start motor driven pumps	2/MFW pump	1/MFW pump	2-1 each MFW pump		~~21~~ 24
f. Automatic actuation logic	2	2	1		22
4. LOSS OF POWER					
a. 4.16 kv emergency bus undervoltage (loss of voltage)	3/bus	2/bus	2/bus		~~20~~ 26
b. 4.16 kv emergency bus undervoltage (degraded voltage)	3/bus	2/bus	2/bus		~~20~~ 26
5. NON-ESSENTIAL SERVICE WATER ISOLATION					
a. Low intake canal level - Note A	4	3	3		20
b. Automatic actuation logic	2	2	1		14
6. ENGINEERED SAFEGAURDS ACTUATION INTERLOCKS - Note B					
a. Pressurizer pressure, P-11	3	2	2		23
b. Low-low T_{avg}, P-12	3	2	2		23
c. Reactor trip, P-4	2	2	1		24
7. RECIRCULATION MODE TRANSFER					
a. RWST Level - Low	4	3	2		25
b. Automatic Actuation Logic and Actuation Relays	2	2	1		14

Note A - When the temporary Service Water supply jumper to the CCHXs is in service in accordance with the footnote to TS 3.14.A.2.b, two low intake canal level probes will be permitted to be in the tripped condition. In this condition, two operable channels are required with one channel to trip. If one of the two operable channels becomes inoperable, the operating Unit must be in HOT SHUTDOWN within the following 6 hours and in COLD SHUTDOWN within the following 30 hours.

Note B - Engineered Safeguards Actuation Interlocks are described in Table 4.1-A

TABLE 3.7-3

INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Actions
1. CONTAINMENT ISOLATION					
a. Phase I					
1) Safety Injection (SI)	See Item #1, Table 3.7-2 (all SI initiating functions and requirements)				
2) Automatic initiation logic	2	2	1		14
3) Manual	2	2	1		21
b. Phase 2					
1) High containment pressure	4	3	3		17
2) Automatic actuation logic	2	2	1		14
3) Manual	2	2	1		~~15~~ 21
c. Phase 3					
1) High containment pressure (Hi-Hi setpoint)	4	3	3		17
2) Automatic actuation logic	2	2	1		14
3) Manual	1 set	1 set	1 set*		15
2. STEAMLINE ISOLATION					
a. High steam flow in 2/3 lines coincident with 2/3 low T_{avg} or 2/3 low steam pressures	See Item #1.e Table 3.7-2 for operability requirements				

* Must actuate 2 switches simultaneously

TABLE 3.7-3 (Continued)

INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Actions
STEAMLINE ISOLATION (continued)					
b. High containment pressure (Hi-Hi setpoint)	4	3	3		17
c. Manual	1/steamline	1/steamline	1/steamline		21
d. Automatic actuation logic	2	2	1		22
3. TURBINE TRIP AND FEEDWATER ISOLATION					
a. Steam generator water-level high-high	3/steam generator	2/steam generator	2/in any one steam generator	▽ When ALL MFIV, SGFWIV's ASSOCIATED BYPASS VALVES ARE CLOSED & DEACTIVATED or ISOLATED BY HAND VALVES	20
b. Automatic actuation logic and actuation relay	2	2	1		22
c. Safety injection	See item #1 of Table 3.7-2 (all SI initiating functions and requirements)				

TABLES 3.7-2 AND 3.7-3
TABLE NOTATIONS

restore the inoperable channel to OPERABLE status within 24 hours or

ACTION 14. With the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, be in at least HOT SHUTDOWN within the next ~~12~~ 6 hours and in COLD SHUTDOWN within the next 30 hours. One channel may be bypassed for up to 8 hours for surveillance testing per Specification 4.1, provided the other channel is OPERABLE.

ACTION 15. With the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 30 hours.

ACTION 16 - DELETED

REACTOR CRITICAL and POWER OPERATION

ACTION 17. With the number of OPERABLE ~~channels~~ one less than the Total Number of Channels, ~~operation~~ may proceed provided the inoperable channel is placed in the tripped condition within ~~6~~ 72 hours and the Minimum OPERABLE Channels requirement is met. One additional channel may be bypassed for up to ~~4~~ 12 hours for surveillance testing per Specification 4.1.

ACTION 18 DELETED

ACTION 19. Deleted ~~With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 6 hours.~~

ACTION 20. With the number of OPERABLE channels ~~one~~ less than the Total Number of Channels, REACTOR CRITICAL and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within ~~6~~ 72 hours.

b. The Minimum OPERABLE Channels requirement is met; however, the inoperable channel may be bypassed for up to ~~4~~ 12 hours for surveillance testing of other channels per Specification 4.1.

If the conditions are not satisfied in the time permitted, be in HOT SHUTDOWN within the next 6 hours and reduce RCS temperature & pressure to less than 350°F/450psig respectively in the following 12 hours

TABLES 3.7-2 ANDS 3.7-3 (Continued)
TABLE NOTATIONS

ACTION 21. With the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement~~s~~, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 22. With the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next ~~10~~ 6 hours and reduce pressure and temperature to less than 450 psig and 350° within the ~~next 8~~ following 12 hours; however, one channel may be bypassed for up to 8 hours for surveillance testing per Specification 4.1 provided the other channel is OPERABLE.

ACTION 23. With the number of OPERABLE channels less than the Minimum OPERABLE Channels requirement, within one hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or be in at least HOT SHUTDOWN within the next 6 hours.

ACTION 24. With the number of OPERABLE channels less than the Total Number of Channels, restore the inoperable channels to OPERABLE status within 48 hours or reduce pressure and temperature to less than 450 psig and 350°F within the next 12 hours.

ACTION 25. With the number of OPERABLE channels one less than the Total Number of Channels, place the inoperable channel in the bypassed condition within ~~8~~ 72 hours or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. One additional channel may be bypassed for up to ~~8~~ 12 hours for surveillance testing per Specification 4.1.

ACTION 26. With the number of OPERABLE channels less than the Total Number of Channels the associated Emergency Diesel Generator may be considered OPERABLE provided the following conditions are satisfied:
a. The inoperable channel is placed in the tripped condition within 72 hours,
b. The Minimum OPERABLE Channels requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels per Specification 4.1.
If the conditions are not satisfied, declare the associated EDG inoperable.

H. If the RWST Water Chemistry exceeds 0.15 PPM for Cl^- and/or F^-, flushing of sensitized stainless steel piping as required by 4.1.E will be performed once the RWST Water Chemistry has been brought within specification limit of less than 0.15 PPM chlorides and/or fluorides. Samples will be taken periodically until the sample indicates the Cl^- and/or F^- and levels are below 0.15 PPM.

BASIS

Check

Failures such as blown instrument fuses, defective indicators, and faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator action, and a periodic check supplements this type of built-in surveillance.

