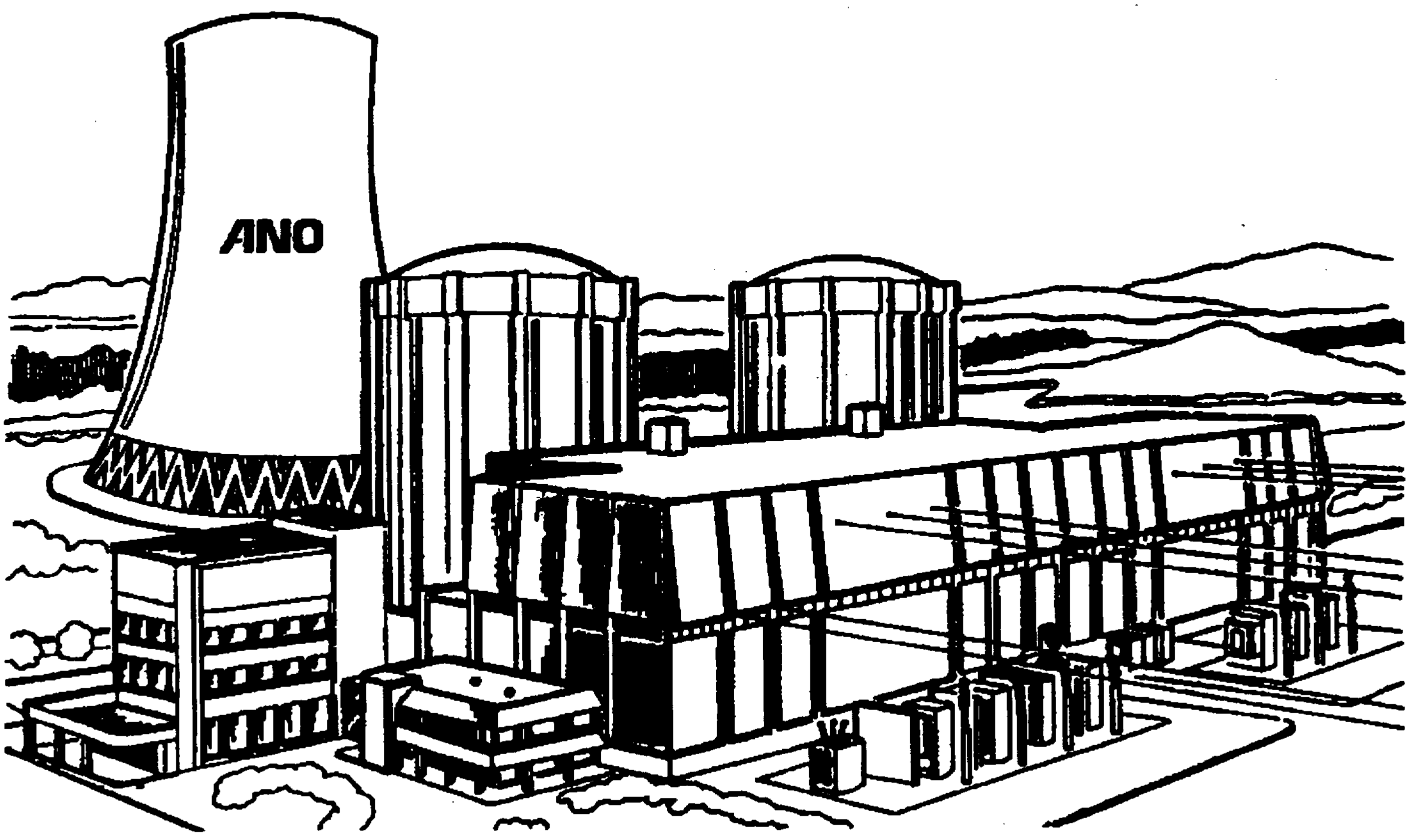
ARKANSAS NUCLEAR ONE - UNIT 1

IMPROVED TECHNICAL SPECIFICATIONS SUBMITTAL



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(Sections 3.9, 4.0, and 5.0)

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3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Reactor Coolant System and the refueling canal shall be maintained within the limit specified in the COLR.

APPLICABILITY: MODE 6.

---NOTE---
Only applicable to the refueling canal when connected to the RCS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend positive reactivity additions.	Immediately
	AND	
	A.3 Initiate action to restore boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.1.1	Verify boron concentration is within the limit specified in the COLR.	72 hours

3.9 REFUELING OPERATIONS

3.9.2 Nuclear Instrumentation

LCO 3.9.2

a. One source range neutron flux monitor shall be OPERABLE, and

b. One additional source range neutron flux monitor shall be OPERABLE during CORE ALTERATIONS.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required source range neutron flux monitor inoperable during CORE ALTERATIONS.	A.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
B. No OPERABLE source range neutron flux monitor.	B.1 Initiate action to restore one source range neutron flux monitor to OPERABLE status.	Immediately
	<u>AND</u>	
	B.2 Perform SR 3.9.1.1.	Once per 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.2.1	Perform CHANNEL CHECK.	12 hours
SR 3.9.2.2	------------------------------NOTE------------------------------ Neutron detectors are excluded from CHANNEL CALIBRATION. --- Perform CHANNEL CALIBRATION.	18 months

3.9 REFUELING OPERATIONS

3.9.3 Reactor Building Penetrations

LCO 3.9.3 The reactor building penetrations shall be in the following status:

a. The equipment hatch is capable of being closed;

b. One door in each air lock is capable of being closed; and

c. Each penetration providing direct access from the reactor building atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by an OPERABLE reactor building isolation valve, except reactor building purge isolation valves, or

3. capable of being closed by an OPERABLE reactor building purge isolation valve with the purge exhaust radiation monitoring channel OPERABLE.

APPLICABILITY: During movement of irradiated fuel assemblies within the reactor building.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more reactor building penetrations not in required status.	A.1 Suspend movement of irradiated fuel assemblies within the reactor building.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Verify each required reactor building penetration is in the required status.	7 days
SR 3.9.3.2	------------------------------NOTE------------------------------ Not required to be met for reactor building isolation valves and reactor building purge isolation valves in penetrations closed to comply with LCO c.1. --- Verify each required reactor building isolation valve and each reactor building purge isolation valve actuates to the isolation position.	18 months
SR 3.9.3.3	Perform CHANNEL CALIBRATION of reactor building purge exhaust radiation monitor.	18 months

3.9 REFUELING OPERATIONS

3.9.4 Decay Heat Removal (DHR) and Coolant Circulation - High Water Level

LCO 3.9.4 One DHR loop shall be OPERABLE and in operation.

--NOTE--
The required DHR loop may be removed from operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause introduction into the Reactor Coolant System, coolant with boron concentration less than that required to meet the minimum required boron concentration of LCO 3.1.1.

