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L-08-167

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

SUBJECT:
Davis-Besse Nuclear Power Station
Docket No. 50-346, License No. NPF-3
Technical Specification Bases Update

In accordance with Davis-Besse Nuclear Power Station Technical Specification 6.17, "Technical Specifications (TS) Bases Control Program", enclosed are the changes made to the TS Bases implemented without prior NRC approval. The enclosure reflects changes made to the TS Bases from June 12, 2006 through May 20, 2008. The attachment lists the pages enclosed.

There are no regulatory commitments contained in this letter. If there are any questions or if additional information is required, please contact Mr. Thomas A. Lentz, Manager – FENOC Fleet Licensing, at (330) 761-6071.

Sincerely,

Barry S. Allen

Barry S. Allen

Attachment: Davis-Besse Nuclear Power Station Technical Specification Bases List Of Changed Pages

Enclosure: Changed Pages from the Davis-Besse Nuclear Power Station Technical Specification Bases

A001
NRR

cc: NRC Region III Administrator
NRC Resident Inspector
NRR Project Manager
Utility Radiological Safety Board

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Davis-Besse Nuclear Power Station
Technical Specification Bases
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Changed Pages From The Technical Specification Bases
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inject water into the reactor coolant system is disabled to ensure operation within reactor coolant system pressure-temperature limits.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of the ASME Code for Operation and Maintenance of Nuclear Power Plants and applicable Addenda.

The pressurizer code safety valves must be set such that the peak Reactor Coolant System pressure does not exceed 110% of design system pressure (2500 psig) or, 2750 psig. The control rod group withdrawal accident will result in the most limiting high pressure in the RCS. The analysis assumes RPS high pressure trip at 2400 psia and the code safety valves open at 2500 psig. The tolerance on the RPS instrument accuracy is 30 psi and, it is +1% for the code safety valve settings. The pressurizer pilot operated relief valve was assumed not to open for this transient. The resulting system peak pressure was calculated to be 2750 psia (2736 psig). Therefore, the code safety valve setpoint is conservatively set at ≤ 2525 psig which is the maximum pressure of 2500 psig +1% for tolerance.

The pressurizer pilot operated relief valve should be set such that it will open before the code safety valves are opened. However, it should not open on any anticipated transients. BAW-1890, September 1985 identified that the turbine trip from full power would cause the largest overpressure transient. This report demonstrated that with a RPS high pressure trip setpoint of 2355 psig the resulting overshoot in RCS pressure would be limited to 50 psi. Consequently, the minimum PORV setpoint needs to accommodate both the RCS pressure overshoot and the RPS instrument string error of 30 psi.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and pilot operated relief valve against water relief.

The low level limit is based on providing enough water volume to prevent the low level interlock from de-energizing the pressurizer heaters during steady state operations. The high level limit is based on providing enough steam volume to prevent water relief through the pressurizer relief valves during the most challenging anticipated pressurizer insurge transient, which is a loss of feedwater. Since prevention of water relief is a goal for abnormal transient operation, rather than a Safety Limit, the value for high pressurizer level is nominal and is not adjusted for instrument error.

The ACTION statement provides 1 hour to restore pressurizer level prior to requiring shutdown. The 1-hour completion time is considered to be a reasonable time for restoring pressurizer level to within limits.

The pilot operated relief valve and steam bubble function to relieve RCS pressure during all design transients. Operation of the pilot operated relief valve minimizes the undesirable opening of the spring-loaded pressurizer code safety valves.

3/4.4.5 STEAM GENERATOR TUBE INTEGRITY

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by LCO 3.4.4.1, "Coolant Loops and Coolant Circulation – Startup and Power Operation," and LCO 3.4.1.2, "Coolant Loops and Coolant Circulation – Shutdown and Hot Standby."

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 6.8.4.g, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 6.8.4.g, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational leakage. The SG performance criteria are described in Specification 6.8.4.g. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

The processes used to meet the SG performance criteria are defined by NEI 97-06, "Steam Generator Program Guidelines."

