Tennessee Valley Authority, Post Office Box 2000, Decatur, Alabama 35609-2000

July 11, 2003

TS-405 10 CFR 50.12

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Washington, D.C. 20555-0001

Gentlemen:

In the Matter of	)	Docket Nos. 50-259
Tennessee Valley Authority	)	50-260
		50-296

BROWNS FERRY NUCLEAR PLANT (BFN) - UNITS 1, 2, AND 3 - EXEMPTION REQUEST FROM THE REQUIREMENTS OF 10 CFR 50 APPENDIX A GENERAL DESIGN CRITERIA (GDC)-41 IN SUPPORT OF TECHNICAL SPECIFICATIONS CHANGE (TS-405) - ALTERNATIVE SOURCE TERM (AST) (TAC NOS. MB5733, MB5734, MB5735)

Pursuant to 10 CFR 50.12(a)(2)(ii), the Tennessee Valley Authority (TVA) is submitting a request for an exemption from 10 CFR 50 Appendix A GDC-41. Specifically, TVA requests exemption from those portions of GDC-41 pertaining to single failure requirements as it relates to the use of the Standby Liquid Control (SLC) system for BFN's AST application.

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On July 31, 2002, TVA requested a license amendment and TS changes for a full scope application of AST methodology for BFN Units 1, 2, and 3. The AST analysis followed the guidance provided by Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." In the BFN AST loss-of-coolant accident (LOCA) analysis, as is typical for Boiling Water Reactor plants, operation of the SLC system is credited for limiting radiological consequences. In particular, the SLC system operation is relied upon to inject sodium pentaborate, which buffers the suppression pool and maintains pool pH greater than 7.0. This buffering action minimizes the re-evolution of iodine into the containment atmosphere, which in turn reduces the dose consequences. The use of the SLC system in this manner is not currently credited in the BFN design or licensing basis for reduction of the concentration of substances in the containment atmosphere following a design basis LOCA.

For AST analyses, per RG-1.183 Section 5.1.2, credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by TS, are powered by emergency power sources, and are either automatically actuated, or in limited cases, have actuation explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed.

Section 2.3.2 of Enclosure 4 of the July 31, 2002, submittal summarizes the attributes of SLC system as described in the BFN Updated Final Safety Analysis Report (UFSAR). The SLC system is categorized in the UFSAR as a special safety system and provides a backup method, independent of the control rods, to make the reactor subcritical. The SLC system is highly redundant, required to be operable by TS, powered by emergency power sources, and will have actuation requirements explicitly addressing the AST function in plant emergency procedures. Although the SLC system meets most of the requirements of RG-1.183, it is classified as a special safety system and does not strictly meet single-failure requirements in all areas.

In an April 23, 2003, meeting with NRC, TVA discussed the issues associated with the use of SLC as a special safety system to mitigate the radiological consequences of a design basis LOCA for AST. At the meeting, TVA proposed an exemption from GDC-41 since the SLC system does not strictly meet single failure requirements in all areas. Another meeting was held between TVA and NRC on June 10, 2003, to further discuss the need and regulatory basis for the exemption, as well as the proposed content of the exemption request. At that meeting, NRC also requested that Probability Safety Assessment (PSA) information on the SLC system be submitted.

Enclosure 1 provides the exemption request and the supporting technical justification for the exemption. The justification for the exemption concludes that the existing design and operational attributes of the SLC system meet the underlying purpose of GDC-41. The SLC system can be relied upon to control the fission products that may be released into the containment environment following a postulated design basis LOCA. As additional justification, TVA is also providing the results of an evaluation of the LOCA dose consequences assuming no SLC operation. The evaluation determined that the resulting dose consequences for offsite locations and the control room do not exceed the 10 CFR 50.67 limits. A summary of the dose evaluation is contained in Enclosure 2.

Enclosure 3 provides a PSA fault tree for the SLC system. This includes the quantification and the RISKMAN® files of common cause and basic event reports for the SLC system. Enclosure 4 provides a sketch showing the orientation of the SLC system containment isolation valves.

TVA has determined that the proposed exemption: 1) is authorized by law in that no law exists which precludes the activities covered by this exemption request; 2) will not present an undue risk to the public health and safety; and 3) is consistent with the common defense and security. Strict adherence to all of the requirements is not necessary for the SLC system to meet the underlying purpose of GDC-41 to control the fission products that may be released into reactor containment environment following a postulated LOCA.

Pursuant to 28 U. S. C. § 1746 (1994), I declare under penalty of perjury that the foregoing is true and correct. Executed on this 11th day of July, 2003.

If you have any questions about this request, please telephone me at (256) 729-2636.

Sincerely,

T. E. Abney
Manager of Licensing
and Industry Affairs

Enclosures:

1. 10 CFR 50.12 Exemption Request
2. Dose Evaluation With No Standby Liquid Control System Operation
3. Standby Liquid Control Probabilistic Safety Analysis (PSA) Fault Tree
4. Standby Liquid Control System Sketches

Enclosures
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ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

EXEMPTION REQUEST FROM THE REQUIREMENTS OF 10 CFR 50 APPENDIX A GENERAL DESIGN CRITERIA (GDC)-41 IN SUPPORT OF TECHNICAL SPECIFICATIONS CHANGE (TS-405) - ALTERNATIVE SOURCE TERM (AST)

10 CFR 50.12 EXEMPTION REQUEST

Background

On July 31, 2002, (Reference 1) TVA requested a license amendment and TS changes for a full scope application of AST methodology for BFN Units 1, 2, and 3. The AST analysis followed the guidance provided by Regulatory Guide (RG)-1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," (Reference 2). In the BFN AST loss-of-coolant accident (LOCA) analysis, as is typical for Boiling Water Reactor (BWR) plants, the Standby Liquid Control (SLC) system is credited for limiting radiological consequences. In particular, the SLC system operation is relied on to inject sodium pentaborate, which buffers the suppression pool and maintains pool pH greater than 7.0. This buffering action minimizes the re-evolution of iodine into the containment atmosphere, which in turn reduces the dose consequences.

The use of the SLC system is not currently credited in the BFN design or licensing basis for reduction of the concentration of substances in the containment atmosphere following a postulated design basis LOCA. For the AST application, SLC system operation is credited in limiting the radiological dose by minimizing the concentration of radioiodines in the containment atmosphere following a design basis LOCA involving fuel damage.

For AST analyses, per RG-1.183 Section 5.1.2, credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by TS, are powered by emergency power sources, and are either automatically actuated, or in limited cases, have actuation explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed.

Enclosure 4, Section 2.3.2 of the July 31, 2002, submittal summarizes the attributes of the SLC system as described in the BFN Updated Final Safety Analysis Report (UFSAR). The SLC system is categorized in the UFSAR as a special safety system, whose design functions include the mitigation of an Anticipated Transient Without Scram (ATWS) event per 10 CFR 50.62 and providing a backup method, independent of the control rods, to make the reactor subcritical. The SLC system is highly redundant, required to be operable by TS, powered by emergency power sources, and will have actuation requirements explicitly addressing the AST function in plant procedures. Although the SLC system meets most of the requirements of RG-1.183, it is classified as a special safety system and does not strictly meet single-failure requirements in all areas.

In an April 23, 2003, meeting with NRC, TVA discussed the issues associated with the use of SLC as a special safety system to mitigate the radiological consequences of a design basis LOCA. At the meeting, TVA proposed an exemption from GDC-41 since the SLC system does not strictly meet all single failure requirements. As indicated in the minutes for the April 23 meeting dated May 7, 2003, (Reference 3) an exemption from 10 CFR 50 Appendix A GDC - 41 for the single failure requirements would be requested by TVA.

Exemption Request

Pursuant to 10 CFR 50.12(a)(2)(ii), TVA is submitting this request for an exemption from 10 CFR 50 Appendix A GDC-41, Containment Atmospheric Clean-up. GDC-41 states:

> *Containment Atmospheric Clean-up.* Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with functioning of other associated systems, the concentration and quantity of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.
>
> Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

As described in UFSAR Section 3.8, the SLC system provides a backup method, which is independent of the control rods, to make the reactor subcritical over its full range of operating conditions including ATWS events. The AST LOCA analysis defines a new safety function for the SLC system in limiting the radiological consequences of a design basis LOCA involving fuel damage. In this respect, the SLC system would function in a similar manner to a Containment Atmospheric Clean-up system described in GDC-41 to control the fission products, which may be released to the containment environment following a postulated design basis LOCA by minimizing the re-evolution of iodine into the containment atmosphere.

The existing SLC system is not designated in the UFSAR as a safety-related system, but rather is categorized as a special safety system with regard to its function of mitigating an ATWS event. The SLC system design and operational basis meets most system attributes that a safety-related system would entail. For instance, the TS require the system to be operable and the system is routinely tested. Redundant components are provided (pumps and explosive valves), which ensures that single active component failures do not result in a loss of the system pumping capability. System electrical components are powered from emergency power sources. The system is not fully redundant since there is a single injection path to the vessel and a common control switch is used for system operation. As a result, while the system is highly redundant, the common elements of the system do not allow it to be considered strictly single failure proof.