APPLICABILITY: MODE 6 with the water level ≥ 23 ft above the top of the irradiated fuel seated in the reactor pressure vessel.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DHR loop requirements not met.	A.1 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
	<u>AND</u>	
	A.2 Suspend loading irradiated fuel assemblies in the core.	Immediately
	<u>AND</u>	
	A.3 Initiate action to satisfy DHR loop requirements.	Immediately
	<u>AND</u>	
	A.4 Close all reactor building penetrations providing direct access from the reactor building atmosphere to outside atmosphere.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify one DHR loop is in operation.	12 hours

3.9 REFUELING OPERATIONS

3.9.5 Decay Heat Removal (DHR) and Coolant Circulation - Low Water Level

LCO 3.9.5 Two DHR loops shall be OPERABLE, and one DHR loop shall be in operation.

---NOTE---

1. All DHR pumps may be de-energized for ≤ 15 minutes when switching from one train to another provided:

   a. The core outlet temperature is maintained > 10 degrees F below saturation temperature;

   b. No operations are permitted that would cause a reduction of the Reactor Coolant System boron concentration; and

   c. No draining operations to further reduce RCS water volume are permitted.

2. One required DHR loop may be inoperable for up to 2 hours for surveillance testing, provided that the other DHR loop is OPERABLE and in operation.

APPLICABILITY: MODE 6 with the water level < 23 feet above the top of the irradiated fuel seated in the reactor pressure vessel.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Less than required number of DHR loops OPERABLE.	A.1	Initiate action to restore DHR loop to OPERABLE status.	Immediately
	<u>OR</u>		
	A.2	Initiate action to establish ≥ 23 feet of water above the top of the irradiated fuel seated in the reactor pressure vessel.	Immediately

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. No DHR loop OPERABLE or in operation.	B.1	Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
	AND		
	B.2	Initiate action to restore one DHR loop to OPERABLE status and to operation.	Immediately
	AND		
	B.3	Close all reactor building penetrations providing direct access from the reactor building atmosphere to outside atmosphere.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify one DHR loop is in operation.	12 hours
SR 3.9.5.2	Verify correct breaker alignment and indicated power available to each required DHR pump.	7 days

3.9 REFUELING OPERATIONS

3.9.6 Refueling Canal Water Level

LCO 3.9.6 Refueling canal water level shall be maintained ≥ 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel.

APPLICABILITY: During movement of irradiated fuel assemblies within the reactor building.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling cavity water level not within limit.	A.1 Suspend movement of irradiated fuel assemblies within the reactor building.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.6.1	Verify refueling canal water level is ≥ 23 feet above the top of irradiated fuel assemblies seated within the reactor pressure vessel.	24 hours

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS) and the refueling canal during refueling ensures that the reactor remains subcritical during MODE 6. The refueling boron concentration is specified for the coolant in each of these volumes since each volume has direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit specified in the COLR ensures an overall core reactivity of $k_{eff} \leq 0.99$ during fuel handling, with all CONTROL RODS out.

SAR, Section 1.4, GDC 26 requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Makeup and Purification System has the ability to initiate and maintain a cold shutdown condition in the reactor.

During refueling, the spent fuel pool, the transfer tube, the refueling canal and the reactor vessel are connected. As a result, the soluble boron concentration is relatively the same in each of these volumes.

Operation of the Decay Heat Removal (DHR) System in the RCS mixes the added concentrated boric acid with the water in the refueling canal. The DHR System is in operation during refueling (see LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation," and LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation-Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS and the refueling canal above the COLR limit.

APPLICABLE SAFETY ANALYSES

During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident in the accident analysis. The boron concentration limit specified in the COLR is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance.

The required boron concentration and the unit refueling procedures ensure the k_{eff} of the core will remain ≤ 0.99 during the refueling operation.

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36. (Ref. 2).

LCO

The LCO requires that a minimum boron concentration be maintained in the RCS and the refueling canal while in MODE 6. The boron concentration limit specified in the COLR ensures a core k_{eff} of ≤ 0.99 is maintained during fuel handling operations with CONTROL RODS and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by unit procedures.

Violation of the LCO provides a potential for an inadvertent criticality during MODE 6.

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical.

Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," LCO 3.1.5, "Safety Rod Insertion Limits," and LCO 3.2.1, "Regulating Rod Insertion Limits," ensure that an adequate amount of negative reactivity is available to shut down the reactor and to maintain it subcritical.

The Applicability is modified by a Note. The Note states that the limits on boron concentration are only applicable to the refueling canal when that volume is connected to the Reactor Coolant System. When the refueling canal is isolated from the RCS, no potential path for boron dilution exists.

ACTIONS

A.1 and A.2

Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO. If the boron concentration of the RCS or the refueling canal is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately.

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position. Operations that add limited positive reactivity (e.g., temperature fluctuations from inventory addition or temperature control fluctuations), but when combined with all other operations affecting core reactivity (e.g., intentional boration) result in overall net negative reactivity addition, are not precluded by this action.

A.3

In addition to immediately suspending CORE ALTERATIONS and positive reactivity additions, action to restore the concentration must be initiated immediately.

There is no unique design basis event analysis that requires a specific rate of boration. The only requirement is to restore the boron concentration to its required value as soon as possible.

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR ensures the coolant boron concentration in the RCS and the refueling canal is within the COLR limits. The boron concentration of the coolant in each volume is determined every 72 hours by chemical analysis. Prior to re-connecting portions of the refueling canal to the RCS, this SR must be met per SR 3.0.4. If any dilution activity has occurred while the cavity was disconnected from the RCS, this SR ensures the correct boron concentration prior to communication with the RCS.

The Frequency is based on industry experience, which has shown 72 hours to be adequate.

REFERENCES

1. SAR, Section 1.4, GDC 26.

2. 10 CFR 50.36.

B 3.9 REFUELING OPERATIONS

B 3.9.2 Nuclear Instrumentation

BASES

BACKGROUND

The source range neutron flux monitors are used during refueling operations to monitor the core reactivity condition. The installed source range neutron flux monitors are part of the Nuclear Instrumentation (NI) System. These detectors are located external to the reactor vessel and detect neutrons leaking from the core. The use of temporary detectors is permitted, provided the LCO requirements are met.

The installed source range neutron flux monitor channels include fission chamber detectors. The detectors monitor the neutron flux in counts per second. The instrument range covers six decades of neutron flux. The instrumentation also provides continuous visual indication in the control room to alert operators to a significant change in neutron flux. The NI system is designed in accordance with the criteria presented in Reference 1.

APPLICABLE SAFETY ANALYSES

An OPERABLE source range neutron flux monitor is required to provide indication to alert the operator to unexpected changes in core reactivity, such as may be caused by a boron dilution accident or an improperly loaded fuel assembly (Ref. 1).

The safety analysis of the uncontrolled boron dilution accident is described in Reference 2. The analysis of the uncontrolled boron dilution accident shows that the reactor remains subcritical. The source range neutron flux monitors are not credited for boron dilution event mitigation in the safety analysis.

The source range neutron flux monitors satisfy Criterion 4 of 10 CFR 50.36 (Ref. 3).