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged or repaired in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is repaired or removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged or repaired, the tube may still have tube integrity.

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall and any repairs made to it, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 6.8.4.g, "Steam Generator (SG) Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

There are three SG performance criteria: structural integrity, accident induced leakage, and operational leakage. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment

of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB and Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.

The accident induced leakage performance criterion ensures that the primary to secondary leakage caused by a design basis accident, other than a steam generator tube rupture (SGTR), is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 1 gpm per SG. The accident induced leakage rate includes any primary to secondary leakage existing prior to the accident in addition to primary to secondary leakage induced during the accident.

The operational leakage performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational leakage is contained in LCO 3.4.6.2, "Reactor Coolant System - Operational Leakage," and limits primary to secondary leakage through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of leakage is due to more than one crack, the cracks are very small, and the above assumption is conservative.

ACTION statement 3.4.5.a applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged or repaired in accordance with the Steam Generator Program as required by SR 4.4.5.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG repair criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged or repaired has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, ACTION statement 3.4.5.b applies.

A completion time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, ACTION statement 3.4.5.a.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged or repaired prior to entering HOT SHUTDOWN following the next refueling outage or SG inspection. This completion time is acceptable since operation until the next inspection is supported by the operational assessment.

If the requirements of ACTION statement 3.4.5.a are not met or if SG tube integrity is not being maintained, the reactor must be brought to HOT STANDBY within 6 hours and COLD SHUTDOWN within the following 30 hours. The allowed completion times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

During shutdown periods the SGs are inspected as required by SR 4.4.5.1 and the Steam Generator Program. NEI 97-06, "Steam Generator Program Guidelines," and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube repair criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, non-destructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the frequency of SR 4.4.5.1. The frequency is determined by the operational assessment and other limits in the SG examination guidelines (EPRI "Pressurized Water Reactor Steam Generator Examination Guidelines"). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 6.8.4.g contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

Surveillance Requirement 4.4.5.2 requires verification that each inspected SG tube that satisfies the tube repair criteria is plugged or repaired in accordance with the SG Program. The tube repair criteria delineated in Specification 6.8.4.g are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). NEI 97-06, "Steam Generator Program Guidelines," provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

Steam generator tube repairs are only performed using approved repair methods as described in the Steam Generator Program.

The SR 4.4.5.2 frequency of "prior to entering HOT SHUTDOWN following a SG tube inspection" ensures that the surveillance has been completed and all tubes meeting the repair criteria are plugged or repaired prior to subjecting the SG tubes to significant primary to secondary pressure differential.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to detect and monitor leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendation of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

Industry experience has shown that, while a limited amount of leakage is expected from the RCS, the UNIDENTIFIED LEAKAGE portion of this can be reduced to a threshold value of less that 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The steam generator tube leakage limit of 150 GPD through any one steam generator ensures that the dosage contribution from tube leakage will be limited to a small fraction of 10 CFR Part 100 limits in the event of either a steam generator tube rupture or steam line break. A 1 GPM total primary to secondary leakage limit is used in the analysis of these accidents.

The limit of 150 GPD per steam generator (SG) is based on the operational leakage performance criterion in NEI 97-06, "Steam Generator Program Guidelines." The Steam Generator Program operational leakage performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

Surveillance Requirement 4.4.6.2.1.d is not applicable to primary to secondary leakage because leakage of 150 gallons per day cannot be measured accurately by an RCS inventory water balance.

Surveillance Requirement 4.4.6.2.1.e verifies that primary to secondary leakage is less than or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary leakage limit ensures that the operational leakage performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.5, "Steam Generator (SG) Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in the EPRI "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines." The operational leakage rate limit applies to leakage through any one SG. If it is not practical

to assign the leakage to an individual SG, all the primary to secondary leakage should be conservatively assumed to be from one SG. The Surveillance Requirement is modified by a note which states that the Surveillance Requirement is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary leakage determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows. The Surveillance Requirement frequency of 72 hours is a reasonable interval to trend primary to secondary leakage and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary leakage is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The CONTROLLED LEAKAGE limit of 10 GPM restricts operation with a total RCS leakage from all RC pump seals in excess of 10 GPM.