Accordingly, TVA is requesting an exemption to GDC-41 as it pertains to the single failure requirements for the SLC system for AST applications. Granting this exemption would allow the use of a special safety system, SLC, to perform the new AST safety function of controlling the re-evolution of iodine, thus limiting the radiological consequences of a design basis LOCA with significant fuel damage.

Basis for the Exemption Request

10 CFR 50.12 authorizes the Commission, upon application by any interested person, to grant exemption from the requirements of the regulations when special circumstances are present. Such special circumstances are present in this instance to warrant exemption from the single failure requirement in 10 CFR 50 Appendix A GDC-41. Granting of this exemption would allow the use of the SLC system to limit the radiological consequences of a design basis LOCA described by RG-1.183. Specifically, Section (ii) of 10 CFR 50.12(a)(2) applies.

Justification Under 10 CFR 50.12(a)(2)(ii)

(ii) "Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule;"

The underlying purpose of GDC-41 is to ensure that Containment Atmospheric Clean-up systems function in a sound, reliable means to control fission products and other substances that may be released into the containment environment following postulated design basis accidents. The SLC system is highly redundant and reliable as supported by system design, physical configuration, system maintenance, and required

TS surveillance testing. Accordingly, allowing the use of the existing SLC system for control of suppression pool pH following a postulated design basis LOCA with fuel damage meets the underlying purpose of GDC-41 to provide a sound and reliable system for containment atmosphere clean-up.

__Technical Basis__

The SLC system is classified a special safety system which is defined in Section 1.2 of the BFN UFSAR as: A safety system the actions of which are essential to a safety action required in response to a special event. UFSAR Section 3.8 characterizes the SLC design function and UFSAR Figures 3.8-1 through 3.8-7 show system flow and logic diagrams. The objective of the SLC system is to provide a backup method, which is independent of the control rods, to make the reactor subcritical over its full range of operating conditions. The design and operational basis for the SLC system include requirements to inject sodium pentaborate into the reactor vessel to meet the requirements for an ATWS per 10 CFR 50.62.

Per GDC-41, a system used for containment atmosphere clean-up shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

Though the current SLC system does not meet all single failure requirements in all respects, it contains many redundant components and features. For instance, the system has redundant pumps and injection valves to ensure one subsystem is available for the injection function. Connections with the reactor pressure vessel are provided through components classified as safety-related. Leak detection is provided through the Local Leak Rate Testing (LLRT) program. Containment isolation capability is provided through two safety-related check valves and containment isolation capability is maintained assuming a single failure. System electrical components are powered from emergency power sources ensuring system operation should offsite power or onsite power not be available.

System Description and Operation

A separate SLC system is installed on each BFN unit. The system design is typical of that used for the BWR-4 line of reactors. Each system consists of a neutron absorber storage tank, two positive displacement injection pumps, two explosive injection valves, interconnecting piping, associated valves, and the instrumentation and controls necessary to inject neutron absorbing solution (sodium pentaborate) into the reactor vessel. The system injects the solution into the reactor vessel near the bottom of the lower core support plate.

Each positive displacement pump is sized to inject the contents of the storage tank into the reactor in 50 to 125 minutes (approximately 50 gallons per minute (GPM)). The AST analysis assumes the TS required flow rate of 39 GPM. The pumps are designed to pump to the reactor vessel at full reactor pressure under ATWS conditions and the pumps are capable of delivering flow at 1500 psig. System piping is arranged such that both injection pumps discharge into a common header to the vessel. A relief valve and accumulator for each pump is located upstream of the check valve between the pump and discharge valve. A check valve is provided in each injection pump discharge leg to prevent back flow through the non-running pump and relief valve. Parallel injection valves are off the common discharge header with subsequent piping joining to form a single reactor injection flow path through two series containment isolation check valves. For piping details, see the sketch in Enclosure 4.

The system piping is welded stainless steel, fabricated and installed in accordance with USAS B31.1- 1967 and specific portions of the American Society of Mechanical Engineers (ASME) Boiler Pressure Vessel Code.

The SLC System is required to be operable in the event of a station power failure. Therefore, the pumps, valves, and controls are powered from the standby AC power supplies. The pumps and valves are powered and controlled from separate buses and circuits so that a single power failure will not prevent system operation. Separate 250-V DC, battery backed, distribution panels powered from their respective 480-V shutdown board powers each pump control circuit and the injection valves. The injection valves are continuously monitored and alarm in the main control room if either circuit opens.

A five-position keylock switch, located in the main control room, manually initiates the SLC system. The five positions are START Pump A, Pump A NORMAL after START, OFF, Pump B NORMAL after START, and START Pump B. Switching to either initiation position (START Pump A or START Pump B) starts the corresponding SLC pump and activates both explosive valves. The two SLC pumps are interlocked to prevent simultaneous operation of both pumps. System operation indication is provided in the control room. If the selected pump fails to start, the redundant pump is activated manually using the same switch.

To mitigate ATWS, SLC must be initiated within a few minutes of the event initiation. In contrast, AST analysis credits SLC system initiation within two hours post-LOCA. As discussed in the July 31, 2002, license amendment request, TVA will revise plant emergency procedures to require the initiation of the SLC system based on indication of fuel failure (high radiation in the primary containment).

The new SLC function does not involve any change to the operator steps needed to initiate SLC injection. The timing requirements for operator response are considerably relaxed for AST compared to ATWS analysis requirements since the AST analysis assumes the system initiation is within two hours of the event.

SLC System Single Failure Considerations

<u>Common Discharge Line Containment Isolation Check Valves</u>

Redundant pumps are arranged in parallel taking suction from a sodium pentaborate solution tank. The pumps discharge through parallel injection valves into a common discharge line through containment and into the reactor pressure vessel. The discharge line contains two series containment isolation check valves as shown on Enclosure 4.

The containment isolation check valves are stainless steel Velan 1½-inch Bolted Bonnet Piston Check Valves (model W7234B13MS), mounted horizontally in the injection line. For an ATWS event, the containment isolation check valves are designed to open against full reactor pressure. For the AST function, the system operating requirements are reduced since the reactor pressure is much lower.

The containment isolation check valves are ASME Code Class 2 valves, and are subject to ASME Section XI In-service Testing and 10 CFR 50 Appendix J Local Leak Rate Testing (LLRT) Program. In accordance with these programs, these check valves are inspected and tested during scheduled refueling outages. TS require a system flow test to the vessel, which demonstrates the operability of the integrated system, at least once an operating cycle (24 months). A review of the maintenance history does not indicate any failures of these valves to open or close.

The SLC system pump discharge check valves are identical to the containment isolation check valves. These are exercised quarterly under the ASME Section XI program and are inspected on a regular basis. The inspections have not identified any indication of wear or other unusual degradation.

Nuclear Experience Review

NUREG/CR 5944 (9/93), "A Characterization of Check Valve Degradation and Failure Experience in the Nuclear Power Industry," documented a review and evaluation of check valve failures. The review found that the overall failure rate of the study for all check valves was 0.00996 per year. The failure rate of the check valves ≤ 2 inches was 0.00706 per year. In the distribution of failures, the restricted flow motion and failed closed modes were responsible for only 7 percent of the valve failures. In the ≤ 2 inches size group, the fraction of stuck closed failures was approximately 0.15 resulting in a failed closed failure rate of 0.001059 per year.

The Nuclear Industry Check Valve Group (NIC) established a centralized check valve failure/reliability database based in part on the existing Institute of Nuclear Power Operations Nuclear Plant Reliability Data System data. A sort of the NIC database for Velan lift/piston check valve failures was performed to identify any reported failures and the associated failure modes. Sixty-one instances of Velan 1½-inch lift/piston check valve failures were identified. Of the failures, only two failures were identified as stuck closed. Both stuck closed valves were carbon steel valves being operated in a wet steam environment. The SLC system containment isolation check valves are stainless steel valves. During the SLC system functional testing, demineralized water is pumped through the valves. Therefore, the operating environment for the failed valves is not consistent with the BFN SLC system.

A failure summary report from the Institute of Nuclear Power Operations Equipment Performance and Information Exchange (EPIX 4.0) database shows no instances of the same model valve as BFN's check valves (Velan W7234B13MS) to fail stuck closed. Additionally, an industry experience review performed for the BFN condition-monitoring program did not identify any failures of this type valve to open. Based on our review, there have been no stuck closed failures identified in the nuclear operating history for the manufacturer and model lift piston check valves used in the BFN SLC system.

In summary, industry data indicates check valves 2 inches and less are very reliable. Further, Velan 1½-inch check valves have experienced only 2 stuck closed failures. The model check valves in the BFN SLC system have not experienced a stuck closed failure. Based on this operating experience, the stuck closed failure of these valves in the common discharge line is highly unlikely. Therefore, the underlying purpose of GDC-41 is met by providing a highly reliable means of controlling fission products with the current design.

Common Start Switch

The SLC system is actuated by a five-position switch located in the main control room. The switch is a General Electric (GE) type SB-1 nine-stage rotary cam-operated switch, and is used in both safety and non-safety related applications at BFN. The nine individual stages are stacked onto a common shaft and mechanically tied together with two bolts threaded into the front support. Each stage has two contacts. The entire contact assembly is enclosed in a metal cover that provides physical protection for the switch contacts. This switch is used throughout the industry, and is of simple construction with few parts vulnerable to failure. The typical mechanical service life for this switch is estimated to be approximately one million cycles.