LCO

This LCO requires one source range neutron flux monitor OPERABLE to ensure that monitoring capability is available to detect changes in core reactivity. One additional source range neutron flux monitor shall be OPERABLE during CORE ALTERATIONS. This additional requirement ensures redundant monitoring capability when positive reactivity changes are being made to the core.

The use of temporary detectors is permitted for purposes of complying with this LCO. If used, the temporary detectors should be functionally equivalent to the installed source range monitors and satisfy applicable Surveillance Requirements.

APPLICABILITY

In MODE 6, the source range neutron flux monitor must be OPERABLE to determine changes in core reactivity. There is no other direct means available to check core reactivity levels. In MODES 2, 3, 4, and 5, source range detectors and circuitry are also required to be OPERABLE by LCO 3.3.9, "Source Range Neutron Flux."

In MODE 1, the neutron flux level is above the indicated range of the monitors. Thus, they are no longer relied upon for reactivity or power level monitoring. Hence, there are no requirements on source range neutron flux monitors in MODE 1.

ACTIONS

A.1 and A.2

With only one required source range neutron flux monitor OPERABLE during CORE ALTERATIONS, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1 must be suspended immediately. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

B.1

With no required source range neutron flux monitor OPERABLE, action to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, action shall be continued until a source range neutron flux monitor is restored to OPERABLE status or until the Applicability is exited.

B.2

With no required source range neutron flux monitor OPERABLE, there is no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and positive reactivity additions are not to be made in accordance with Required Actions A.1 and A.2, the core reactivity condition is stabilized until the source range neutron flux monitors are restored to an OPERABLE status. This stabilized condition is verified by performing SR 3.9.1.1 to ensure that the required boron concentration exists.

The Completion Time of once per 12 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration. The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this time period.

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

SR 3.9.2.1 is the performance of a CHANNEL CHECK, which is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. Changes in fuel loading and core geometry can also result in significant differences between source range channels, but each channel should be consistent with its local conditions. When in MODE 6 with only one channel OPERABLE, a CHANNEL CHECK is still required. However, in this condition, a redundant source range instrument may not be available for comparison. The CHANNEL CHECK provides verification that the OPERABLE source range channel is energized and indicating a value consistent with current unit status.

The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified for the same instruments in LCO 3.3.9.

SR 3.9.2.2

SR 3.9.2.2 is the performance of a CHANNEL CALIBRATION every 18 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the source range nuclear instrument is a complete check and re-adjustment of the channel, from the pre-amplifier input to the indicator. The 18 month Frequency is based on industry experience which has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

REFERENCES

1. SAR, Section 1.4, GDC 13, GDC 26, GDC 28, and GDC 29.

2. SAR, Section 14.1.2.4.

3. 10 CFR 50.36.

B 3.9 REFUELING OPERATIONS

B 3.9.3 Reactor Building Penetrations

BASES

BACKGROUND

During the movement of irradiated fuel assemblies within the reactor building, a release of fission product radioactivity within the reactor building will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, the containment of fission products is accomplished by maintaining the reactor building OPERABLE as described in LCO 3.6.1, "Reactor Building". In MODE 6, the potential for reactor building pressurization as a result of an accident is not likely; therefore, requirements to isolate the reactor building from the outside atmosphere can be less stringent. In order to make this distinction, the penetration requirements are referred to as "reactor building closure" rather than "reactor building OPERABILITY." Reactor building closure means that all potential direct release paths are closed or capable of being closed. Since there is no potential for significant reactor building pressurization, the Appendix J leakage criteria and tests are not required.

The reactor building serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10CFR100. Additionally, the reactor building provides radiation shielding from the fission products that may be present in the reactor building atmosphere following accident conditions.

The reactor building equipment hatch, which is part of the reactor building pressure boundary, provides a means for moving large equipment and components into and out of the reactor building. During the movement of irradiated fuel assemblies within the reactor building, the equipment hatch must be capable of being closed.

During the movement of irradiated fuel assemblies within the reactor building, administrative controls shall ensure that appropriate personnel are aware that the equipment hatch is open, that a specific individual(s) is designated and available to close the equipment hatch cover following a required evacuation of the reactor building, and that any obstruction(s) (e.g., cables and hoses) that could prevent closure of the equipment hatch cover be capable of being quickly removed (Ref. 1). Should a fuel handling accident occur inside the reactor building, the equipment hatch will be closed following evacuation of the reactor building. For closure, the equipment hatch cover will be in place with a minimum of four bolts securing the cover to the sealing surface. During outages, a temporary equipment hatch cover may be used in lieu of the permanent equipment hatch cover (Ref. 2).

The reactor building air locks, which are also part of the reactor building pressure boundary, provide a means for personnel access. During MODES 1, 2, 3, and 4 unit operation is in accordance with LCO 3.6.2, "Reactor Building Air Locks." Each air lock has a door at each end. The doors are normally interlocked to prevent simultaneous opening when the reactor building OPERABILITY is required. During unit shutdown when reactor building OPERABILITY is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods. During the movement of irradiated fuel assemblies within the reactor building, closure requires that one door in each air lock be capable of being closed. The door interlock mechanism may remain disabled.

During the movement of irradiated fuel assemblies within the reactor building, administrative controls shall ensure that appropriate personnel are aware that both personnel airlock doors are open, that a specific individual(s) is designated and available to close an airlock door following a required evacuation of the reactor building, and any obstruction(s) (e.g., cables and hoses) that could prevent closure of an airlock door be capable of being quickly removed (Ref. 3). Should a fuel handling accident occur inside the reactor building, at least one of the personnel and/or emergency air lock doors will be closed following evacuation of the reactor building.

The requirements on reactor building penetration closure ensure that a release of fission product radioactivity from within the reactor building will be restricted to within regulatory limits.

The Reactor Building Purge System includes a supply penetration and exhaust penetration. During MODES 1, 2, 3, and 4, the valves in the supply and exhaust penetrations are secured in the closed position. The system is not subject to a Specification in MODE 5.

In MODE 6, the purge system is used for temperature control, and all four valves may be closed by an operator based on an indication of high radiation. This LCO requires that an OPERABLE radiation monitor be present on the purge exhaust flow path to provide the necessary indication to the operator.

Other reactor building penetrations that provide direct access from the reactor building atmosphere to outside atmosphere must be isolated on at least one side by a closed manual or automatic isolation valve, blind flange, or equivalent, or capable of being isolated by an OPERABLE isolation valve. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, atmospheric pressure ventilation barrier for the other reactor building penetrations during fuel movements.

APPLICABLE SAFETY ANALYSES

During the movement of irradiated fuel assemblies within the reactor building, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 4). The requirements of LCO 3.9.6, "Refueling Canal Water Level," and the minimum decay time of 100 hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity subsequent to a fuel handling accident results in doses that are within the requirements specified in Reference 4.

Reactor building penetrations satisfy Criterion 4 of 10 CFR 50.36 (Ref. 5).