~~Based on experience in operation of both conventional and nuclear unit systems, when the unit is in operation, the minimum checking frequencies set forth are deemed adequate for reactor and steam system instrumentation.~~

Calibration

Calibration shall be performed to ensure the presentation and acquisition of accurate information.

The nuclear flux (power level) channels shall be calibrated daily against a heat balance standard to account for errors induced by changing rod patterns and core physics parameters.

Other channels are subject only to the "drift" errors induced within the instrumentation itself and, consequently, can tolerate longer intervals between calibration. Process systems instrumentation errors resulting from drift within the individual instruments are normally negligible.

During the interval between periodic channel tests and daily check of each channel, a comparison between redundant channels will reveal any abnormal condition resulting from a calibration shift, due to instrument drift of a single channel.

During the periodic channel test, if it is deemed necessary, the channel may be tuned to compensate for the calibration shift. However, it is not expected that this will be required at any fixed or frequent interval.

Thus, minimum calibration frequencies of once-per-day for the nuclear flux (power level) channels, and once per 18 months for the process system channels are considered acceptable.

<u>Testing</u>

~~The minimum testing frequency for those instrument channels connected to the safety system is based on an average unsafe failure rate of 2/5 x 10^{-6} failure/hr per channel. This is based on operating experience at conventional and nuclear units. An unsafe failure is defined as one which negates channel operability and which, due to its nature, is revealed only when the channel is tested or when it attempts to respond to a proper signal.~~

delete

SEE INSERT 'A'

Insert A - Replace the Testing Section of the bases with the following three paragraphs:

The OPERABILITY of the Reactor Trip System and ESFAS instrumentation systems and interlocks ensures that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic and sufficient redundancy are maintained to permit a channel to be out of service for testing or maintenance consistent with maintaining an appropriate level of reliability of the RTS and ESFAS instrumentation, and 3) sufficient system functional capability is available from diverse parameters.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. Specific surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, EVALUATION OF SURVEILLANCE FREQUENCIES AND OUT OF SERVICE TIMES FOR THE REACTOR TRIP INSTRUMENTATION SYSTEM, and supplements to that report, WCAP-10271 Supplement 2, EVALUATION OF SURVEILLANCE FREQUENCIES AND OUT OF SERVICE TIMES FOR THE ENGINEERED SAFETY FEATURES ACTUATION SYSTEM, and supplements to that report, and WCAP-14333P, PROBABILISTIC RISK ANALYSIS OF THE RPS AND ESF TEST TIMES AND COMPLETION TIMES, as approved by the NRC and documented in SERs dated February 21, 1985, February 22, 1989, the SSER dated April 30, 1990 for WCAP-10271 and July 15, 1998 for WCAP-14333P.

Surveillance testing of instrument channels is routinely performed with the channel in the tripped condition. Only those instrument channels with hardware permanently installed that permits bypassing without lifting a lead or installing a jumper are routinely tested in the bypass condition. However, an inoperable channel may be bypassed by lifting a lead or installing a jumper to permit surveillance testing of another instrument channel of the same functional unit.

delete For the specified one month test interval, the average unprotected time is 360 hrs in case of a failure occurring between test intervals. Thus, the probability of failure of one channel between test intervals is $360 \times 2.5 \times 10^{-6}$ or $.9 \times 10^{-3}$. Since two channels must fail in order to negate the safety function, the probability of simultaneous failure of two-out-of-three channels is $3(.9 \times 10^{-3})^2 = 2.4 \times 10^{-6}$. This represents the fraction of time in which each three-channel system would have one operable and two inoperable channels and equals $2.4 \times 10^{-6} \times 8760$ hours per year, or (approximately) 1 minute/year.

delete It must also be noted that to thoroughly and correctly test a channel, the channel components must be made to respond in the same manner and to the same type of input as they would be expected to respond to during their normal operation. This, of necessity, requires that during the test the channel be made inoperable for a short period of time. This factor must be, and has been, taken into consideration in determining testing frequencies.

delete Because of their greater degree of redundancy, the 2/4 logic arrays provide an even greater measure of protection and are thereby acceptable for the same testing interval. Those items specified for monthly testing are associated with process components where other means of verification provide additional assurance that the channel is operable, thereby requiring less frequent testing.

TABLE 4.1-1

MINIMUM FREQUENCIES FOR CHECK, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

Channel Description	Check	Calibrate	Test	Remarks
1. Nuclear Power Range	S	D(1,5) Q(3,5) R(4)	~~M~~(2) Q	1) Against a heat balance standard, above 15% RATED POWER 2) Signal at ΔT; bistable action (permissive, rod stop, trip) 3) Upper and lower chambers for symmetric offset by means of the movable incore detector system 4) Neutron detectors may be excluded from CHANNEL CALIBRATION 5) The provisions of Specification 4.0.4 are not applicable
2. Nuclear Intermediate Range (below P-10 setpoint)	*S	R(2,3)	P(1)	1) Log level; bistable action (permissive, rod stop, trip) 2) Neutron detectors may be excluded from CHANNEL CALIBRATION 3) The provisions of Specification 4.0.4 are not applicable
3. Nuclear Source Range (below P-6 setpoint)	*S	R(2,3)	P(1)	1) Bistable action (alarm, trip) 2) Neutron detectors may be excluded from CHANNEL CALIBRATION 3) The provisions of Specification 4.0.4 are not applicable
4. Reactor Coolant Temperature	*S	R	~~M(1)~~ Q(1) ~~M(2)~~ Q(2)	1) Overtemperature ΔT 2) Overpower ΔT
5. Reactor Coolant Flow	S	R	~~M~~ Q	
6. Pressurizer Water Level	S	R	~~M~~ Q	
7. Pressurizer Pressure (High & Low)	S	R	~~M~~ Q	
8. 4 KV Voltage and Frequency	N.A.	R	~~M~~ Q(1)	1) SETPOINT VERIFICATION NOT REQUIRED.
9. Analog Rod Position	*S(1,2) (4)	R	M(3)	1) With step counters 2) Each six inches of rod motion when data logger is out of service 3) Rod bottom bistable action 4) N.A. when reactor is in HOT, INTERMEDIATE OR COLD SHUTDOWN

TABLE 4.1-1 (Continued)