A review of EPIX exchange database identified three GE SB-1 switch failures in the industry. Contact corrosion attributed to two of the failures. The remaining failure was a result of sticking or binding. These failed switches were not in a main control room environment. The BFN switches are located in a temperature and humidity controlled environment in the main control room and not subject to contact corrosion.

A highly unlikely catastrophic failure of the switch must occur to result in a failure of both SLC subsystems to start. However, should the switch fail to initiate either SLC subsystem, the pH would remain greater than 7.0 for several hours and both time and main control room location would allow access for troubleshooting and repairs.

Previous SLC system functional testing has not identified any problems with the switch that would preclude SLC system injection. Based on the above, any failure that would prevent the start of at least one pump is highly improbable and TVA expects the SLC control switch to operate when required.

SLC System Single Failure Considerations Summary

Although the SLC system design does not strictly meet single failure requirements, the degree of redundancy and the reliability of the equipment necessary for the injection function provide adequate confidence that the system will operate, if needed. Hence, the current SLC system configuration fully meets the intent of GDC-41 in providing a highly sound and reliable system for containment atmospheric clean-up.

Maintenance Rule Program

The SLC system is within the scope of the BFN 10 CFR 50.65 Maintenance Rule program. The BFN Probabilistic Safety Analysis (PSA) establishes the system performance criteria, balancing unavailability and reliability for risk significant critical structures, systems, and components. TVA's Maintenance Rule program requires that each SLC pump flow path (subsystem) shall maintain an unavailability factor less than or equal to 1.347E-02 on a rolling 24-month interval. The observed unavailability factors for the SLC system are considerably less than the required unavailability factor.

The Maintenance Rule program monitors and trends all SLC system unavailability that could affect the flow path availability. Both A and B pumps and the containment isolation check valves are included in the program requirements. The total recorded unplanned unavailability for the flow paths is as follows:

	Pump 2A[1]	Pump 2B[1]	Pump 3A[2]	Pump 3B[2]
Total Unplanned Unavailable Hours	12.9	0.0	9.1	1.4

1) Since June 1994
2) Since November 1995

Unit 1 is in long-term layup and no recent SLC system data is available. The Unit 1 system is identical to Units 2 and 3. Therefore, TVA anticipates that the Unit 1 SLC performance will be consistent with Units 2 and 3.

The inclusion of the SLC system into the 10 CFR 50.65 Maintenance Rule and the confirmation of acceptable performance provides a continued assurance of the availability for performance of the AST function.

Quality Assurance Program

Equipment that provides the 10 CFR 50.62 ATWS mitigation function is a special safety system and, as such, is required by the Quality Assurance Program to meet quality related standards subject to the requirements of the Augmented Quality Program. The TVA Nuclear Quality Assurance Plan defines "quality related" as:

> "...a term which encompasses quality assurance program requirements that describe activities which affect structures, systems, and components. These requirements provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. In addition to safety related structures, systems, components, and activities, the term "quality related" encompasses the broad class of plant features covered (not necessarily explicitly) in the General Design Criteria of 10 CFR 50, Appendix A, that contribute in an important way to the safe operation and protection of the public in all phases and aspects of facility operation (i.e., normal operation and transient control as well as accident mitigation)."

Although the controls of the Augmented Quality Program are not as stringent as those for equipment classified as safety-related, the significance of the "quality related" classification is that it provides for a consistent means to control quality features, including matters such as procurement of replacement parts and control of maintenance activities. The SLC system parts are procured like for like, by part number and description. The procured part is traceable to the requisition contract and traceability is maintained for the life of the plant. Receipt inspection ensures that the part received is the part requested, meets the procurement requirements (part number and description), verifies no shipping damage, establishes a shelf life, and establishes any special preventative maintenance activities for storage of the part prior to installation in the plant. System maintenance activities are second party verified by an individual qualified to perform the task.

The rigorous controls imposed by the Quality Assurance Program provide adequate quality control elements to ensure SLC system component reliability for the required special safety function under the ATWS rule and for the performance of the AST function.

Technical Specifications Testing and Surveillance Requirements

TS Limiting Condition for Operation 3.1.7 requires that two SLC subsystems, (pumps, controls, and explosive valves) be operable when the reactor is in Modes 1 and 2. With one subsystem inoperable, 7 days are allowed to return it to operable status. With two subsystems inoperable, 8 hours are allowed to return one subsystem to operable status or be in Mode 3 within 12 hours. The proposed AST TS-405 will also require that two SLC subsystems be operable in Mode 3.

The TS require periodic verification and testing of the SLC system and component performance. These surveillances include: verify available volume of sodium pentaborate daily; verify continuity of injection valve explosive valve charge monthly; and verify proper concentration of Boron-10 of the sodium pentaborate storage tank monthly. A quarterly pump operability test verifies the ability of SLC injection pumps to meet the requirements of the BFN In-service Test Program.

Once every operating cycle (24 months), system functional testing verifies one subsystem's pump discharge relief valve setpoint, the pumping capacity, and the ability to inject into the reactor vessel. The functional test alternates each subsystem being tested. During the functional test, operation of the control circuits, indicators, and the alarm annunciator operation are verified. Once the surveillance is completed, system valve realignment returns the system to standby configuration, and the system realignment is independently verified. LLRT performed each refueling outage verifies containment valve isolation capability.

The SLC pump relief valves are tested periodically in accordance with American Society of Mechanical Engineers OM Code, 1995 Edition through 1996 addenda, Appendix I, and NUREG-1482, "Guidance to In-service Test Programs as Nuclear Power Plants."

The required TS and Section XI testing provide assurance of a high degree of system reliability and confidence that the system injection function would perform satisfactorily if called upon following a design basis LOCA.

SLC System Availability Probability

The BFN PSA models SLC system availability, and considers system testing and maintenance requirements. Common cause failure modes are considered for the active components of the system including failure of the pumps to start/run and failure of the explosive valves to actuate. The current PSA models the SLC system check valves with a failure to open (demand failure-failed closed) as a probability of 2.69E-04. The valve failure probability was obtained from the Pickard, Lowe, and Garrick database (PLG-0500, Volume 2 R0, "Database for Probabilistic Risk Assessment of Light Water Nuclear Power Plants").

The overall failure probability of the SLC system is 5.21E-03. This value represents a reliable system commensurate with the importance of the system function for ATWS mitigation and as proposed for AST. Enclosure 3 provides the SLC system PSA Fault Trees.

Dose Evaluation with No SLC Operation

An accident source term is intended to represent a major accident involving significant fuel damage postulated in conjunction with a large break LOCA. As required by 10 CFR 50.46 and 10 CFR 50 Appendix K, emergency core cooling system (ECCS) performance analyses demonstrate that fuel is not damaged during any postulated LOCA assuming single failures.

BFN LOCA analyses likewise shows that the 10 CFR 50.46 fuel limits are not exceeded during a spectrum of postulated breaks and single failures. For LOCA radiological consequences (AST and non-AST), guidance on methods of analysis is not provided in the regulation (10 CFR 50), but are in RG-1.3 (Reference 4), RG-1.183, Standard Review Plan (SRP)-15.6.5 (Reference 5), and SRP-15.0.1 (Reference 6). Pre-AST guidance in RG-1.3 and SRP-15.6.5 does not include discussions for consideration of single failures (except for the main steam isolation valve leakage control system as provided in the SRP) in the determination of radiological consequences. Notwithstanding this lack of guidance for single failure consideration, current radiological dose calculations for BFN include consideration of systems that are designed with redundancy.

Although ECCS performance analyses are performed to ensure that core damage will not occur for any postulated LOCA, the radiological dose analyses are performed to assess offsite and control room operator doses based on assumed significant fuel damage with subsequent release of appreciable quantities of fission products. These radiological design basis accident analyses were not intended to be actual event sequences, but rather were intended to be surrogates to enable deterministic evaluation of the response of a facility.

Radiological accident analyses are intentionally conservative in order to compensate for known uncertainties in accident progression, fission product transport, and atmospheric dispersion.

In assessing the significance of a single active failure of the SLC system to the AST LOCA radiological analysis, the following factors are considered:

To reach the point of significant fuel damage, multiple failures in redundant safety related equipment would have to occur. Current analyses for ECCS performance evaluate the most limiting single failures to ensure that fuel damage does not occur. Multiple and redundant safety systems are provided to ensure that adequate core cooling can be maintained for all postulated design basis accidents. Plant emergency operating procedures (including the severe accident guidelines) provide the operator with prescribed plans that consider multiple losses to ECCS systems and still maintain core cooling functions. The underlying purpose of the single failure criterion is enforced redundancy to assure the availability of engineered safeguard system capability for protection of the barriers to radiological release. In design basis LOCA radiological analysis, the consideration of single failure is the worst case event which can be postulated to result in significant core damage. Therefore, LOCA accident sequences that would lead to substantial fuel failure would inherently include multiple failures affecting redundant safety-related components.

The AST radiological dose analysis is conservatively performed to account for uncertainties in the analysis. This includes conservatisms in the calculations of suppression pool pH following the design basis LOCA. The likelihood of an additional failure in the SLC system that prevents boron injection following multiple failures leading to substantial fuel failure is extremely remote.