LCO

This LCO limits the consequences of a fuel handling accident in the reactor building by limiting the potential escape paths for fission product radioactivity from the reactor building. The LCO requires any penetration providing direct access from the reactor building atmosphere to the outside atmosphere to be closed or capable of being closed by an OPERABLE reactor building isolation valve. This LCO requires the reactor building purge isolation valves and the purge exhaust flow path radiation monitor be OPERABLE.

The reactor building personnel airlock doors and/or the equipment hatch may be open during movement of irradiated fuel in the reactor building provided that one door is capable of being closed in the event of a fuel handling accident. Administrative controls shall ensure that appropriate personnel are aware that both personnel airlock doors and/or equipment hatch are open, that a specific individual(s) is designated and available to close an airlock door and the equipment hatch cover following a required evacuation of the reactor building, and any obstruction(s) (e.g. cables and hoses) that could prevent closure of an airlock door and the equipment hatch cover be capable of being quickly removed (Ref. 1 and 3). For closure, the equipment hatch cover will be in place with a minimum of four bolts securing the cover to the sealing surface. During outages, a temporary equipment hatch cover may be used in lieu of the permanent equipment hatch cover (Ref. 2).

The definition of "direct access from the reactor building atmosphere to the outside atmosphere" is any path that would allow for the transport of reactor building atmosphere to any atmosphere located outside of the reactor building structure. This includes the Auxiliary Building. As a general rule, closed systems do not constitute a direct path between the reactor building and the outside environments. All permanent and temporary penetration closures should be evaluated to assess the possibility for a release path to the outside environment. For the purpose of determining what constitutes a "direct access" path, no failure mechanisms should be applied to create a scenario which results in a "direct access" path. For example, line breaks, valve failures, power losses or natural phenomenon should not be postulated as part of the evaluation process.

APPLICABILITY

The reactor building penetration requirements are applicable during movement of irradiated fuel assemblies within the reactor building because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, the reactor building penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when movement of irradiated fuel assemblies within the reactor building is not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on reactor building penetration status.

ACTIONS

A.1

With the reactor building equipment hatch, air locks, or any reactor building penetration that provides direct access from the reactor building atmosphere to the outside atmosphere not in the required status, the unit must be placed in a condition in which the isolation function is not needed. This is accomplished by immediately suspending movement of irradiated fuel assemblies within the reactor building. Performance of this action shall not preclude moving a component to a safe position.

These actions remove the potential for an event which may require reactor building closure to prevent a significant radioactivity release.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This Surveillance demonstrates that each of the reactor building penetrations required to be in its closed position is in that position.

The Surveillance is performed every 7 days during the movement of irradiated fuel assemblies within the reactor building. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations.

This Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the reactor building will not result in a release of fission product radioactivity to the environment in excess of that recommended by Standard Review Plan Section 15.7.4 (Ref. 1, 3 and 6).

SR 3.9.3.2

This Surveillance demonstrates that each reactor building isolation valve actuates to its isolation position on manual initiation. The 18 month Frequency maintains consistency with other similar reactor building isolation valve testing requirements found in Section 3.6. This Surveillance will ensure that the isolation valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the reactor building.

The SR is modified by a Note stating that this surveillance is not required to be met for valves in isolated penetrations. The LCO provides the option to close penetrations in lieu of requiring automatic actuation capability.

SR 3.9.3.3

This SR requires a CHANNEL CALIBRATION of the reactor building purge exhaust radiation monitor. The CHANNEL CALIBRATION is a complete check of the instrument loop and sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. The CHANNEL CALIBRATION is performed consistent with the setpoint requirements. The 18 month Frequency is based on operating experience and is consistent with the typical operating cycle.

REFERENCES

1. Safety Evaluation Report related to ANO-1 Amendment No. 195, April 16, 1999.
2. SAR, Section 5.2.2.1.3.
3. Safety Evaluation Report related to ANO-1 Amendment No. 184, September 20, 1996.
4. SAR, Section 14.2.2.3.
5. 10 CFR 50.36.
6. NUREG-0800, Section 15.7.4, Rev. 1, July 1981.

B 3.9 REFUELING OPERATIONS

B 3.9.4 Decay Heat Removal (DHR) and Coolant Circulation - High Water Level

BASES

BACKGROUND

The purposes of the DHR System in MODE 6 are to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34 (Ref. 1), and to provide mixing of the reactor coolant to prevent boron stratification (Ref. 2). Heat is removed from the RCS by circulating reactor coolant through the DHR heat exchanger(s), where the heat is transferred to the Service Water System. The coolant is then returned to the reactor vessel via the core flood tank injection nozzles. Operation of the DHR System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by control of the flow of reactor coolant through the DHR heat exchanger(s), bypassing the heat exchanger(s) and throttling of Service Water through the heat exchanger(s). Mixing of the reactor coolant is provided by the continuous operation of the DHR System.

APPLICABLE SAFETY ANALYSES

Without a DHR loop in operation, the reactor coolant temperature may not be maintained below the boiling point. This could lead to inadequate cooling of the reactor fuel as a result of a loss of coolant in the reactor vessel due to boiling. The loss of reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Operation of one train of the DHR System in MODE 6 is sufficient to prevent this challenge. The LCO does permit de-energizing the DHR pump for short durations under the condition that the boron concentration is not reduced. This conditional de-energizing of the DHR pump does not result in a challenge to the fission product barrier.

The DHR System satisfies Criterion 4 of 10 CFR 50.36 (Ref. 3).

LCO

Only one DHR loop is required for decay heat removal in MODE 6, with a water level ≥ 23 ft above the top of the fuel assemblies seated in the reactor pressure vessel. The operating DHR loop provides:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

To be considered OPERABLE, a DHR loop includes a DHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the temperature. The flow path starts in the 'A' hot leg and is returned to the reactor vessel via the core flood tank injection nozzles.

Additionally, to be considered OPERABLE, each DHR loop must be capable of being manually aligned (remote or local) in the decay heat removal mode.

The LCO is modified by a Note that allows the required DHR loop to be removed from operation for up to 1 hour in an 8 hour period, provided no operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1. Boron concentration reduction with coolant at boron concentrations less than required to assure the RCS boron concentration is maintained is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This allowance permits operations such as core mapping, alterations or maintenance in the vicinity of the reactor vessel nozzles and RCS to DHR isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling canal.

APPLICABILITY

One DHR loop must be OPERABLE and in operation in MODE 6, with the water level ≥ 23 ft above the top of the fuel assemblies seated in the reactor pressure vessel, to provide decay heat removal. Requirements for the DHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). DHR loop requirements in MODE 6, with the water level < 23 feet above the top of the fuel assemblies seated in the reactor vessel, are located in LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation - Low Water Level."

ACTIONS

A.1

If DHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

A.2

If DHR loop requirements are not met, actions shall be taken immediately to suspend the loading of irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling canal water level 23 feet above the fuel assemblies seated in the reactor vessel provides an adequate available heat sink. Suspending any operation that would increase decay heat load, such as loading an irradiated fuel assembly, is prudent under this condition.

A.3

If DHR loop requirements are not met, actions shall be initiated immediately in order to satisfy DHR loop requirements.