The surveillance requirements for RCS Pressure Isolation Valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS Pressure Isolation Valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.7 CHEMISTRY

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3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of the Part 100 limit following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in the specific site parameters of the site, such as site boundary location and meteorological conditions, were not considered in this evaluation. The NRC is finalizing site specific criteria which will be used as the basis for the reevaluation of the specific activity limits of this site. This reevaluation may result in higher limits.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation and air lock door requirements, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

As described in Administrative Controls Section 6.16, the Containment Leakage Rate Testing Program is based on Option B of Appendix J of 10 CFR 50. The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak design basis loss of coolant accident pressure of 38 psig, P_a. As an added conservatism, the measured, overall, as-left integrated leakage rate is further limited to $\leq 0.75\ L_a$, during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The special test for the containment purge and exhaust isolation valves is intended to detect gross degradation of seals on the valve seats. The special test is performed in addition to the Appendix J requirements.

USAR 6.2.4 identifies all penetrations that are secondary containment bypass leakage paths.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to ensure CONTAINMENT INTEGRITY and to meet the restrictions on overall containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests. Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Containment Leakage Rate Testing Program, which is described in Administrative Controls Section 6.16.

One inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a design basis accident.

3/4.6.1.3 CONTAINMENT AIR LOCKS (Continued)

The air lock interlock allows only one air lock door of an air lock to be opened at a time. This provision ensures that a gross breach of containment does not exist when CONTAINMENT INTEGRITY is required. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, in MODES 1 through 4, both doors are kept closed when the air lock is not being used for entry and exit, i.e., containment entries/exits, air lock maintenance, or air lock testing.

The surveillance requirement which verifies that only one door in each air lock can be opened at a time is not part of the Containment Leakage Rate Testing Program. Therefore, its test frequency is subject to the provisions of Specification 4.0.2.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the annulus atmosphere of 0.5 psi and 2) the containment peak pressure does not exceed the maximum internal pressure of 40 psig during LOCA conditions.

The initial pressure condition used in the containment analysis was 1 psig. This resulted in a maximum peak pressure from a LOCA of 38 psig.

3/4.6.1.5 AIR TEMPERATURE

The limitations on Containment Vessel average air temperature ensure that the overall Containment Vessel average air temperature does not exceed the initial temperature condition assumed in the LOCA accident analysis of peak containment temperature.

Peak pressure during a LOCA is not sensitive to the initial average air temperature in the Containment Vessel.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

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3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

Maintaining the containment purge supply and exhaust isolation valves closed with control power removed at all times during MODES 1, 2, 3 and 4 provides assurance that the safety function of containment isolation is maintained in the event of a LOCA.

The ACTION statement assures that at least one containment purge supply and exhaust isolation valve is closed in each containment penetration and provides reasonable time to permit closure of an open valve.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the containment spray system ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

Borated Water Storage Tank (BWST) outlet isolation valves DH-7A and DH-7B are de-energized during MODES 1, 2, 3, and 4 to preclude postulated inadvertent closure of the valves in the event of a fire, which could result in a loss of the availability of the BWST. Re-energization of valves DH-7A and DH-7B is permitted on an intermittent basis during MODES 1, 2, 3 and 4 under administrative controls. Station procedures identify the precautions which must be taken when re-energizing these valves under such controls.

Containment Emergency Sump Recirculation Valves DH-9A and DH-9B are de-energized during MODES 1, 2, 3, and 4 to preclude postulated inadvertent opening of the valves in the event of a fire, which could result in draining the Borated Water Storage Tank to the Containment Emergency Sump and the loss of this water source for normal plant shutdown. Re-energization of valves DH-9A and DH-9B is permitted on an intermittent basis during MODES 1, 2, 3, and 4 under administrative controls. Station procedures identify the precautions which must be taken when re-energizing these valves under such controls.