MINIMUM FREQUENCIES FOR CHECK, CALIBRATIONS, AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Test	Remarks
10.	Rod Position Bank Counters	S(1,2) Q(3)	N.A.	N.A.	1) Each six inches of rod motion when data logger is out of service 2) With analog rod position 3) For the control banks, the bench-board indicators shall be checked against the output of the bank overlap unit.
11.	Steam Generator Level	S	R	~~M~~ Q	
12.	Deleted ~~Charging Flow~~	~~N.A.~~	~~R~~	~~N.A.~~	
13.	Deleted ~~Residual Heat Removal Pump Flow~~	~~N.A.~~	~~R~~	~~N.A.~~	
14.	Deleted ~~Boric Acid Tank Level~~	~~*D~~	~~R~~	~~N.A.~~	
15.	Recirculation Mode Transfer				
	a. Refueling Water Storage Tank Level-Low	S	R	~~M~~ Q	
	b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	M	
16.	Deleted ~~Volume Control Tank Level~~	~~N.A.~~	~~R~~	~~N.A.~~	
17.	Reactor Containment Pressure-CLS	*D	R	~~M~~ Q(1)	1) Isolation valve signal and spray signal
18.	Deleted ~~Boric Acid Control~~	~~N.A.~~	~~R~~	~~N.A.~~	
19.	~~Item~~ Deleted				
20.	Deleted				
21.	Deleted ~~Containment Pressure-Vacuum Pump System~~	~~S~~	~~R~~	~~N.A.~~	
22.	Steam Line Pressure	S	R	~~M~~ Q	

TABLE 4.1-1 (Continued)

MINIMUM FREQUENCIES FOR CHECK, CALIBRATIONS, AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Test	Remarks
23.	Turbine First Stage Pressure	S	R	~~M~~ Q	
24.	Deleted ~~Emergency Plan Radiation Instr.~~	~~M~~	~~R~~	~~M~~	
25.	Deleted ~~Environmental Radiation Monitors~~	~~M~~	~~N.A.~~	~~N.A.~~	~~TLD Dosimeters~~
26.	Logic Channel Testing	N.A.	N.A.	M(1)(2)	1) Reactor protection, safety injection and the consequence limiting safeguards system logic channels are tested monthly per this line item. 2) The master and slave relays are not included in the monthly logic channel test of the safety injection system.
27.	Deleted ~~Turbine Overspeed Protection Trip Channel (Electrical)~~	~~N.A.~~	~~R~~	~~R~~	
28.	Turbine Trip				Setpoint verification is not applicable
	A. Stop valve closure	N.A.	N.A.	P	
	B. Low fluid oil pressure	N.A.	N.A.	P	
29.	Deleted ~~Seismic Instrumentation~~	~~M~~	~~R~~	~~M~~	
30.	Reactor Trip Breaker	N.A.	N.A.	M	The test shall independently verify operability of the undervoltage and shunt trip attachments
31.	Deleted ~~Reactor Coolant Pressure (Low)~~	~~N.A.~~	~~R~~	~~N.A.~~	

TABLE 4.1-1(Continued)

MINIMUM FREQUENCIES FOR CHECK, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Test	Remarks
32.	Auxiliary Feedwater				
	a. Steam Generator Water Level Low-Low	S	R	~~M(1)~~ Q	1) The auto start of the turbine driven pump is not included in the monthly test, but is tested within ~~30~~ 31 days prior to each startup.
	b. RCP Undervoltage	S	R	~~N.A.~~ R (1)(2)	1) The actuation logic and relays are tested within ~~30~~ 31 days prior to each startup. (2) setpoint verification not required
	c. S.I.	(All Safety Injection surveillance requirements)			
	d. Station Blackout	N.A.	R	N.A.	
	e. Main Feedwater Pump Trip	N.A.	N.A.	R	
33.	Loss of Power				
	a. 4.16 KV Emergency Bus Undervoltage (Loss of Voltage)	N.A.	R	~~M~~ Q(1)	1) setpoint verification not required.
	b. 4.16 KV Emergency Bus Undervoltage (Degraded Voltage)	N.A.	R	~~M~~ Q(1)	1) setpoint verification not required.
34.	Deleted				
35.	Manual Reactor Trip	N.A.	N.A.	R	The test shall independently verify the operability of the undervoltage and shunt trip attachments for the manual reactor trip function. The test shall also verify the operability of the bypass breaker trip circuit.
36.	Reactor Trip Bypass Breaker	N.A.	N.A.	M(1), R(2)	1) Remote manual undervoltage trip immediately after placing the bypass breaker into service, but prior to commencing reactor trip system testing or required maintenance. 2) Automatic undervoltage trip.
37.	Safety Injection Input to RPS	N.A.	N.A.	R	
38.	Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	R	

TABLE 4.1-1 (Continued)

MINIMUM FREQUENCIES FOR CHECK, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

Channel Description	Check	Calibrate	Test	Remarks
39. Steam/Feedwater Flow and Low S/G Water Level	S	R	~~M~~(1) Q	1) The provisions of Specification 4.0.4 are not applicable.
40. Intake Canal Low (See Footnote 1)	D	R	M(1), Q(2)	1) Logic Test 2) Channel Electronics Test
41. Turbine Trip and Feedwater Isolation				
a. Steam generator water level high	S	R	~~M~~ Q	
b. Automatic actuation logic and actuation relay	N.A.	R	M(1)	1) Automatic actuation logic only, actuation relays tested each refueling
42. Reactor Trip System Interlocks				
a. Intermediate range neutron flux, P-6	N.A.	R(1)	~~M(2,3)~~ R	1) Neutron detectors may be excluded from the calibration
b. Low reactor trips block, P-7	N.A.	R(1)	~~M(2,3)~~ R	~~(2) With power greater than or equal to the interlock setpoint, the required test shall consist of verifying that the interlock is in the required state by observing the permissive annunciator window~~
c. Power range neutron flux, P-8	N.A.	R(1)	~~M(2,3)~~ R	
d. Power range neutron flux, P-10	N.A.	R(1)	~~M(2,3)~~ R	
e. Turbine impulse pressure	N.A.	R	R	2 ~~3~~) The provisions of Specification 4.0.4 are not applicable.

Footnote 1:

Check Consists of verifying for an indicated intake canal level greater than 23'-6" that all four low level sensor channel alarms are not in an alarm state.

Calibration Consists of uncovering the level sensor and measuring the time response and voltage signals for the immersed and dry conditions. It also verifies the proper action of instrument channel from sensor to electronics to channel output relays and annunciator. Only the two available sensors on the shutdown unit would be tested.

Tests
1) The logic test verifies the three out of four logic development for each train by using the channel test switches for that train.
2) Channel electronics test verifies that electronics module responds properly to a superimposed differential millivolt signal

TABLE 4.1-1(Continued)
MINIMUM FREQUENCIES FOR CHECK, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

Channel Description	Check	Calibrate	Test	Remarks
43. Engineered Safeguards Actuation Interlocks				
a. Reactor trip, P-4	N.A.	N.A.	R	
b. Pressurizer pressure, P-11	N.A.	R	R	
c. Low, low T_{avg}, P-12	N.A.	R	R	

S - Each Shift

D - Daily

N.A. - Not Applicable

Q - Every 90 92 ~~effective full power~~ days

• See Specification 4.1.D

M - ~~Monthly~~ 31 days

P - Prior to each startup if not done within the previous ~~week~~ 31 days.