Restoration of one decay heat removal loop is required because this is the only active method of removing decay heat. Dissipation of decay heat through natural convection to the large inventory of water in the refueling canal should not be relied upon for an extended period of time. The immediate Completion Time reflects the importance of restoring an adequate decay heat removal loop.

A.4

If DHR loop requirements are not met, all reactor building penetrations providing direct access from the reactor building atmosphere to outside atmosphere shall be closed within 4 hours.

If no means of decay heat removal can be restored, the core decay heat could raise temperatures and cause boiling in the core which could result in increased levels of radioactivity in the reactor building atmosphere. Closure of the penetrations providing access to the outside atmosphere will prevent the uncontrolled release of radioactivity to the environment.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that the DHR loop is in operation and circulating reactor coolant. Verification includes flow, temperature, or pump status monitoring, which help assure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the DHR System.

REFERENCES

1. SAR, Section 1.4.

2. SAR, Section 9.5.

3. 10 CFR 50.36.

B 3.9 REFUELING OPERATIONS

B 3.9.5 Decay Heat Removal (DHR) and Coolant Circulation - Low Water Level

BASES

BACKGROUND

The purposes of the DHR System in MODE 6 are to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34 (Ref. 1), and to provide mixing of the reactor coolant to prevent boron stratification (Ref. 2). Heat is removed from the RCS by circulating reactor coolant through the DHR heat exchanger(s), where the heat is transferred to the Service Water System. The coolant is then returned to the reactor vessel via the core flood tank injection nozzles. Operation of the DHR System for normal cooldown/decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by control of the flow of reactor coolant through the DHR heat exchanger(s), bypassing the heat exchanger(s) and by throttling of Service Water through the heat exchanger(s). Mixing of the reactor coolant is provided by the continuous operation of the DHR System.

APPLICABLE SAFETY ANALYSES

Without a DHR loop in operation, the reactor coolant temperature may not be maintained below the boiling point. This could lead to inadequate cooling of the reactor fuel as a result of a loss of coolant in the reactor vessel due to boiling. The loss of reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Operation of one train of the DHR System in MODE 6 is sufficient to prevent this challenge. However, without a large water inventory to provide a backup means of decay heat removal, an additional train of the DHR System is required to be OPERABLE in order to provide a backup.

The DHR System satisfies Criterion 4 of 10 CFR 50.36 (Ref. 3).

LCO

In MODE 6, with the water level < 23 feet above the top of the fuel seated in the reactor vessel, two DHR loops must be OPERABLE. Additionally, one DHR loop must be in operation to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

This LCO is modified by two Notes. Note 1 permits the DHR pumps to be de-energized for ≤ 15 minutes when switching from one train to another. The circumstances for stopping both DHR pumps are to be limited to situations when the outage time is short and the core outlet temperature is maintained > 10 degrees F below saturation temperature. The Note prohibits boron dilution or draining operations when DHR forced flow is stopped.

The second Note allows one DHR loop to be inoperable for a period of 2 hours provided the other loop is OPERABLE and in operation. Prior to declaring the loop inoperable, consideration should be given to the existing plant configuration. This consideration should include that the core time to boil is short, there is no draining operation to further reduce RCS water level and that capability exists to inject borated water into the reactor vessel. This permits surveillance tests to be performed on the inoperable loop during a time when these tests are safe and possible.

To be considered OPERABLE, a DHR loop must consist of a DHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the temperature. The flow path starts in the 'A' hot leg and is returned to the reactor vessel via the core flood tank injection nozzles.

Additionally, to be considered OPERABLE, each DHR loop must be capable of being manually aligned (remote or local) in the decay heat removal mode for removal of decay heat. Operation of one subsystem can maintain the reactor coolant temperature as required.

Both DHR pumps may be aligned to the Borated Water Storage Tank (BWST) to support filling of the refueling canal or the performance of required testing.

APPLICABILITY

Two DHR loops are required to be OPERABLE, and one in operation in MODE 6, with the water level < 23 feet above the top of the fuel seated in the reactor vessel, to provide decay heat removal. Requirements for the DHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). DHR loop requirements in MODE 6 are located in LCO 3.9.4.

ACTIONS

A.1 and A.2

With fewer than the required loops OPERABLE, action shall be immediately initiated and continued until the DHR loop is restored to OPERABLE status or until ≥ 23 feet of water level is established above the fuel seated in the reactor vessel. When the water level is established at ≥ 23 feet above the fuel seated in the reactor vessel, the Applicability will change to that of LCO 3.9.4, and only one DHR loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary due to the increased risk of operating without a large available heat sink.

B.1

If no DHR loop is in operation or no DHR loop is OPERABLE, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

B.2

If no DHR loop is in operation or no DHR loop is OPERABLE, actions shall be initiated immediately and continued without interruption to restore one DHR loop to OPERABLE status and operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE DHR loops and one operating DHR loop should be accomplished expeditiously.

If no DHR loop is OPERABLE or in operation, alternate actions shall have been initiated immediately under Condition A to establish ≥ 23 ft of water above the top of fuel assemblies seated in the reactor vessel. Furthermore, when the LCO cannot be fulfilled, alternate decay heat removal methods, as specified in the unit's Abnormal and Emergency Operating Procedures, should be implemented. This includes decay heat removal using the charging or safety injection pumps through the Chemical and Volume Control System with consideration for the boron concentration. The method used to remove decay heat should be the most prudent as well as the safest choice, based upon unit conditions. The choice could be different if the reactor vessel head is in place rather than removed.

B.3

If no DHR loop is in operation, all reactor building penetrations providing direct access from the reactor building atmosphere to the outside atmosphere must be closed within 4 hours. With the DHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the reactor building atmosphere. Closing reactor building penetrations that are open to the outside atmosphere ensures that dose limits are not exceeded.

The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This Surveillance demonstrates that one DHR loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help assure that forced flow is providing heat removal.

The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator to monitor the DHR system in the control room.

SR 3.9.5.2

Verification that each required pump is available ensures that an additional DHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. Alternatively, verification that a DHR pump is in operation as required by SR 3.9.4.1 also verifies proper breaker alignment and power availability. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. SAR, Section 1.4.
2. SAR, Section 9.5.
3. 10 CFR 50.36.

B 3.9 REFUELING OPERATIONS

B 3.9.6 Refueling Canal Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies within the reactor building requires a minimum water level of 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel. During refueling, this maintains sufficient water level to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident within 10 CFR 100 limits, as provided by the guidance of Reference 3.

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of the fuel handling accident in the reactor building postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 12% of the total fuel rod iodine inventory (Ref. 2).

The fuel handling accident analysis inside the reactor building is described in Reference 2. With a minimum water level of 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel, and a minimum decay time of 100 hours prior to fuel handling, the analysis demonstrates that the iodine release due to a postulated fuel handling accident is adequately captured by the water, and offsite doses are maintained within allowable limits (Ref. 3).

Refueling canal water level satisfies Criterion 2 of 10 CFR 50.36 (Ref. 4).