3/4.6.2.2 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the containment cooling system ensures that 1) the containment air temperature will be maintained within limits during normal operation, and 2) adequate heat removal capacity is available when operated in conjunction with the containment spray systems during post-LOCA conditions.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the required time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA. Containment isolation valves and their required isolation times are addressed in the USAR. The opening of a closed inoperable containment isolation valve on an intermittent basis during plant operation is permitted under administrative control. Operating procedures identify those valves which may be opened under administrative control as well as the safety precautions which must be taken when opening valves under such controls.

The containment purge supply and exhaust system isolation valves are considered OPERABLE with respect to containment isolation when they meet the requirements of Specification 3.6.1.7.

Technical Specification (TS) 3.6.3.1 applies to all containment isolation valves (as identified in USAR Table 6.2-23), regardless of whether or not an SFAS automatic closure feature exists for the valve.

The OPERABILITY of systems which are required to have an open flowpath during accident conditions, but which have this flowpath secured due to implementation of a TS 3.6.3.1 Action statement, will be governed by the LCOs for those systems.

An inoperable containment isolation valve that has been disabled in the closed position should still be declared inoperable and the TS 3.6.3.1 Action statement entered. For example, an inoperable motor-operated containment isolation valve that is closed with power removed meets TS Action 3.6.3.1b, however the TS Action statement should not be exited until the valve is returned to OPERABLE status. Closure of the valve could impact other TS LCOs.

3/4.6.4 Deleted

3/4.6.5 SHIELD BUILDING

3/4.6.5.1 EMERGENCY VENTILATION SYSTEM

The OPERABILITY of the emergency ventilation systems ensures that containment vessel leakage occurring during LOCA conditions into the annulus will be filtered through the HEPA filters and charcoal adsorber trains prior to discharge to the atmosphere. This requirement is necessary to meet the assumptions used in the safety analyses and limit the site boundary radiation doses to within the limits of 10 CFR 100 during LOCA conditions. The proper functioning of the EVS fans, dampers, filters, adsorbers, etc., as a system is verified by the ability of each train to produce the required system flow rate.

The required emergency ventilation filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations).

3/4.6.5.2 SHIELD BUILDING INTEGRITY

Shield building integrity ensures that the release of radioactive material from the containment vessel will be restricted to those leakage paths and associated leak rates assumed in the safety analysis. The closure of the airtight doors and blowout panels listed in Table 4.6-1 ensure that the Emergency Ventilation System (EVS) can provide a negative pressure between 0.25 and 1.5 inches Water Gauge within the annulus between the shield building and containment vessel and within the interconnecting mechanical penetration rooms after a loss-of-coolant accident (LOCA). This restriction, in conjunction with the operation of the EVS, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

In the event shield building integrity, including the capability of the EVS to provide a negative pressure of greater than or equal to 0.25 inches Water Gauge, is not maintained, shield building integrity must be restored within 24 hours. Twenty-four hours is a reasonable completion time considering the limited leakage design of the containment and the low probability of a Design Basis Accident occurring during this time period.

3/4.6.5.3 SHIELD BUILDING STRUCTURAL INTEGRITY

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3/4.7.8 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from by-product, source, and special nuclear material sources will not exceed allowable intake values.

3/4.7.9 STEAM GENERATOR LEVEL

The steam generator water level limits are consistent with the initial assumptions in the USAR. While in MODE 3, examples of Main Feedwater Pumps that are incapable of supplying feedwater to the Steam Generators are tripped pumps or a manual valve closed in the discharge flowpath. The reactivity requirements to ensure adequate SHUTDOWN MARGIN are provided in plant operating procedures.

The steam generator minimum water level requirement is met by verifying the indicated steam generator level is greater than or equal to the value that corresponds to the required actual minimum level above the tubesheet.

3/4.7.10 FIRE BARRIERS -- DELETED

Surveillance Requirement 4.8.1.1.1.b is performed at least once each REFUELING INTERVAL during shutdown by demonstrating the capability of transferring (both manually from the control room and automatically) each 13.8 kV bus power supply from the unit auxiliary transformer to each startup transformer circuit. The transfer capability between the two startup transformers as well as the transfer capability between the two 13.8 – 4.16 kV bus tie transformers is not required to be tested per this Surveillance Requirement, since these transfer capabilities can not be relied upon for OPERABILITY of the subject equipment.

Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.c.4 verify proper starting of the Emergency Diesel Generators from standby conditions. Verification that an Emergency Diesel Generator has achieved a frequency of 60 Hz within the required time constraints meets the requirement for verifying the Emergency Diesel Generator has accelerated to 900 RPM.

Surveillance Requirements (SR) 4.8.1.1.2.a.7 and 4.8.1.1.2.c.7 verify that the automatic load sequence timer is operable with each load sequence time within 10% of its required value. In addition to equipment directly actuated by the Safety Features Actuation System Sequence Logic Channels, the operating Makeup Pump(s) will trip following a Loss of Offsite Power, and restart approximately 2.5 seconds after the associated Emergency Diesel Generator output breaker closes. SR 4.8.1.1.2.a.7 and 4.8.1.1.2.c.7 are applicable to the 2.5 second delay.

NRC Log Number 5668, dated May 31, 2000, provides guidance relative to the operability of the offsite A.C. electrical power sources. In summary, whenever switchyard equipment is removed from service or switchyard breakers are opened that leaves the remaining switchyard equipment vulnerable to a single point failure that would result in a loss of offsite power, TS 3.8.1.1 Action a must be entered and the appropriate actions taken as specified. For example, with either the Lemoyne line or the BayShore line out of service, the remaining two circuits are susceptible to a single event, and TS 3.8.1.1 Action a entry is appropriate. With the Ohio Edison line out of service, unless startup transformer No. 2 is also out of service, TS 3.8.1.1 Action a entry is not required.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that 1) the facility can be maintained in the shutdown or refueling condition for extended time periods and 2) sufficient instrumentation and control capability is available for monitoring and maintaining the facility status.

3/4.9.4 CONTAINMENT PENETRATIONS (Continued)

The individual designated to be continuously available to close the air lock door must be stationed at the auxiliary building side of the air lock. A containment personnel air lock door is considered capable of being closed if the door is not blocked in such a way that it cannot be expeditiously closed, and any hoses and cables running through the airlock employ a means to allow safe, quick disconnect or severance, and are tagged at the airlock with specific instructions to expedite removal. The LCO 3.9.10 requirement to maintain a minimum of 23 feet of water over the top of irradiated fuel assemblies seated within the reactor pressure vessel during movement of fuel assemblies within the reactor pressure vessel while in MODE 6 ensures that sufficient water depth is available to remove 99% of the assumed iodine gap activity released from the rupture of an irradiated fuel assembly. Further, sufficient time is available to close the personnel air lock following a loss of shutdown cooling before boiling occurs.

Regarding LCO 3.9.4.c, the phrase "atmosphere outside containment" refers to anywhere outside the containment vessel, including (but not limited to) the containment annulus and the auxiliary building. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Since containment pressurization is not expected, the 10 CFR 50 Appendix J leakage criteria and tests are not required. The LCO is modified by a Note allowing penetration flow paths with direct access from the containment to the outside atmosphere to be open under administrative controls. Admistrative controls ensure that 1) appropriate personnel are aware of the open status of the penetration flow path during CORE ALTERATIONS or movement of irradiated fuel assemblies within the containment, and 2) specified individuals are designated and readily available to isolate the flow path in the event of a Fuel Handling Accident.

The containment equipment hatch cover may be off during CORE ALTERATIONS or movement of irradiated fuel in containment provided the requirements of Specification 3.9.12 are satisfied. The requirements of Specification 3.9.12 ensure that the emergency ventilation system servicing the storage area is OPERABLE with the ability to filter any radioactive release through the containment equipment hatch following a fuel handling accident. Since containment closure is not credited for mitigating the consequences of the fuel handling accident as described in the Updated Safety Analysis Report, the equipment hatch cover need not be installed to ensure adequate protection of the public health or safety.