R - Once per 18 months

Amendment Nos. ~~213 and 213~~

TS 4.1-8c
~~06-11-98~~

Attachment 3

Proposed Unit 1 and Unit 2 Technical Specifications Changes

Surry Power Station
Units 1 and 2
Virginia Electric and Power Company

TECH SPEC CHANGE REQUEST NO. 301

TABULATION OF CHANGES

License Nos. DPR-32/37 / Docket Nos. 50-280/281

Summary of change:

This proposed change to the Technical Specifications is being made to reduce the unnecessary testing at power of the RPS/ESF instrumentation, which will reduce unnecessary challenges to the trip and safety systems, provide risk-informed relief in AOTs and bypass times for instrument channels.

DELETE	DATED	SUBSTITUTE
License Page 3	11-01-99	License Page 3
3.7-1	07-08-93	3.7-1
3.7-2	07-08-93	3.7-2
3.7-10	07-08-93	3.7-10
3.7-11	07-08-93	3.7-11
3.7-12	07-08-93	3.7-12
3.7-13	07-08-93	3.7-13
3.7-14	07-08-93	3.7-14
3.7-15	07-08-93	3.7-15
3.7-16	07-08-93	3.7-16
3.7-17	07-08-93	3.7-17
3.7-19	07-08-93	3.7-19
3.7-20	08-26-98	3.7-20
3.7-21	07-08-93	3.7-21
3.7-22	07-08-93	3.7-22
3.7-23	07-08-93	3.7-23
3.7-24	07-08-93	3.7-24
4.1-2	12-28-90	4.1-2
4.1-3	06-11-98	4.1-3
4.1-4	12-28-90	4.1-4
4.1-6	03-12-93	4.1-6
4.1-7	09-14-95	4.1-7
4.1-8	09-14-95	4.1-8
4.1-8a	03-12-99	4.1-8a
4.1-8b	03-12-93	4.1-8b
4.1-8c	06-11-98	4.1-8c

3.7 INSTRUMENTATION SYSTEMS

Operational Safety Instrumentation

Applicability

Applies to reactor and safety features instrumentation systems.

Objectives

To ensure the automatic initiation of the Reactor Protection System and the Engineered Safety Features in the event that a principal process variable limit is exceeded, and to define the limiting conditions for operation of the plant instrumentation and safety circuits necessary to ensure reactor and plant safety.

Specification

A. The Reactor Protection System instrumentation channels and interlocks shall be OPERABLE as specified in Table 3.7-1.

B. The Engineered Safeguards Actions and Isolation Function Instrumentation channels and interlocks shall be OPERABLE as specified in Tables 3.7-2 and 3.7-3, respectively.

C. The Engineered Safety Features initiation instrumentation setting limits shall be as stated in Table 3.7-4.

D. The explosive gas monitoring instrumentation channel shown in Table 3.7-5(a) shall be OPERABLE with its alarm setpoint set to ensure that the limits of Specification 3.11.A.1 are not exceeded.

   1. With an explosive gas monitoring instrumentation channel alarm setpoint less conservative than required by the above specification, declare the channel inoperable and take the action shown in Table 3.7-5(a).

2. With less than the minimum number of explosive gas monitoring instrumentation channels OPERABLE, take the action shown in Table 3.7-5(a). Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, prepare and submit a Special Report to the Commission (Region II) to explain why the inoperability was not corrected in a timely manner.

E. The accident monitoring instrumentation listed in Table 3.7-6 shall be OPERABLE in accordance with the following:

1. With the number of OPERABLE accident monitoring instrumentation channels less than the Total Number of Channels shown in Table 3.7-6, items 1 through 9, either restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

2. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum OPERABLE Channels requirement of Table 3.7-6, items 1 through 9, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

F. The containment hydrogen analyzers and associated support equipment shall be OPERABLE in accordance with the following:

1. Two independent containment hydrogen analyzers shall be OPERABLE during REACTOR CRITICAL or POWER OPERATION.

   a. With one hydrogen analyzer inoperable, restore the inoperable analyzer to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 6 hours.

TABLE 3.7-1
REACTOR TRIP
INSTRUMENT OPERATING CONDITIONS

Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Action
1. Manual	2	2	1		1
2. Nuclear Flux Power Range	4	3	2	Low trip setting at P-10	2
3. Nuclear Flux Intermediate Range	2	2	1	P-10	3
4. Nuclear Flux Source Range				P-6	
a. Below P-6 - Note A	2	2	1		4
b. Shutdown - Note B	2	1	0		5
5. Overtemperature ΔT	3	2	2		6
6. Overpower ΔT	3	2	2		6
7. Low Pressurizer Pressure	3	2	2	P-7	7
8. Hi Pressurizer Pressure	3	2	2		6

Note A - With the reactor trip breakers closed and the control rod drive system capable of rod withdrawal.

Note B - With the reactor trip breakers open.

TABLE 3.7-1
REACTOR TRIP
INSTRUMENT OPERATING CONDITIONS

Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Action
9. Pressurizer-Hi Water Level	3	2	2	P-7	7
10. Low Flow	3/loop	2/loop in each operating loop	2/loop in any operating loop	P-8	7
			2/loop in any 2 operating loops	P-7	7
11. Turbine Trip					
a. Stop valve closure	4	1	4	P-7	7
b. Low fluid oil pressure	3	2	2	P-7	7
12. Lo-Lo Steam Generator Water Level	3/loop	2/loop in each operating loop	2/loop in any operating loops		6
13. Underfrequency 4KV Bus	3-1/bus	2	2	P-7	7
14. Undervoltage 4KV Bus	3-1/bus	2	2	P-7	7
15. Safety Injection (SI) Input From ESF	2	2	1		11
16. Reactor Coolant Pump Breaker Position	1/breaker	1/breaker per operating loop	1 2	P-8 P-7	9 9

TABLE 3.7-1
REACTOR TRIP
INSTRUMENT OPERATING CONDITIONS

Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Action
17. Low steam generator water level with steam/feedwater flow mismatch	2/loop-level and 2/loop-flow mismatch	1/loop-level and 2/loop-flow mismatch or 2/loop-level and 1/loop-flow mismatch	1/loop-level coincident with 1/loop-flow mismatch in same loop		6
18. a. Reactor Trip Breakers	2	2	1		8
b. Reactor Trip Bypass Breakers - Note C	2	1	1		
19. Automatic Trip Logic	2	2	1		11
20. Reactor Trip System Interlocks - Note D					
a. Intermediate range neutron flux, P-6	2	2	1		13
b. Low power reactor trips block, P-7					
Power range neutron flux, P-10	4	3	2		13
and					
Turbine impulse pressure	2	2	1		13
c. Power range neutron flux, P-8	4	3	2		13
d. Power range neutron flux, P-10	4	3	2		13
e. Turbine impulse pressure	2	2	1		13

Note C - With the Reactor Trip Breaker open for surveillance testing in accordance with Specification Table 4.1-1 (Item 30)

Note D - Reactor Trip System Interlocks are described in Table 4.1-A

TABLE 3.7-1 (Continued)
TABLE NOTATION

ACTION STATEMENTS

ACTION 1. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN and open the reactor trip breakers within the next 6 hours.