LCO

A minimum refueling canal water level of 23 feet above the top of the irradiated fuel assemblies seated in the reactor pressure vessel is required to ensure that the radiological consequences of a postulated fuel handling accident inside the reactor building are within acceptable limits as provided by 10 CFR 100.

APPLICABILITY

LCO 3.9.6 is applicable during movement of irradiated fuel assemblies within the reactor building. The LCO minimizes the possibility of a fuel handling accident in the reactor building that is beyond the assumptions of the safety analysis. If irradiated fuel is not present in the reactor building, there can be no significant radioactivity release as a result of a postulated fuel handling accident in the reactor building.

ACTIONS

A.1

With a water level of < 23 feet above the top of the irradiated fuel assemblies seated with the reactor pressure vessel, all operations involving the movement of irradiated fuel assemblies shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of irradiated fuel movement shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. Water at the required level above the top of the irradiated fuel assemblies seated within the reactor pressure vessel limits the consequences of damaged fuel rods that are postulated to result from a postulated fuel handling accident inside the reactor building (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls, which make significant unplanned level changes unlikely.

REFERENCES

1. Regulatory Guide 1.25, March 23, 1972.

2. SAR Section 14.2.2.3.

3. 10 CFR 100.10.

4. 10 CFR 50.36.

ADMINISTRATIVE

A1 The designated change represents a non-technical, non-intent change to the Arkansas Nuclear One, Unit 1 Current Technical Specifications (CTS) made to make the ANO-1 Improved Technical Specifications (ITS) consistent with the B&W Standard Technical Specification, NUREG-1430, Revision 1. This change does not alter the requirements of the CTS or RSTS. Examples of this type of change include: wording preference; convention adoption; editorial, numbering and formatting changes; and hierarchy structure.

A2 The ANO-1 CTS Bases will be administratively deleted in their entirety in favor of the NUREG-1430 Bases. The CTS Bases will be reviewed for technical content that will be identified for retention in the ITS Bases.

A3 CTS 3.8.9 provides the required actions should one or more of the preceding Specifications not be met. CTS 3.8.9 establishes measures that are considered equivalent to the Required Actions of ITS 3.9.1 Condition A, ITS 3.9.2 Condition A and Required Action B.1, and ITS 3.9.3 Condition A. Although the exact wording is not the same, these are considered equivalent actions and adoption of the ITS requirements constitutes an administrative change. In addition, the Completion Time of "immediately" has been annotated on the CTS markup. This is implicit in a number of CTS actions and explicit in other CTS actions. The addition of this immediate Completion Time establishes Required Actions consistent with those specified in the ITS.

A4 The CTS 3.8.3.a Note * to allow the decay heat removal loop to be secured for periods up to 1 hour per 8 hour period was modified to reflect the exact wording of the ITS LCO 3.9.4 Note. The modification of the CTS 3.8.3.a Note * involved two changes that are both considered administrative in nature.

The first change added words that state that reactor coolant boron concentration reductions are not allowed during the period of time associated with the secured decay heat removal loop. This is consistent with the CTS (per CTS 3.1.1.1.B) which permits boron concentration reductions only when at least one decay heat removal pump is circulating reactor coolant. This requirement is implicitly retained in the ITS through 3.9.4 Required Action A.1 which directs that operations involving a reduction of the reactor coolant boron concentration be immediately suspended should the required reactor coolant circulation not be present, and is explicitly established in the LCO Bases for 3.9.4.

The second change involved the deletion of the words that restricted the applicability of this Note to "during the performance of core alterations." The allowance to secure the decay heat removal loop for a limited period of time in the CTS was dependent upon the availability of a backup source of decay heat removal because the Note modified the decay heat loop OPERABILITY requirements when reactor coolant level was greater than 23 feet above the fuel seated in the reactor pressure vessel. This restriction is inherently present in the ITS through the structure of the Applicability statements for LCOs 3.9.4 and 3.9.5 and the presence of the Note in LCO 3.9.4.

A5 CTS 3.8.9 and 3.8.10 state that the provisions of CTS 3.0.3 are not applicable. This exception is necessary in the CTS because of the concurrent use of CTS 3.8.9 as the Required Actions and associated Completion Times for a number of CTS Specifications (CTS 3.8.1 through CTS 3.8.8), several of which are MODE independent. The ITS 3.9, "REFUELING OPERATIONS" series of specifications will contain appropriate MODES, Applicabilities, Conditions and Surveillance Requirements such that the exception to LCO 3.0.3 will no longer be necessary. Further, the LCO 3.0.3 exception is unnecessary for the ITS 3.9 series of specifications because LCO 3.0.3 does not apply in MODES 5 and 6. This change is classified as administrative because the operating flexibility employed by the CTS 3.0.3 exception is inherent in the structure of the ITS.

A6 The CTS markup was annotated to show adoption of ITS LCO 3.9.4 Applicability. ITS LCO 3.9.4 is comparable to CTS 3.8.3.a. However, the CTS did not explicitly establish an Applicability for this Specification. This is considered an administrative change because the intended Applicability for the CTS was during refueling activities which corresponds to MODE 6 in the ITS. In addition, CTS 3.8.3.b established LCO requirements comparable to those stated by ITS 3.9.5 (i.e., DHR requirements when less than 23 feet of water covered the irradiated fuel). Because CTS 3.8.3.b established LCO requirements when the water level was less than 23 feet above the fuel, it is implied that CTS 3.8.3.a had an Applicability when the water level was greater than 23 feet above the fuel. Based on this reasoning, the adoption of the ITS 3.9.4 Applicability is administrative.

A7 ITS 3.9.5 Required Action A.2 is shown as being adopted on the CTS markup. This Required Action is an alternative to A.1 which requires restoration of the inoperable DHR loop. Required Action A.2 serves to remove the unit from the MODE of Applicability. This is cited as an Administrative change because this action (i.e., removing the unit from the Applicability) was available as an option in the CTS although not explicitly written as a Required Action. This change is consistent with NUREG-1430.

A8 CTS 3.8.3.a was annotated to show the explicit Completion Time of "immediately" for the ITS Required Actions that reference CTS 3.8.3.a. This is shown as an administrative adoption because the assigned Completion Time is consistent with other CTS required actions in this series of Specifications. This change is consistent with NUREG-1430.

ANO-251

A9 ITS 3.9.1 Applicability Note is shown as being adopted on the CTS markup. This Note was incorporated in NUREG-1430 as a result of TSTF-272, Rev 1, and requires that boron concentration is only applicable to the refueling canal when connected to the RCS. The CTS does not specifically state whether the requirements for boron concentration must be maintained only when the refueling canal is connected to the RCS. However, CTS 3.8.4 does state that the boron concentration must be met during fuel loading and unloading. Since these activities can only be performed with the refueling canal connected to the RCS, the incorporation of this change is consistent with the current license basis.

A10 Not used.