However, in support of this exemption request, TVA initiated a defense-in-depth evaluation to assess the dose consequences in the unlikely event that the SLC system fails to operate. The evaluation performed is a bounding evaluation that determined the contribution that SLC injection plays in the BFN radiological consequences analyses. The evaluation concluded that even if SLC failed, the control room and offsite doses would not exceed the regulatory limits of 10 CFR 50.67. A summary of the dose evaluation is in Enclosure 2.

Summary

In the BFN AST analysis, the SLC system is credited for limiting the radiological consequences following a design basis LOCA involving significant fuel damage. The use of the SLC system to provide a buffering solution for the suppression pool following the postulated design basis LOCA credits SLC injection as a new function. The adequacy of the SLC system to perform this new function is provided by the system design and physical configuration. High system reliability is supported by TS surveillance testing, the In-service Test Program, inclusion in the Maintenance Rule Program, and the requirements of the BFN Quality Assurance Program. Although the SLC system is not safety related and does not strictly meet all single failure requirements, the SLC system is suitable for the AST function.

SLC system initiation (and injection) for AST will continue to be a manual operator action requiring the same operator steps necessary for an ATWS event except that the timing requirements are relaxed. In the AST analyses, the SLC system initiation is based on plant alarm response to conditions indicating postulated fuel damage (high containment radiation). The initiation conditions are consistent with the conditions expected within the primary containment following the release of fission products.

The PSA values assigned to SLC indicate the system is highly reliable. The system is very simple to operate. One switch in the main control room controls the SLC system operation. Redundant pumps and explosive valves ensure that at least one subsystem of SLC will operate when required. Operating and control power for the system is backed by emergency power sources. Review of the single failure aspects of this system determined the portions of the SLC system that do not meet single failure requirements are highly reliable and are unlikely to fail. The single line into containment and the reactor vessel contain two series check valves. TVA's review of industry operating experience for these valves determined that they are very reliable. TVA's review of industry experience for the GE SB-1 main control room switch indicates it is very reliable.

Additionally, TVA has evaluated the consequences of SLC failing to control pH in the suppression pool and determined the resulting dose consequences would still remain within regulatory limits.

Conclusion

10 CFR 50.12 authorizes the Commission, upon application by any interested person, to grant exemption from the requirements of the regulations when special circumstances are present. TVA believes that such circumstances are present in this instance to warrant exemption from the regulatory requirements of 10 CFR 50 Appendix A GDC-41 pertaining to single failure provisions. Granting this exemption will allow the use the SLC system to limit the radiological consequences of a design basis LOCA. Specifically, Section (ii) of 10 CFR 50.12(a)(2) applies.

The proposed exemption 1) is authorized by law in that no law exists which precludes the activities covered by this exemption request, 2) will not present an undue risk to the public health and safety, and 3) is consistent with the common defense and security. The strict adherence to all of the redundancy provisions of GDC-41 is not necessary for the SLC system to meet the underlying purpose and intent of GDC-41 in specifying system attributes for controlling the fission products that may be released in to reactor containment environment following a postulated LOCA. A full application of the GDC-41 single failure requirements in this instance would not serve the underlying purpose of the provision.

References:

1. TVA letter to NRC dated July 31, 2002, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - License Amendment - Alternative Source Term."

2. NRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000.

3. NRC letter dated May 7, 2003, from Kahtan N. Jabbour, Senior Project Manager, Section 2, to Allen G. Howe, Chief, Section 2, "Summary of the April 23, 2003, Drop-In Meeting With Tim Abney Regarding the Alternative Source Term For Browns Ferry Nuclear Plant, Units 1, 2, and 3."

4. NRC Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," dated June 1974.

5. NRC Standard Review Plan 15.6.5, "Loss-of-Coolant Accidents Resulting From a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Boundary," dated July 1981.

6. NRC Standard Review Plan 15.0.1, "Radiological Consequence Analysis Using Alternative Source Term," dated July 2000.

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

EXEMPTION REQUEST FROM THE REQUIREMENTS OF 10 CFR 50 APPENDIX A GENERAL DESIGN CRITERIA (GDC)-41 IN SUPPORT OF TECHNICAL SPECIFICATIONS CHANGE (TS-405) - ALTERNATIVE SOURCE TERM (AST)

DOSE EVALUATION WITH NO STANDBY LIQUID CONTROL (SLC) OPERATION

As discussed in Enclosure 1, for the LOCA dose consequences analyses in AST applications, BWRs credit the operation of the SLC system, which buffers the suppression pool. This buffering action minimizes the re-evolution of radiolytic iodine species and reduces dose consequences since less iodine is available for leakage out of containment.

In support of the GDC-41 exemption request, TVA performed a LOCA dose consequences evaluation assuming no SLC operation. This evaluation included the dose consequences for the Exclusion Area Boundary (EAB), the Low Population Zone (LPZ), and the Control Room (CR). A summary of the evaluation methodology and results is provided below. As would be expected, the estimated dose consequences increase. However, the dose estimates remain below regulatory limits even with no credit being taken for SLC operation.

In BWR AST analysis, radioiodines released from the fuel are initially removed from radiological consideration due to sedimentation in the suppression pool. This radioiodine dissolves in the pool water and will remain there as long as the pH of the water is maintained greater than or equal to 7.0.

The primary effect of SLC operation is minimization of the re-evolution of radioiodine removed initially by sedimentation in the suppression pool. If radioiodine re-evolution is assumed to be instantaneous and complete, then the containment airborne radioiodine inventory would closely resemble that contained in the current licensing basis analysis LOCA dose methodology for BFN under TID-14844 and RG-1.3. Therefore, the re-evaluation used the dose results of the current licensing basis TID-14844 (Reference 1) and RG-1.3 (Reference 2) source term as a starting point.

The radioiodine inventory released from the suppression pool following AST methodology per RG-1.183 (Reference 3) assumptions differs from the TID-14844 inventory in two main areas as follows:

- Under AST, the iodine source term is a delayed release and suppression pool acidification would also occur gradually over hours and perhaps even days depending on the chemical form of the cesium that is released from the core along with the iodine.
- Under AST, a larger inventory of iodine would be available (for airborne release) for a given power level. This greater "per MWt" inventory is primarily the result of the 30 percent of the core inventory assumed to be airborne in RG-1.183 versus 25 percent assumed to be airborne in RG-1.3. Also, RG-1.183 requires consideration of the dose consequences of other radionuclides including Tellurium-131m (parent of Iodine-131) and Tellurium-132 (parent of Iodine-132). "Per MWt" noble gas inventories are also affected, but to a much lesser degree.

An instantaneous release of iodines is a conservative assumption. If dose multipliers are determined and applied to the TID-14844 and RG-1.3 (hereafter referred to TID doses) doses for other effects, then the thyroid and whole body doses for an amount of radioiodine (and noble gas) that represents an entire AST (RG-1.183) release being airborne can be reasonably determined by permuting the TID dose terms as described below.

<u>Correction Applied to the Control Room and Offsite TID Doses to Remove Standby Gas Treatment (SGT) System Filtration</u>

The top of stack and stack base plume pathway TID doses include the effects of SGT system filtration; however, the AST analysis does not. Hence, a dose correction is needed to account for the absence of filter credit.

The TID thyroid doses were multiplied by a factor of ten to reflect no credit being taken for SGT system filtration (SGT system filtration efficiency is 90 percent). Then, the fraction of the whole body dose that is due to radioiodine was estimated by comparing the products of the core inventories available to become airborne and the whole body Dose Conversion Factors (DCFs). The whole body dose was increased by computing a net adjustment for the contributions of the filterable radioiodine. Using these adjustments, the dose results were increased for thyroid and whole body doses for the top of stack and stack base plume pathways. These were added to the MSIV leakage plume pathway dose to obtain total TID doses with no removal of iodine by SGT system adsorbers.

<u>Correction Applied to the Control Room Dose to Remove Control Room Emergency Ventilation (CREV) System Filtration</u>

The TID control room doses include the effects of CREV system filtration however, the AST analysis does not. As a result, a dose correction is needed to account for the absence of CREV system filter credit.

To account for an increased dose affect due to the absence of CREV system filtration credit, an adjustment factor of filtered versus unfiltered flow was applied. For TID, the control room dose is the a function of radioactivity entering the control room via an effective unfiltered 4017 cfm of makeup flow (3717 cfm assumed unfiltered inleakage plus 10 percent of the filtered CREV system flow of 3000 cfm). If credit is not taken for the 90 percent CREV system filtration of the 3000 cfm makeup flow, then activity entering the control room increases by a factor of 1.67 (6717 cfm/4017 cfm). This multiplier was applied to the control room thyroid dose as well as to the control room whole body dose contribution from noble gases.

Definition of the Extended Power Uprate (EPU) Power Level Factor

The licensing basis analysis TID doses, are based on a power level of 3458 MWt. To adjust these doses to the Extended Power Uprate power level of 3952 MWt, a power level multiplier of 1.14 was applied to the calculated doses. This factor is a ratio of the respective power levels.