ACTION 2. With the number of OPERABLE channels equal to the Minimum OPERABLE Channels requirement, REACTOR CRITICAL and POWER OPERATION may proceed provided the following conditions are satisfied:

1. The inoperable channel is placed in the tripped condition within 72 hours.
2. The Minimum OPERABLE Channels requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of the redundant channel(s) per Specification 4.1.
3. Either, THERMAL POWER is restricted to ≤ 75% of RATED POWER and the Power Range, Neutron Flux trip setpoint is reduced to ≤ 85% of RATED POWER within 78 hours; or, the QUADRANT POWER TILT is monitored at least once per 12 hours.

TABLE 3.7-1 (Continued)

4. The QUADRANT POWER TILT shall be determined to be within the limit when above 75 percent of RATED POWER with one Power Range Channel inoperable by using the moveable incore detectors to confirm that the normalized symmetric power distribution, obtained from 2 sets of 4 symmetric thimble locations or a full-core flux map, is consistent with the indicated QUADRANT POWER TILT at least once per 12 hours.

With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, be in at least HOT SHUTDOWN within 6 hours

ACTION 3. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement and with the THERMAL POWER level:

a. Below the P-6 (Block of Source Range Reactor Trip) setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint.

b. Above the P-6 (Block of Source Range Reactor Trip) setpoint, but below 10% of RATED POWER, decrease power below P-6 or, increase THERMAL POWER above 10% of RATED POWER within 24 hours.

c. Above 10% of RATED POWER, POWER OPERATION may continue.

TABLE 3.7-1 (Continued)

ACTION 4. With the number of channels OPERABLE one less than required by the Minimum OPERABLE Channels requirement and with the THERMAL POWER level:

a. Below P-6, (Block of Source Range Reactor Trip) setpoint, immediately suspend reactivity changes that are more positive than necessary to meet the required shutdown margin or refueling boron concentration limit and restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour. With two Source Range Channels inoperable, open the reactor trip breakers immediately. Two Source Range channels must be OPERABLE prior to increasing THERMAL POWER above the P-6 setpoint.

b. Above P-6, operation may continue.

ACTION 5. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, verify compliance with the Shutdown Margin requirements within 1 hour and at least once per 12 hours thereafter.

ACTION 6. With the number of OPERABLE channels less than the Total Number of Channels, REACTOR CRITICAL and POWER OPERATION may proceed provided the following conditions are satisfied:

1. The inoperable channel is placed in the tripped condition within 72 hours.

2. The Minimum OPERABLE Channels requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels per Specification 4.1.

If the conditions are not satisfied in the time permitted, be in at least HOT SHUTDOWN within 6 hours.

TABLE 3.7-1 (Continued)

ACTION 7. With the number of OPERABLE channels less than the Total Number of Channels, REACTOR CRITICAL and POWER OPERATION may proceed provided the following conditions are satisfied:

1. The inoperable channel is placed in the tripped condition within 72 hours.
2. The Minimum OPERABLE Channels requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing per Specification 4.1.

If the conditions are not satisfied in the time permitted, reduce power to less than the P-7 setpoint within the next 6 hours.

ACTION 8.A. With the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, be in at least HOT SHUTDOWN within 6 hours. In conditions of operation other than REACTOR CRITICAL or POWER OPERATIONS, with the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour. However, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.1 provided the other channel is OPERABLE, or one reactor trip breaker may be bypassed for up to 4 hours for concurrent surveillance testing of the Reactor trip breaker and automatic trip logic provided the other train is OPERABLE.

8.B. With one of the diverse trip features (undervoltage or shunt trip device) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply Action 8.A. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

TABLE 3.7-1 (Continued)

ACTION 9. With one channel inoperable, restore the inoperable channel to OPERABLE status within 72 hours or reduce THERMAL POWER to below the P-7 (Block of Low Reactor Coolant Pump Flow and Reactor Coolant Pump Breaker Position) setpoint within the next 6 hours.

ACTION 10. Deleted

ACTION 11. With the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within 6 hours. In conditions of operation other than REACTOR CRITICAL or POWER OPERATIONS, with the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour. However, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.1 provided the other channel is OPERABLE.

ACTION 12. Deleted

ACTION 13. With the number of OPERABLE channels less than the Minimum OPERABLE Channels requirement, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or be in at least HOT SHUTDOWN within the next 6 hours.

TABLE 3.7-2 (Continued)
ENGINEERED SAFEGUARDS ACTION
INSTRUMENT OPERATING CONDITIONS

Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Actions
2. CONTAINMENT SPRAY					
a. Manual	1 set	1 set	1 set♦		15
b. High containment pressure (Hi-Hi)	4	3	3		17
c. Automatic actuation logic	2	2	1		14
3. AUXILIARY FEEDWATER					
a. Steam generator water level low-low					
1) Start motor driven pumps	3/steam generator	2/steam generator	2/steam generator any 1 generator		20
2) Starts turbine driven pump	3/steam generator	2/steam generator	2/steam generator any 2 generators		20
b. RCP undervoltage starts turbine driven pump	3	2	2		20
c. Safety injection - start motor driven pumps	See #1 above (all SI initiating functions and requirements)				
d. Station blackout - start motor driven pumps	1/bus 2 transfer buses/unit	1/bus 2 transfer buses/unit	2		24

♦ Must actuate 2 switches simultaneously

TABLE 3.7-2 (Continued)
ENGINEERED SAFEGUARDS ACTION
INSTRUMENT OPERATING CONDITIONS

Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Actions
3. AUXILIARY FEEDWATER (continued)					
e. Trip of main feedwater pumps - start motor driven pumps	2/MFW pump	1/MFW pump	2-1 each MFW pump		24
f. Automatic actuation logic	2	2	1		22
4. LOSS OF POWER					
a. 4.16 kv emergency bus undervoltage (loss of voltage)	3/bus	2/bus	2/bus		26
b. 4.16 kv emergency bus undervoltage (degraded voltage)	3/bus	2/bus	2/bus		26
5. NON-ESSENTIAL SERVICE WATER ISOLATION					
a. Low intake canal level - Note A	4	3	3		20
b. Automatic actuation logic	2	2	1		14
6. ENGINEERED SAFEGAURDS ACTUATION INTERLOCKS - Note B					
a. Pressurizer pressure, P-11	3	2	2		23
b. Low-low T_{avg}, P-12	3	2	2		23
c. Reactor trip, P-4	2	2	1		24
7. RECIRCULATION MODE TRANSFER					
a. RWST Level - Low	4	3	2		25
b. Automatic Actuation Logic and Actuation Relays	2	2	1		14

Note A - When the temporary Service Water supply jumper to the CCHXs is in service in accordance with the footnote to TS 3.14.A.2.b, two low intake canal level probes will be permitted to be in the tripped condition. In this condition, two operable channels are required with one channel to trip. If one of the two operable channels becomes inoperable, the operating Unit must be in HOT SHUTDOWN within the following 6 hours and in COLD SHUTDOWN within the following 30 hours.