TECHNICAL CHANGE -- MORE RESTRICTIVE

3.9-07

M1 The CTS markup was annotated to show adoption of NUREG-1430 SR 3.9.5.2 (ITS SR 3.9.5.2) which requires verification of correct breaker alignment and indicated power availability to the required DHR pump that is not in operation with a Frequency of 7 days. This SR verifies the availability of the non-operating DHR loop required when the reactor coolant level is less than 23 feet above the top of the fuel seated in the reactor pressure vessel. The adoption of this ITS SR results in additional operational requirements or constraints beyond those imposed by the CTS. This change is consistent with NUREG-1430.

M2 Not used.

3.9-07

M3 The last paragraph of CTS 3.8.3 established the last of the required actions for CTS 3.8.3.a and 3.8.3.b. This paragraph is connected to the previous paragraphs with an "otherwise" which would imply this to be an alternative to the previous required actions. The CTS action established by this paragraph will be connected to the equivalent ITS Required Actions with an "and." This conjunction will eliminate the apparent alternative that is present in the CTS. Thus, the ITS Required Actions (3.9.4 RA A.3, 3.9.5 RA A.1 and 3.95 RA B.2) that reference this specification will be more restrictive than the CTS. This change is consistent with NUREG-1430.

M4 CTS 3.8.4 established the requirement for minimum boron concentration during "reactor vessel head removal and while loading and unloading fuel from the reactor." The Applicability for ITS LCO 3.9.1 will be MODE 6. MODE 6 is entered with the detensioning of the first reactor vessel head stud and will be in effect as long as fuel is in the vessel until the last reactor vessel head stud is retensioned. Thus, the Applicability of ITS LCO 3.9.1 will be more inclusive and more restrictive than the requirements of the CTS because it includes the period of time associated with vessel head reinstallation. This change is consistent with NUREG-1430.

M5 The CTS markup was annotated to show the adoption of ITS LCO 3.9.2 Required Action B.2. ITS 3.9.2 Condition B establishes the Required Actions should both of the required source range neutron flux monitors become inoperable. Required Action B.1 is established by CTS 3.8.9. ITS 3.9.2 Required Action B.2 requires performance of SR 3.9.1.1 with a Completion Time of once per 12 hours. ITS SR 3.9.1.1 verifies that the boron concentration of the RCS, refueling canal and refueling cavity is within its limits. No comparable CTS required action exists. Therefore, through the adoption of ITS 3.9.2 Required Action B.2, the ITS will impose an additional restriction on the unit. The adoption of ITS 3.9.2 Required Action B.2, in conjunction with the current requirements of ITS 3.9.2 Condition A and Required Action B.1, ensures that the core's reactivity condition is not changing during the period when no OPERABLE source range nuclear instrument is available for the detection of changes in core reactivity. This change is consistent with NUREG-1430.

M6 The CTS markup was annotated to show the adoption of ITS SR 3.9.2.1, SR 3.9.2.2 and the SR 3.9.2.2 Note. SR 3.9.2.1 established requirements for a CHANNEL CHECK every 12 hours. SR 3.9.2.2 established requirements that a CHANNEL CALIBRATION be performed every 18 months. The SR 3.9.2.2 Note excludes the neutron detectors from the CHANNEL CALIBRATION requirements because of the inability to calibrate these detectors. The ANO-1 CTS did not include similar surveillance requirements in this MODE of Applicability. Therefore, the ITS will impose additional restrictions on the unit. These SRs are necessary because they serve to demonstrate the functional capability of the source range nuclear instruments to respond to changes in core conditions. This change is consistent with NUREG-1430.

M7 The CTS markup was annotated to show adoption of ITS SR 3.9.3.2 and its associated Note. SR 3.9.3.2 requires verification that each required reactor building isolation valve and each reactor building purge isolation valve can actuate to the isolation position with a Frequency of 18 months. This SR demonstrates that each of the reactor building isolation valves are capable of being placed in its closed position. The 18 month surveillance Frequency is commensurate with the normal duration of an operating cycle. The SR Note is administrative in nature in that it establishes that the application of this SR requirement does not apply to valves that have been closed in accordance with ITS LCO 3.9.3.c.1. The CTS does not presently contain such a Surveillance Requirement. Thus, the adoption of this SR results in the ITS being more restrictive than the CTS. This change is consistent with the NUREG-1430.

M8 The CTS markup was annotated to show adoption of ITS SR 3.9.3.1. SR 3.9.3.1 requires verification that each required reactor building penetration is in the required status with a Frequency of 7 days. This SR demonstrates that each of the reactor building penetrations required to be in its closed position is in that position. The 7 day surveillance Frequency is commensurate with the normal duration of fuel handling activities during a refueling. The CTS does not presently contain such a

Surveillance Requirement. Thus, the adoption of this SR results in the ITS being more restrictive than the CTS. This change is consistent with the NUREG-1430.

M9 CTS 3.8.10 established the LCO requirements for the reactor building purge isolation system. These requirements are comparable to the LCO requirements of NUREG-1430 3.9.3. However, the CTS does not establish specific required actions or associated completion times should the LCO not be satisfied. ITS 3.9.3 Condition A will establish the Required Actions and associated Completion Times for this LCO in the ITS. The Required Actions remove the unit from the LCO Applicability and eliminate the possibility of fuel handling accident during the period of the inoperable reactor building purge isolation valve(s). The CTS markup was annotated to show ITS 3.9.3 Action A as correlated to CTS 3.8.9 because its contains the intended ITS Actions. This really constitutes the adoption of the ITS Required Actions and Completion Times for Condition A when applied to CTS 3.8.10 LCO requirements. The imposition of the Actions for CTS 3.8.10 will establish additional restrictions that are not present in the CTS. The establishment of Required Actions and associated Completion Times for inoperability of the reactor building purge isolation valves is consistent with NUREG-1430.

M10 The CTS markup was annotated to show adoption of ITS SR 3.9.6.1. SR 3.9.6.1 requires verification that the refueling canal level is greater than or equal to 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel. This SR demonstrates that the Fuel Handling Accident analysis initial condition assumptions regarding the refueling canal level are satisfied during the movement of irradiated fuel assemblies within the reactor building. The 24 hour surveillance Frequency is considered appropriate in view of the large volume of water and the normal procedural controls in place during fuel handling activities. The CTS does not presently contain such a Surveillance Requirement. Thus, the adoption of this SR results in the ITS being more restrictive than the CTS.

M11 Not used.

M12 CTS Table 4.1-3 is annotated to show the NUREG-1430 SR 3.9.1.1 Frequency of 72 hours. The adoption of the 72 hour Frequency reduces the degree of scheduling freedom present in CTS Table 4.1-3 Item 1.f, Boron Concentration, sampling frequency of 3 times per week. This CTS frequency does not stipulate that the samples obtained at approximately equal intervals. The ITS 72 hour Frequency imposes a more structured requirement with specific sampling intervals that are not as flexible as the CTS Frequency. The adoption of this Frequency establishes requirements that are consistent with NUREG-1430.