Definition of the Per MWt Noble Gas and Radioiodine Airborne Inventory Factor

As noted previously, treatment of the "per MWt" noble gas and radioiodine airborne isotope inventory is treated differently under TID methodology than AST. Adjustments to the TID terms were determined and applied as follows:

The base inventories of noble gas and radioiodine isotopes were obtained for the TID dose analysis and for the AST dose analysis. The radioiodine inventories were then multiplied by appropriate airborne release fractions (0.25 for the current TID values and 0.3 for the AST values). The AST inventory values for the tellurium parents of Iodine-131 and Iodine-132 were obtained, and five percent (the AST tellurium release fraction) of these values were added to adjust the Iodine-131 and Iodine-132 inventories to conservatively account for daughter product in growth. The resulting AST values were then divided by the current TID values to obtain a multiplier for each radionuclide. Since doses are dominated by Iodine-131, a multiplier of 1.22 was used to adjust the TID inventory doses (noble gas and radioiodines) for AST noble gas and radioiodine inventory. This multiplier value is conservative for the noble gas inventories.

Noble Gas and Radioiodine Plume Inhalation and Whole Body Dose and Reactor Building Shine Dose Considering Complete Radioiodine Re-evolution

The product of the power multiplier factor (1.14) and the inventory multiplier factor (1.22) is 1.39 and was applied to the TID source term thyroid and whole body doses to obtain doses that represent an AST noble gas and radioiodine source term inventory.

Conversion of Doses to TEDE Values

The current TID calculated dose contributions are based on the International Commission on Radiation Protection standards. AST doses use TEDE terms and a conversion of the calculated doses to TEDE terms was necessary. This was accomplished as follows:

The thyroid dose was multiplied by the thyroid dose Committed Effective Dose Equivalent (CEDE) weighting factor and that result was added to the whole body dose.

To convert the doses to TEDE values, the contribution of each of the iodine radionuclides to the thyroid dose was determined. The weighted ratio of CEDE DCF to thyroid DCF is dominated by Iodine-131 and Iodine-133 because of their very large contribution to thyroid dose. The percentage contribution of the other iodine radionuclides is very low. Thus, even with higher ratios of CEDE to thyroid dose, their contribution to the inhalation component of the TEDE will be small.

Using a bounding weighted-average ratio of CEDE to thyroid for the important iodine radionuclides, the thyroid doses were converted to CEDE and added to the whole body dose to obtain TEDE values. The results represent an estimate of the TEDE for each location for the noble gas and radioiodine contribution of the AST source inventory with complete radioiodine re-evolution and EPU considered.

AST Plume Exposure Dose Contribution from Radionuclides other than Noble Gas and Radioiodine

The AST release includes radionuclides identified in RG-1.183 that contribute to the TEDE dose, which are not modeled in the TID analysis. To account for these, the STARDOSE check model results of the control room dose were used to determine the contribution of these "other" radionuclides. This contribution was determined to be 0.08 rem out of a total control room dose of 0.48 rem (20 percent). Therefore, this value (.08 rem) was added to the total control room dose.

For offsite dose contribution of the other radionuclides, an additional dose equal to the full AST calculated offsite doses (1.02 rem for the EAB and 1.25 rem for the LPZ) was assumed to represent the other contributions. These were added to the adjusted TID doses. This is a conservative addition since the offsite dose contribution due to other radionuclides must be less than the total dose.

Increase in the Reactor Building Shine Dose due to Radionuclides other than Noble Gas and Radioiodine

In the AST analysis, it was shown by comparing reactor building activity to the current TID analysis at many points in time, that the reactor building shine analysis for the current TID bounds AST if the TID analysis value is adjusted for the EPU power level. The TID reactor building shine contribution was, therefore, increased to account for the effect of the power level increase as well as the re-evolved radioiodine and noble gas available AST source term inventory. In addition, the contribution of other AST analysis radionuclides (non-noble gas, non-radioiodine) were considered.

Other radionuclides airborne in the primary and secondary containments are in particulate form and are removed from the containment atmosphere by natural mechanisms within 24 hours. Since these radionuclides are not affected by re-evolution, their impact on the control room integrated dose in terms of an airborne contribution to reactor building shine is small. For the waterborne dose contribution (core spray piping), there is already considered in the TID shine dose a factor of 1.67 more radioiodine than is needed according to the AST specification (50 percent versus 30 percent of the core inventory). Finally, as indicated earlier, the plume exposure contribution of the other radionuclides for the AST is 20 percent. Therefore, it is reasonable to increase the reactor building shine dose contribution by this 20 percent factor to account for the effect of "other" radionuclides on the reactor building shine dose.

Therefore, the TID reactor building shine dose contribution was increased by the power level and AST inventory multipliers plus an additional multiplier of 1.2. The result was added to the control room dose previously adjusted.

Summary

This evaluation established that the resulting estimated values of radiological doses for a failure of the SLC system to control pH do not exceed the regulatory limits of 10 CFR 50.67. The evaluation results bound potential AST analysis of re-evolved iodine. Because of the conservative values and methodology utilized, these results will encompass specific AST analysis re-evolution of iodine.

The final dose values of this evaluation are as follows (doses in rem):

	Total TEDE	10 CFR 50.67 Limits
EAB	3.99	25
LPZ	25	25
CR	4.74	5

These doses include the effects of EPU and increased Ci/MWt values, as well as a representation of complete radioiodine re-evolution. The evaluation results do not credit SGT system filtration or CREV system filtration. The impact of non-noble gas and non-radioiodine radionuclides included in the AST analysis is reflected in both the plume doses and the reactor building shine doses that contribute to the TEDE values.

References:

1. Technical Information Document (TID) - 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," U.S. Atomic Energy Commission, Dated March 23, 1962.

2. NRC Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," dated June 1974.

3. NRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Dated July 2000.

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

EXEMPTION REQUEST FROM THE REQUIREMENTS OF 10 CFR 50 APPENDIX A GENERAL DESIGN CRITERIA (GDC)-41 IN SUPPORT OF TECHNICAL SPECIFICATIONS CHANGE (TS-405) - ALTERNATIVE SOURCE TERM (AST)

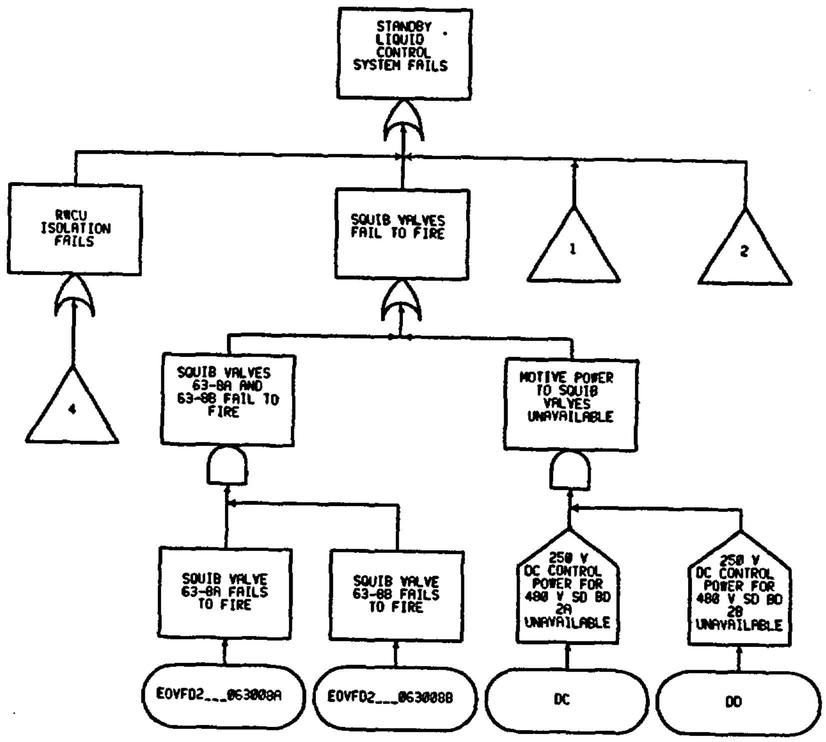
STANDBY LIQUID CONTROL SYSTEM PROBABILISTIC SAFETY ANALYSIS (PSA) FAULT TREE

Attached are the SLC system fault tree and the indicated quantification and RISKMAN® files of common cause and basic event reports. Failure probabilities are taken from the Pickard Lowe Garrett (PLG) database (PLG-0500, Vol 2, R0, "Database for PRA of Light Water Nuclear Power Plants"). The fault trees, which evaluate the current design function of the SLC system, reflects redundancy in components that ensures the SLC system will control the suppression pool pH following a postulated design basis Loss-of Coolant accident and meets the underlying purpose of GDC-41 in providing a sound and reliable system to clean-up the containment atmosphere.