Note B - Engineered Safeguards Actuation Interlocks are described in Table 4.1-A

TABLE 3.7-3
INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Actions
1. CONTAINMENT ISOLATION					
a. Phase I					
1) Safety Injection (SI)	See Item #1, Table 3.7-2 (all SI initiating functions and requirements)				
2) Automatic initiation logic	2	2	1		14
3) Manual	2	2	1		21
b. Phase 2					
1) High containment pressure	4	3	3		17
2) Automatic actuation logic	2	2	1		14
3) Manual	2	2	1		21
c. Phase 3					
1) High containment pressure (Hi-Hi setpoint)	4	3	3		17
2) Automatic actuation logic	2	2	1		14
3) Manual	1 set	1 set	1 set♦		15
2. STEAMLINE ISOLATION					
a. High steam flow in 2/3 lines coincident with 2/3 low T_{avg} or 2/3 low steam pressures	See Item #1.e Table 3.7-2 for operability requirements				

♦ Must actuate 2 switches simultaneously

TABLE 3.7-3 (Continued)
INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Actions
STEAMLINE ISOLATION (continued)					
b. High containment pressure (Hi-Hi setpoint)	4	3	3		17
c. Manual	1/steamline	1/steamline	1/steamline		21
d. Automatic actuation logic	2	2	1		22
3. TURBINE TRIP AND FEEDWATER ISOLATION				When all MFRV, SG FWIV & associated bypass valves are closed & deactivated or isolated by manual valves.	
a. Steam generator water-level high-high	3/steam generator	2/steam generator	2/in any one steam generator		20
b. Automatic actuation logic and actuation relay	2	2	1		22
c. Safety injection	See Item #1 Table 3.7-2 (all SI initiating functions and requirements)				

TABLES 3.7-2 AND 3.7-3

TABLE NOTATIONS

ACTION 14. With the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the next 30 hours. One channel may be bypassed for up to 8 hours for surveillance testing per Specification 4.1, provided the other channel is OPERABLE.

ACTION 15. With the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 30 hours.

ACTION 16. Deleted

ACTION 17. With the number of OPERABLE channels one less than the Total Number of Channels, REACTOR CRITICAL and POWER OPERATION may proceed provided the inoperable channel is placed in the tripped condition within 72 hours and the Minimum OPERABLE Channels requirement is met. One additional channel may be bypassed for up to 12 hours for surveillance testing per Specification 4.1.

ACTION 18. Deleted

ACTION 19. Deleted

ACTION 20. With the number of OPERABLE channels less than the Total Number of Channels, REACTOR CRITICAL and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 72 hours.

b. The Minimum OPERABLE Channels requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels per Specification 4.1.

If the conditions are not satisfied in the time permitted, be in HOT SHUTDOWN within the next 6 hours and reduce RCS temperature & pressure to less than 350°F/450 psig, respectively in the following 12 hours.

TABLES 3.7-2 ANDS 3.7-3 (Continued)

TABLE NOTATIONS

ACTION 21. With the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 22. With the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 6 hours and reduce pressure and temperature to less than 450 psig and 350° within the following 12 hours; however, one channel may be bypassed for up to 8 hours for surveillance testing per Specification 4.1 provided the other channel is OPERABLE.

ACTION 23. With the number of OPERABLE channels less than the Minimum OPERABLE Channels requirement, within one hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or be in at least HOT SHUTDOWN within the next 6 hours.

ACTION 24. With the number of OPERABLE channels less than the Total Number of Channels, restore the inoperable channels to OPERABLE status within 48 hours or reduce pressure and temperature to less than 450 psig and 350°F within the next 12 hours.

ACTION 25. With the number of OPERABLE channels one less than the Total Number of Channels, place the inoperable channel in the bypassed condition within 72 hours or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. One additional channel may be bypassed for up to 12 hours for surveillance testing per Specification 4.1.

ACTION 26. With the number of OPERABLE channels less than the Total Number of Channels, the associated Emergency Diesel Generator may be considered OPERABLE provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped conditions within 72 hours.

b. The Minimum OPERABLE Channels requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels per Specification 4.1.

If the conditions are not satisfied, declare the associated EDG inoperable.

H. If the RWST Water Chemistry exceeds 0.15 PPM for Cl^- and/or F^-, flushing of sensitized stainless steel piping as required by 4.1.E will be performed once the RWST Water Chemistry has been brought within specification limit of less than 0.15 PPM chlorides and/or fluorides. Samples will be taken periodically until the sample indicates the Cl^- and/or F^- and levels are below 0.15 PPM.

BASIS

Check

Failures such as blown instrument fuses, defective indicators, and faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator action, and a periodic check supplements this type of built-in surveillance.

Calibration

Calibration shall be performed to ensure the presentation and acquisition of accurate information.

The nuclear flux (power level) channels shall be calibrated daily against a heat balance standard to account for errors induced by changing rod patterns and core physics parameters.

Other channels are subject only to the "drift" errors induced within the instrumentation itself and, consequently, can tolerate longer intervals between calibration. Process systems instrumentation errors resulting from drift within the individual instruments are normally negligible.

During the interval between periodic channel tests and daily check of each channel, a comparison between redundant channels will reveal any abnormal condition resulting from a calibration shift, due to instrument drift of a single channel.

During the periodic channel test, if it is deemed necessary, the channel may be tuned to compensate for the calibration shift. However, it is not expected that this will be required at any fixed or frequent interval.

Thus, minimum calibration frequencies of once-per-day for the nuclear flux (power level) channels, and once per 18 months for the process system channels are considered acceptable.

The OPERABILITY of the Reactor Trip System and ESFAS instrumentation systems and interlocks ensures that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic and sufficient redundancy are maintained to permit a channel to be out of service for testing or maintenance consistent with maintaining an appropriate level of reliability of the RTS and ESFAS instrumentation, and 3) sufficient system functional capability is available from diverse parameters.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. Specific surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, EVALUATION OF SURVEILLANCE FREQUENCIES AND OUT OF SERVICE TIMES FOR THE REACTOR TRIP INSTRUMENTATION SYSTEM, and supplements to that report, WCAP-10271 Supplement 2, EVALUATION OF SURVEILLANCE FREQUENCIES AND OUT OF SERVICE TIMES FOR THE ENGINEERED SAFETY FEATURES ACTUATION SYSTEM, and supplements to that report, and WCAP-14333P, PROBABILISTIC RISK ANALYSIS OF THE RPS AND ESF TEST TIMES AND COMPLETION TIMES, as approved by the NRC and documented in SERs dated February 21, 1985, February 22, 1989, the SSER dated April 30, 1990 for WCAP-10271 and July 15, 1998 for WCAP-14333P.