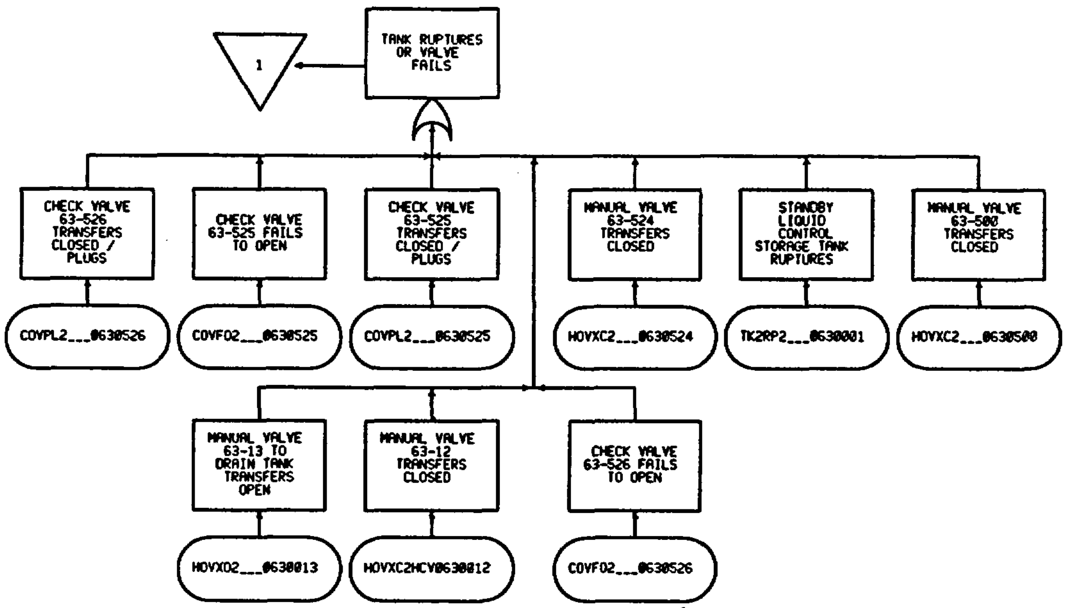
4. SECTION 4

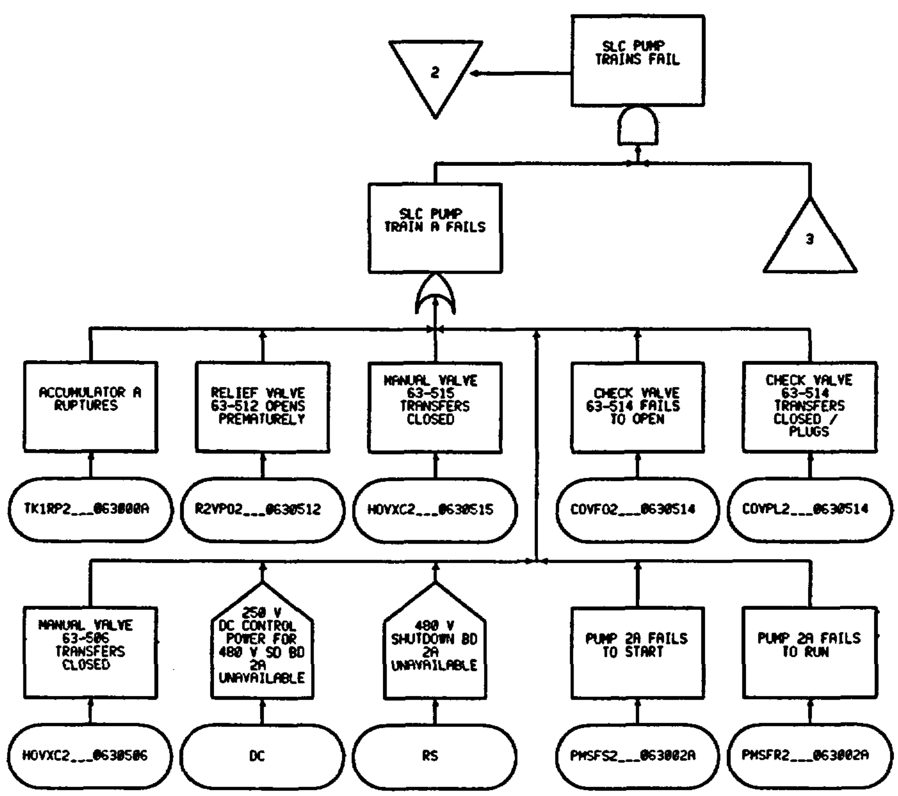
SYSTEM FAULT TREE

Top Event: SL

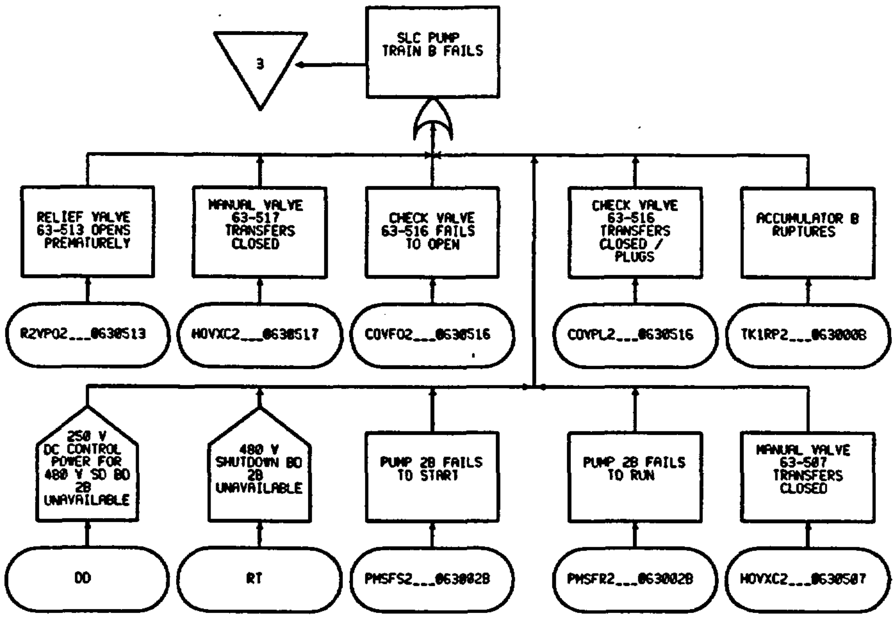


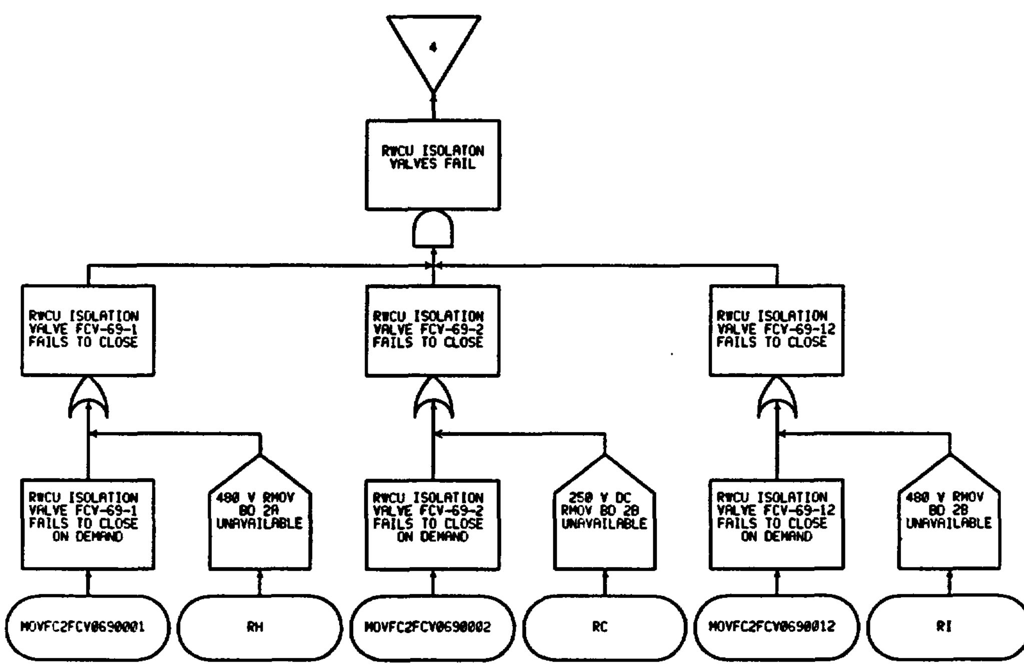
Top Event: SL


1
TANK RUPTURES OR VALVE FAILS
CHECK VALVE 63-526 TRANSFERS CLOSED / PLUGS
CHECK VALVE 63-525 FAILS TO OPEN
CHECK VALVE 63-525 TRANSFERS CLOSED / PLUGS
MANUAL VALVE 63-524 TRANSFERS CLOSED
STANDBY LIQUID CONTROL STORAGE TANK RUPTURES
MANUAL VALVE 63-500 TRANSFERS CLOSED
COVPL2___0630526
COVFO2___0630525
COVPL2___0630525
HOVXC2___0630524
TK2RP2___0630001
HOVXC2___0630500
MANUAL VALVE 63-13 TO DRAIN TANK TRANSFERS OPEN
MANUAL VALVE 63-12 TRANSFERS CLOSED
CHECK VALVE 63-526 FAILS TO OPEN
HOVXO2___0630013
HOVXC2HCV0630012
COVFO2___0630526


2
SLC PUMP TRAINS FAIL
SLC PUMP TRAIN A FAILS
3
ACCUMULATOR A RUPTURES
RELIEF VALVE 63-512 OPENS PREMATURELY
MANUAL VALVE 63-515 TRANSFERS CLOSED
CHECK VALVE 63-514 FAILS TO OPEN
CHECK VALVE 63-514 TRANSFERS CLOSED / PLUGS
TK1RP2___063000A
R2VPO2___0630512
HOVXC2___0630515
COVFO2___0630514
COVPL2___0630514
MANUAL VALVE 63-506 TRANSFERS CLOSED
250 V DC CONTROL POWER FOR 480 V SD BD 2A UNAVAILABLE
480 V SHUTDOWN BD 2A UNAVAILABLE
PUMP 2A FAILS TO START
PUMP 2A FAILS TO RUN
HOVXC2___0630506
DC
RS
PMSFS2___063002A
PMSFR2___063002A

Top Event: SL


3
SLC PUMP TRAIN B FAILS
RELIEF VALVE 63-513 OPENS PREMATURELY
MANUAL VALVE 63-517 TRANSFERS CLOSED
CHECK VALVE 63-516 FAILS TO OPEN
CHECK VALVE 63-516 TRANSFERS CLOSED / PLUGS
ACCUMULATOR B RUPTURES
R2VPO2___0630513
HOVXC2___0630517
COVFO2___0630516
COVPL2___0630516
TK1RP2___063000B
250 V DC CONTROL POWER FOR 480 V SD BD 2B UNAVAILABLE
480 V SHUTDOWN BD 2B UNAVAILABLE
PUMP 2B FAILS TO START
PUMP 2B FAILS TO RUN
MANUAL VALVE 63-507 TRANSFERS CLOSED
DD
RT
PMSFS2___063002B
PMSFR2___063002B
HOVXC2___0630507


4
RWCU ISOLATON VALVES FAIL
RWCU ISOLATION VALVE FCV-69-1 FAILS TO CLOSE
RWCU ISOLATION VALVE FCV-69-2 FAILS TO CLOSE
RWCU ISOLATION VALVE FCV-69-12 FAILS TO CLOSE
RWCU ISOLATION VALVE FCV-69-1 FAILS TO CLOSE ON DEMAND
480 V RMOV BD 2A UNAVAILABLE
RWCU ISOLATION VALVE FCV-69-2 FAILS TO CLOSE ON DEMAND
250 V DC RMOV BD 2B UNAVAILABLE
RWCU ISOLATION VALVE FCV-69-12 FAILS TO CLOSE ON DEMAND
480 V RMOV BD 2B UNAVAILABLE
MOVFC2FCV0690001
RH
MOVFC2FCV0690002
RC
MOVFC2FCV0690012
RI

ENCLOSURE 4

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

EXEMPTION REQUEST FROM THE REQUIREMENTS OF 10 CFR 50 APPENDIX A GENERAL DESIGN CRITERIA (GDC)-41 IN SUPPORT OF TECHNICAL SPECIFICATIONS CHANGE (TS-405) - ALTERNATIVE SOURCE TERM (AST)

STANDBY LIQUID CONTROL SYSTEM SKETCH

The attached is an sketch depicting the horizontal orientation of containment isolation check valves.

5. SECTION 5

QUANTIFICATION AND RISKMAN® FILES

SYSTEM NOTEBOOK
For TOP EVENT: SL
STANDBY LIQUID CONTROL SYSTEM
09:23:39 04 AUG 2000

SYSTEM NOTEBOOK
Common Cause Report for SL
Section I

MODEL Name: BFNU2R0
CCF Model Report for Top Event SL

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Page 1

Group ID : A

Basic Events	Description
PMSFS2___063002A	PUMP 2A FAILS TO START
PMSFS2___063002B	PUMP 2B FAILS TO START

Algebraic Method: MGL
Order = 1 out of 2

Failure Mode ID : START
Total Failure Rate = BTPSLS
Beta = BBSLPS

Group ID : B

Basic Events	Description
PMSFR2___063002A	PUMP 2A FAILS TO RUN
PMSFR2___063002B	PUMP 2B FAILS TO RUN

Algebraic Method: MGL
Order = 1 out of 2

Failure Mode ID : RUN
Total Failure Rate = BTPSLR * @T
Beta = BBSLPR

Group ID : C

Basic Events	Description
EOVFD2___063008A	SQUIB VALVE 63-8A FAILS TO FIRE
EOVFD2___063008B	SQUIB VALVE 63-8B FAILS TO FIRE

Algebraic Method: MGL
Order = 1 out of 2

Failure Mode ID : DEMAND
Total Failure Rate = ZTVSQD
Beta = ZBVSQD

Group ID : D

Basic Events	Description
MOVFC2FCV0690001	RWCU ISOLATION VALVE FCV-69-1 FAILS TO CLOSE
MOVFC2FCV0690002	RWCU ISOLATION VALVE FCV-69-2 FAILS TO CLOSE
MOVFC2FCV0690012	RWCU ISOLATION VALVE FCV-69-12 FAILS TO CLOSE ON DEMAND

Algebraic Method: MGL
Order = 2 out of 3

Failure Mode ID : CLOSE
Total Failure Rate = ZTVMOD
Beta = ZBVM3D

MODEL Name: BFNU2R0
CCF Model Report for Top Event SL

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Page 2

Gamma = ZGVM3D

SYSTEM NOTEBOOK
Basic Event Report for SL
Section II

MODEL Name: BFNU2R0
Basic Event Report for Top Event SL

09:23:48 04 AUG 2000
Page 1

```
Basic Events      Description
----------------  ------------------------------------------------------------------------------------

COVFO2___0630514  CHECK VALVE 63-514 FAILS TO OPEN

     ZTVCOD = 2.6907E-04

COVFO2___0630516  CHECK VALVE 63-516 FAILS TO OPEN

     ZTVCOD = 2.6907E-04

COVFO2___0630525  CHECK VALVE 63-525 FAILS TO OPEN

     ZTVCOD = 2.6907E-04

COVFO2___0630526  CHECK VALVE 63-526 FAILS TO OPEN

     ZTVCOD = 2.6907E-04

COVPL2___0630514  CHECK VALVE 63-514 TRANSFERS CLOSED / PLUGS

     ZTVCOP * @T = 2.0822E-08

COVPL2___0630516  CHECK VALVE 63-516 TRANSFERS CLOSED / PLUGS

     ZTVCOP * @T = 2.0822E-08

COVPL2___0630525  CHECK VALVE 63-525 TRANSFERS CLOSED / PLUGS

     ZTVCOP * @T = 2.0822E-08

COVPL2___0630526  CHECK VALVE 63-526 TRANSFERS CLOSED / PLUGS

     ZTVCOP * @T = 2.0822E-08

DC                250V DC POWER FOR 4KV SD BD C & 480V SD BD 1B UNAVAILABLE

     Constant Value: 1.0

DD                250V DC POWER FOR 4KV SD BD D & 480V SD BD 2B UNAVAILABLE

     Constant Value: 1.0

EOVFD2___063008A  SQUIB VALVE 63-8A FAILS TO FIRE

     EOVFD2___063008A is replaced in Common Cause Group C

{EOVFD2___063008A} Common Cause: Group C, 1/2

     (1-(ZBVSQD))*(ZTVSQD) = 2.4775E-03

EOVFD2___063008B  SQUIB VALVE 63-8B FAILS TO FIRE

     EOVFD2___063008B is replaced in Common Cause Group C

{EOVFD2___063008B} Common Cause: Group C, 1/2

     (1-(ZBVSQD))*(ZTVSQD) = 2.4775E-03

HOVXC2HCV0630012  MANUAL VALVE 63-12 TRANSFERS CLOSED

     ZTVHOT * (@T + @TE2) = 3.3574E-07

HOVXC2___0630500  MANUAL VALVE 63-500 TRANSFERS CLOSED

     ZTVHOT * (@T+@TE1) = 4.6920E-05

HOVXC2___0630506  MANUAL VALVE 63-506 TRANSFERS CLOSED
```