Surveillance testing of instrument channels is routinely performed with the channel in the tripped condition. Only those instrument channels with hardware permanently installed that permits bypassing without lifting a lead or installing a jumper are routinely tested in the bypass condition. However, an inoperable channel may be bypassed by lifting a lead or installing a jumper to permit surveillance testing of another instrument channel of the same functional unit.

TABLE 4.1-1

MINIMUM FREQUENCIES FOR CHECK, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Test	Remarks
1.	Nuclear Power Range	S	D(1,5) Q(3,5) R(4)	Q(2)	1) Against a heat balance standard, above 15% RATED POWER 2) Signal at ΔT; bistable action (permissive, rod stop, trip) 3) Upper and lower chambers for symmetric offset by means of the movable incore detector system 4) Neutron detectors may be excluded from CHANNEL CALIBRATION 5) The provisions of Specification 4.0.4 are not applicable
2.	Nuclear Intermediate Range (below P-10 setpoint)	*S	R(2,3)	P(1)	1) Log level; bistable action (permissive, rod stop, trip) 2) Neutron detectors may be excluded from CHANNEL CALIBRATION 3) The provisions of Specification 4.0.4 are not applicable
3.	Nuclear Source Range (below P-6 setpoint)	*S	R(2,3)	P(1)	1) Bistable action (alarm, trip) 2) Neutron detectors may be excluded from CHANNEL CALIBRATION 3) The provisions of Specification 4.0.4 are not applicable
4.	Reactor Coolant Temperature	*S	R	Q(1) Q(2)	1) Overtemperature ΔT 2) Overpower ΔT
5.	Reactor Coolant Flow	S	R	Q	
6.	Pressurizer Water Level	S	R	Q	
7.	Pressurizer Pressure (High & Low)	S	R	Q	
8.	4 KV Voltage and Frequency	N.A.	R	Q(1)	1) Setpoint verification not required
9.	Analog Rod Position	*S(1,2) (4)	R	M(3)	1) With step counters 2) Each six inches of rod motion when data logger is out of service 3) Rod bottom bistable action 4) N.A. when reactor is in HOT, INTERMEDIATE OR COLD SHUTDOWN

TABLE 4.1-1(Continued)
MINIMUM FREQUENCIES FOR CHECK, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Test	Remarks
10.	Rod Position Bank Counters	S(1,2) Q(3)	N.A.	N.A.	1) Each six inches of rod motion when data logger is out of service 2) With analog rod position 3) For the control banks, the benchboard indicators shall be checked against the output of the bank overlap unit.
11.	Steam Generator Level	S	R	Q	
12.	Deleted				
13.	Deleted				
14.	Deleted				
15.	Recirculation Mode Transfer				
	a. Refueling Water Storage Tank Level-Low	S	R	Q	
	b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	M	
16.	Deleted				
17.	Reactor Containment Pressure-CLS	*D	R	Q(1)	1) Isolation valve signal and spray signal
18.	Deleted				
19.	Deleted				
20.	Deleted				
21.	Deleted				
22.	Steam Line Pressure	S	R	Q	

TABLE 4.1-1(Continued)

MINIMUM FREQUENCIES FOR CHECK, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

Channel Description	Check	Calibrate	Test	Remarks
23. Turbine First Stage Pressure	S	R	Q	
24. Deleted				
25. Deleted				
26. Logic Channel Testing	N.A.	N.A.	M(1)(2)	1) Reactor protection, safety injection and the consequence limiting safeguards system logic are tested monthly per this line item. 2) The master and slave relays are not included in the monthly logic channel test of the safety injection system.
27. Deleted				
28. Turbine Trip				Setpoint verification is not applicable
A. Stop valve closure	N.A.	N.A.	P	
B. Low fluid oil pressure	N.A.	N.A.	P	
29. Deleted				
30. Reactor Trip Breaker	N.A.	N.A.	M	The test shall independently verify operability of the undervoltage and shunt trip attachments
31. Deleted				

TABLE 4.1-1(Continued)

MINIMUM FREQUENCIES FOR CHECK, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Test	Remarks
32.	Auxiliary Feedwater				
	a. Steam Generator Water Level Low-Low	S	R	Q(1)	1) The auto start of the turbine driven pump is not included in the monthly test, but is tested within 31 days prior to each startup.
	b. RCP Undervoltage	S	R	R(1)(2)	1) The actuation logic and relays are tested within 31 days prior to each startup. 2) Setpoint verification not required.
	c. S.I.	(All Safety Injection surveillance requirements)			
	d. Station Blackout	N.A.	R	N.A.	
	e. Main Feedwater Pump Trip	N.A.	N.A.	R	
33.	Loss of Power				
	a. 4.16 KV Emergency Bus Undervoltage (Loss of Voltage)	N.A.	R	Q(1)	1) Setpoint verification not required.
	b. 4.16 KV Emergency Bus Undervoltage (Degraded Voltage)	N.A.	R	Q(1)	1) Setpoint verification not required.
34.	Deleted				
35.	Manual Reactor Trip	N.A.	N.A.	R	The test shall independently verify the operability of the undervoltage and shunt trip attachments for the manual reactor trip function. The test shall also verify the operability of the bypass breaker trip circuit.
36.	Reactor Trip Bypass Breaker	N.A.	N.A.	M(1), R(2)	1) Remote manual undervoltage trip immediately after placing the bypass breaker into service, but prior to commencing reactor trip system testing or required maintenance. 2) Automatic undervoltage trip.
37.	Safety Injection Input to RPS	N.A.	N.A.	R	
38.	Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	R	

TABLE 4.1-1(Continued)

MINIMUM FREQUENCIES FOR CHECK, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Test	Remarks
39.	Steam/Feedwater Flow and Low S/G Water Level	S	R	Q(1)	1) The provisions of Specification 4.0.4 are not applicable
40.	Intake Canal Low (See Footnote 1)	D	R	M(1), Q(2)	1) Logic Test 2) Channel Electronics Test
41.	Turbine Trip and Feedwater Isolation				
	a. Steam generator water level high	S	R	Q	
	b. Automatic actuation logic and actuation relay	N.A.	R	M(1)	1) Automatic actuation logic only, actuation relays tested each refueling
42.	Reactor Trip System Interlocks				
	a. Intermediate range neutron flux, P-6	N.A.	R(1)	R(2)	1) Neutron detectors may be excluded from the calibration 2) The provisions of Specification 4.0.4 are not applicable.
	b. Low reactor trips block, P-7	N.A.	R(1)	R(2)	
	c. Power range neutron flux, P-8	N.A.	R(1)	R(2)	
	d. Power range neutron flux, P-10	N.A.	R(1)	R(2)	
	e. Turbine impulse pressure	N.A.	R	R	

Footnote 1:

Check — Consists of verifying for an indicated intake canal level greater than 23'-6" that all four low level sensor channel alarms are not in an alarm state.