MODEL Name: BFNU2R0
Basic Event Report for Top Event SL

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Page 2

Basic Events	Description

ZTVHOT * (@T+@TE1) = 4.6920E-05

HOVXC2___0630507 MANUAL VALVE 63-507 TRANSFERS CLOSED

ZTVHOT * (@T+@TE1) = 4.6920E-05

HOVXC2___0630515 MANUAL VALVE 63-515 TRANSFERS CLOSED

ZTVHOT * (@T+@TE1) = 4.6920E-05

HOVXC2___0630517 MANUAL VALVE 63-517 TRANSFERS CLOSED

ZTVHOT * (@T+@TE1) = 4.6920E-05

HOVXC2___0630524 MANUAL VALVE 63-524 TRANSFERS CLOSED

ZTVHOT * (@T+@TE) = 3.7477E-04

HOVXO2___0630013 MANUAL VALVE 63-13 TO DRAIN TANK TRANSFERS OPEN

ZTVHOT * @T = 8.3936E-08

MOVFC2FCV0690001 RWCU ISOLATION VALVE FCV-69-1 FAILS TO CLOSE

MOVFC2FCV0690001 is replaced in Common Cause Group D

[MOVFC2FCV0690001] Common Cause: Group D, 1/3

(1-(ZBVM3D))*(ZTVMOD) = 4.0584E-03

MOVFC2FCV0690002 RWCU ISOLATION VALVE FCV-69-2 FAILS TO CLOSE

MOVFC2FCV0690002 is replaced in Common Cause Group D

[MOVFC2FCV0690002] Common Cause: Group D, 1/3

(1-(ZBVM3D))*(ZTVMOD) = 4.0584E-03

MOVFC2FCV0690012 RWCU ISOLATION VALVE FCV-69-12 FAILS TO CLOSE ON DEMAND

MOVFC2FCV0690012 is replaced in Common Cause Group D

[MOVFC2FCV0690012] Common Cause: Group D, 1/3

(1-(ZBVM3D))*(ZTVMOD) = 4.0584E-03

PMSFR2___063002A PUMP 2A FAILS TO RUN

PMSFR2___063002A is replaced in Common Cause Group B

[PMSFR2___063002A] Common Cause: Group B, 1/2

(1-(BBSLPR))*(BTPSLR * @T) = 9.4417E-05

PMSFR2___063002B PUMP 2B FAILS TO RUN

PMSFR2___063002B is replaced in Common Cause Group B

[PMSFR2___063002B] Common Cause: Group B, 1/2

(1-(BBSLPR))*(BTPSLR * @T) = 9.4417E-05

PMSFS2___063002A PUMP 2A FAILS TO START

MODEL Name: BFNU2R0
Basic Event Report for Top Event SL

09:23:49 04 AUG 2000
Page 3

Basic Events	Description
	PMSFS2___063002A is replaced in Common Cause Group A
[PMSFS2___063002A]	Common Cause: Group A, 1/2
	(1-(BBSLPS))*(BTPSLS) = 7.2466E-04
PMSFS2___063002B	PUMP 2B FAILS TO START
	PMSFS2___063002B is replaced in Common Cause Group A
[PMSFS2___063002B]	Common Cause: Group A, 1/2
	(1-(BBSLPS))*(BTPSLS) = 7.2466E-04
R2VPO2___0630512	RELIEF VALVE 63-512 OPENS PREMATURELY
	ZTVR2T * @T = 1.2124E-05
R2VPO2___0630513	RELIEF VALVE 63-513 OPENS PREMATURELY
	ZTVR2T * @T = 1.2124E-05
RC	250 DC RMOV BOARD 2B UNAVAILABLE
	Constant Value: 1
RH	480V RMOV BD 2A UNAVAILABLE
	Constant Value: 1.0
RI	480V RMOV BD 2B UNAVAILABLE
	Constant Value: 1.0
RS	480V SHUTDOWN BD 2A UNAVAILABLE
	Constant Value: 1.0
RT	480V SD BD 2B UNAVAILABLE
	Constant Value: 1.0
TK1RP2___063000A	ACCUMULATOR A RUPTURES
	ZTTK1B * @T = 5.3278E-08
TK1RP2___063000B	ACCUMULATOR B RUPTURES
	ZTTK1B * @T = 5.3278E-08
TK2RP2___0630001	STANDBY LIQUID CONTROL TANK RUPTURES
	ZTTK1B * @T = 5.3278E-08
[EOVFD2___063008A,EOVFD2___063008B]	Common Cause: Group C, 2/2
	(ZBVSQD)*(ZTVSQD) = 1.8640E-04
[MOVFC2FCV0690001,MOVFC2FCV0690002]	Common Cause: Group D, 2/3
	5.0000E-01*(ZBVM3D)*(1-(ZGVM3D))*(ZTVMOD) = 8.8198E-05
[MOVFC2FCV0690001,MOVFC2FCV0690012]	Common Cause: Group D, 2/3
	5.0000E-01*(ZBVM3D)*(1-(ZGVM3D))*(ZTVMOD) = 8.8198E-05

MODEL Name: BFNU2R0
Basic Event Report for Top Event SL

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Basic Events	Description

[MOVFC2FCV0690002,MOVFC2FCV0690012]Common Cause: Group D, 2/3

5.0000E-01*(ZBVM3D)*(1-(ZGVM3D))*(ZTVMOD) = 8.8198E-05

[PMSFR2___063002A,PMSFR2___063002B]Common Cause: Group B, 2/2

(BBSLPR)*(BTPSLR * @T) = 5.5708E-06

[PMSFS2___063002A,PMSFS2___063002B]Common Cause: Group A, 2/2

(BBSLPS)*(BTPSLS) = 1.1819E-04

[MOVFC2FCV0690001,MOVFC2FCV0690002,MOVFC2FCV0690012]Common Cause: Group D, 3/3

(ZBVM3D)*(ZGVM3D)*(ZTVMOD) = 6.0456E-05

MODEL Name: BFNU2R0
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Local Variables	Description	Equation
@T	MISSION TIME	2.0
@TE	EXPOSURE	24*31*24/2
@TE1	EXPOSURE1	3*31*24/2
@TE2	EXPOSURE2	12/2
@HS44A	MISALIGNMENT	ZHERLL

MODEL Name: BFNU2R0
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Database Variables

BBSLPR

BBSLPS

BTPSLR

BTPSLS

ZBVM3D

ZBVSQD

ZGVM3D

ZHERLL

ZTPMSR

ZTPMSS

ZTTK1B

ZTVCOD

ZTVCOP

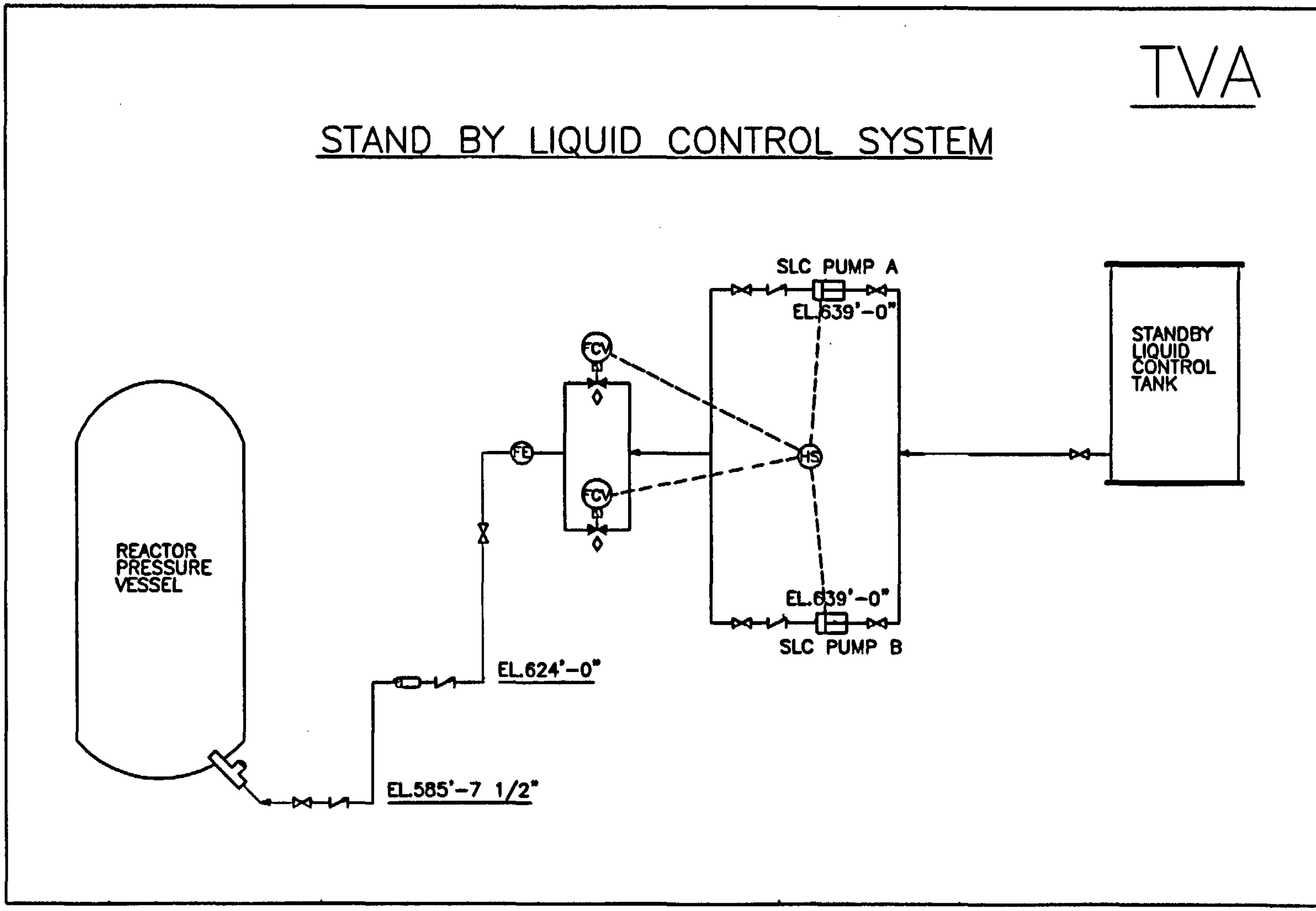
ZTVHOT

ZTVMOD

ZTVR2T

ZTVSOD

ZTVSQD


TVA
STAND BY LIQUID CONTROL SYSTEM
SLC PUMP A
EL.639'-0"
STANDBY LIQUID CONTROL TANK
FCV
FE
HS
FCV
REACTOR PRESSURE VESSEL
EL.639'-0"
SLC PUMP B
EL.624'-0"
EL.585'-7 1/2"