Calibration — Consists of uncovering the level sensor and measuring the time response and voltage signals for the immersed and dry conditions. It also verifies the proper action of instrument channel from sensor to electronics to channel output relays and annunciator. Only the two available sensors on the shutdown unit would be tested.

Tests
1) The logic test verifies the three out of four logic development for each train by using the channel test switches for that train.
2) Channel electronics test verifies that electronics module responds properly to a superimposed differential millivolt signal which is equivalent to the sensor detecting a "dry" condition.

TABLE 4.1-1(Continued)

MINIMUM FREQUENCIES FOR CHECK, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

Channel Description	Check	Calibrate	Test	Remarks
43. Engineered Safeguards Actuation Interlocks				
a. Reactor trip, P-4	N.A.	N.A.	R	
b. Pressurizer pressure, P-11	N.A.	R	R	
c. Low, low T_{avg}, P-12	N.A.	R	R	

S - Each Shift

D - Daily

N.A. - Not Applicable

Q - Every 92 days

* See Specification 4.1.D

M - 31 days

P - Prior to each startup if not done within the previous 31 days

R - Once per 18 months

Attachment 4

Significant Hazards Consideration Determination

Surry Power Station
Units 1 and 2
Virginia Electric and Power Company

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Virginia Electric and Power Company has reviewed the requirements of 10 CFR 50.92 as they relate to the proposed Reactor Protection System (RPS) and Engineered Safety Features Actuation System (ESFAS) Technical Specification changes for the Surry Units 1 and 2 and determined that a significant hazards consideration is not involved. In support of this conclusion, the following evaluation is provided.

Criterion 1 - Operation of Surry Units 1 and 2 in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. The determination that the results of the proposed changes remain within acceptable criteria was established in the SER(s) prepared for WCAP-10271, WCAP-10271 Supplement 1, WCAP-10271 Supplement 2, WCAP-10271 Supplement 2, Revision 1 and WCAP-14333 issued by letters dated February 21, 1985, February 22, 1989, April 30, 1998, and July 15, 1998.

Implementation of the proposed changes is expected to result in an increase in total RPS and ESFAS yearly unavailability. The proposed changes have been shown to result in a small increase in the core damage frequency (CDF) due to the combined effects of increased RPS and ESFAS unavailability and reduced inadvertent reactor trips.

The values determined by the WOG and presented in the WCAP for the increase in CDF were verified by Brookhaven National Laboratory (BNL) as part of an audit and sensitivity analyses for the NRC Staff. Based on the small value of the increase compared to the range of uncertainty in the CDF, the increase is considered acceptable. The analysis performed by the WOG and presented in the WCAP included changes to the surveillance frequencies for the automatic actuation logic and actuation relays and the reactor trip and bypass breakers. The overall increase in the CDF, including the changes to the surveillance frequencies for the automatic actuation logic and actuation relays and the reactor trip and bypass breakers, was approximately 6 percent. However, even with this increase, the overall CDF remains lower than the NRC safety goal of 10 E-4/reactor year.

Changes to surveillance test frequencies for the RPS and ESFAS interlocks do not represent a significant reduction in testing. The currently specified test interval for interlock channels allows the surveillance requirement to be satisfied by verifying that the permissive logic is in its required state using the annunciator status light. The surveillance as currently required only verifies the status of the permissive logic and does not address verification of channel setpoint or operability. The setpoint verification and channel operability is verified after a refueling shutdown. The definition of the channel check includes comparison of the channel status with other channels for the same parameter. The requirement to routinely verify permissive status is a different consideration than the availability of trip or actuation channels which are required to change state on the occurrence of an event and for which the function availability is more dependent on the surveillance interval. Therefore, the change in the interlock surveillance requirement to at least once every 18 months does not represent a significant change in channel surveillance and does not involve a significant increase in unavailability of the RPS and ESFAS.

For the additional relaxations in WCAP-14333, the WOG evaluated the impact of the additional relaxation of allowed outage times and completion times, and action statements on core damage frequency. The change in core damage frequency is 3.1 percent for those plants with two out of three logic schemes that have not implemented the proposed surveillance test interval, allowed outage times, and completion times evaluated in WCAP-10271 and its supplements. This analysis calculates a significantly lower increase in core damage frequency than the WCAP-10271 analysis calculated. This can be attributed to more realistic maintenance intervals used in the current analysis and crediting the AMSAC system as an alternative method of initiating the auxiliary feedwater pumps. Therefore, the overall increase in CDF is estimated to be 3.1% for the proposed changes per the generic Westinghouse analysis.

The NRC performed an independent evaluation of the impact on core damage frequency (CDF) and large early release fraction (LERF). The results of the staff's review indicate that the increase in core damage frequency is small (approximately 3.2%) and the large early release fraction would increase by only 4 percent for 2 out of 3 logic schemes that have not implemented the proposed surveillance test interval, allowed outage times, and completion times evaluated in WCAP-10271 and its supplements. Further, the absolute values for CDF still remain within NRC safety goals.

Therefore, the proposed changes do not result in a significant increase in the severity or consequences of an accident previously evaluated. Implementation of the proposed changes affects the probability of failure of the RPS and ESFAS but does not alter the manner in which protection is afforded or the manner in which limiting criteria are established.

Criterion 2 - The proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes do not result in a change in the manner in which the RPS or ESFAS provide plant protection. No change is being made which alters the functioning of the RPS or ESFAS (other than in a test mode). Rather the likelihood or probability of the RPS or ESFAS functioning properly is affected as described above. Therefore, the proposed changes do not create the possibility of a new or different kind of accident as defined in the Safety Analysis Report.

The proposed changes do not involve hardware changes. Some existing instrumentation is designed to be tested in bypass and current Technical Specifications allow testing in bypass. Testing in bypass is also recognized by IEEE Standards. Therefore, testing in bypass has been previously approved and implementation of the proposed changes for testing in bypass does not create the possibility of a new or different kind of accident from any previously evaluated. Furthermore, since the other proposed changes do not alter the physical operation or functioning of the RPS or ESFAS, the possibility of a new or different kind of accident from any previously evaluated has not been created.

Criterion 3 - The proposed license amendment does not involve a significant reduction in a margin of safety.

The proposed changes do not alter the safety limits, limiting safety system setpoints or limiting conditions for operation. The RPS and ESFAS analog instrumentation remain operable to mitigate as assumed in the accident analysis. The impact of reduced testing other than as addressed above is to allow a longer time interval over which instrument uncertainties (e.g., drift) may act.

Implementation of the proposed changes is expected to result in an overall improvement in safety by less frequent testing of the RPS and ESFAS analog instruments and will result in less inadvertent reactor trips and actuation of Engineered Safety Features components.

This analysis demonstrates that the proposed amendment to the Surry Units 1 and 2 Technical Specifications does not involve a significant increase in the probability or consequences of a previously evaluated accident, does not create the possibility of a new or different kind of accident and does not involve a significant reduction in a margin of safety.