M13 CTS 3.8.10 is annotated to show its correlation to ITS SR 3.9.3.3 which specifies a Frequency of 18 months. The 18 month surveillance Frequency is consistent with the refueling frequency when this SR can be performed. Because the CTS established the Frequency based on a time commensurate with refueling activities, the imposition of a fixed 18 month increment will be more restrictive than CTS requirements. In addition, the CTS simply required that the radiation monitors be tested and verified to be OPERABLE. The ITS will specify that this is accomplished by a CHANNEL CALIBRATION. This change is consistent with the NUREG-1430.

TECHNICAL CHANGE -- LESS RESTRICTIVE

L1 CTS 3.8.7 requires that isolation valves in lines containing automatic containment isolation valves be OPERABLE, or at least one shall be closed. ITS 3.9.3.c requires that each penetration providing direct access from the reactor building atmosphere to the outside atmosphere be 1) closed by a manual valve or automatic isolation valve, blind flange, or equivalent, or 2) be capable of being closed by an OPERABLE isolation valve. CTS 3.8.7 requires containment closure capability of components in fluid systems that are ordinarily incapable of releasing radioactive material from the reactor building atmosphere to the outside atmosphere because they are not exposed to the reactor building atmosphere (i.e. the system is intact). ITS 3.9.3 will only apply to those penetrations providing direct access from the reactor building atmosphere to the outside atmosphere. Thus, the scope of the penetrations requiring closure by a manual or power operated isolation valve will be reduced. However, the reduction in scope of penetrations subject to the closure specification will not appreciably change the protective nature of the reactor building. This is because fluid systems that are not open to the reactor building atmosphere have never been a credible release path. Only those penetrations that allow reactor building atmosphere release to the environment are credible offsite dose contributors. Therefore, the reduction in the scope of reactor building penetrations requiring closure still results in the same level of protection for a member of the public. This change is consistent with NUREG-1430.

L2 CTS Table 4.1-3, Item 1.f required the determination of the RCS boron concentration with a Frequency of "3 times per week." The CTS did not establish that these samples were to be obtained on an equal interval. But if they were drawn at equal intervals, the interval would equate to three equal increments of 56 hours each. NUREG-1430 SR 3.9.1.1 specifies a Frequency of 72 hours. The ITS will retain the NUREG Frequency for this SR. This results in the SR being performed less frequently. The less frequent determination of the RCS boron concentration is acceptable based on: 1) administrative actions taken to prevent boron dilution events, 2) the relatively large inventory present during much of the time spent in MODE 6, and 3) historical experience associated with boron concentration changes during refueling conditions. This change is consistent with NUREG-1430.

L3 CTS 3.8.10 requires that the reactor building purge isolation valves "be tested and verified to be operable within 7 days prior to refueling operations." The ITS equivalent Surveillance Requirement is SR 3.9.3.2 which will have a Frequency of 18 months. This can be less restrictive than CTS requirements: 1) if refueling activities should occur on a more frequent or unexpected basis, or 2) if the SR is performed at a time other than refueling which would reestablish the SR interval such that it overlapped refueling activities; thus, avoiding the performance of this SR prior to the subsequent refueling activities. This change is consistent with NUREG-1430.

L4 CTS 3.6.2 established a requirement that reactor building integrity be maintained when the reactor coolant system (RCS) is open to the reactor building atmosphere and the requirements for a refueling shutdown are not met. When combined with the definition of a refueling shutdown (CTS 1.2.6), this establishes a conditional requirement that only exists when the RCS is open to the reactor building atmosphere and the degree of subcriticality is less than 1% $\Delta K/K$ assuming all rods are removed from the core. This reactivity condition is prohibited in the ITS through the imposition of a SHUTDOWN MARGIN requirement in MODE 5 (ITS 3.1.1) and imposition of a required degree of subcriticality ($K_{eff} \leq 0.99$) in MODE 6 (ITS 3.9.1). In both of these ITS Specifications, the Required Actions will be to restore the required SHUTDOWN MARGIN or degree of subcriticality, and while in MODE 6, terminate those activities that may result in the possibility of fission product release to the reactor building atmosphere or otherwise affect the core reactivity condition, for example, CORE ALTERATIONS. Thus, the ITS will be less restrictive than the CTS in that reactor building integrity will not have to be established as a direct result of a loss of SHUTDOWN MARGIN or degree of subcriticality. This change is acceptable because the ITS will direct actions to restore the required SHUTDOWN MARGIN or degree of subcriticality which are not present in the CTS. This change is consistent with NUREG-1430.

L5 CTS 3.8.3 established specific LCO requirements and explicit required actions for Decay Heat Removal. In addition, CTS 3.8.9 established a generic set of required actions for all of the preceding CTS 3.8 series of LCO requirements. CTS 3.8.3 directed that the operator "suspend all operations involving an increase in the reactor decay heat load." CTS 3.8.9 directed that "movement of the fuel into the reactor core shall cease." These actions correspond to ITS 3.9.4 Required Action A.2 which directs the operator to "suspend loading of irradiated fuel assemblies in the core." The ITS will be less restrictive than the CTS 3.8.9 requirements in that it would allow the continued introduction of non-irradiated fuel assemblies. ITS 3.9.4 Required Action A.2 is appropriate because it addresses the unavailability of a decay heat removal system to dissipate the decay heat being generated by the irradiated fuel assemblies within the reactor vessel. Non-irradiated fuel assemblies would not contribute to an increased decay heat load within the reactor vessel. This change is consistent with NUREG-1430.

ANO-244

L6 The CTS 3.8.3.b requirements are revised to allow the DHR pumps to be de-energized for ≤ 15 minutes when switching from one train to another. The addition of this allowance (LCO 3.9.5 Note 1) is acceptable since additional restrictions on application of the allowance are provided by the LCO Note. The circumstances for stopping both DHR pumps are to be limited to situations when the outage time is short and the core outlet temperature is maintained > 10 degrees F below saturation temperature. The Note prohibits boron dilution or draining operations when DHR forced flow is stopped. This change is consistent with NUREG-1430, as modified by generic change TSTF-349, Rev 1.

ANO-245

L7 The CTS 3.8.3.b requirements are revised to allow one DHR loop to be inoperable for a period of 2 hours provided the other loop is OPERABLE and in operation. The purpose of this allowance is to allow for proper surveillance testing of the DHR systems. The addition of this allowance (LCO 3.9.5 Note 2) is acceptable since its use requires consideration that the core time to boil is short, there is no draining operation to further reduce RCS water level and that capability exists to inject borated water into the reactor vessel. This change is consistent with NUREG-1430, as modified by generic change TSTF-361, Rev 2.

ANO-243

3.9-07

L8 The CTS 3.1.1.1.B, 3.8.3.a and associated footnote, 3.8.3.b, and 3.8.9 requirements are revised to allow operations that may result in a limited addition of positive reactivity in the event one source range monitor is inoperable, or DHR flow is not available. During these conditions, various unit operations must be continued. RCS inventory must be maintained, and RCS temperature must be controlled. These activities necessarily involve additions to the RCS of cooler water (a positive reactivity effect in most cases) and may involve inventory makeup from sources that are at boron concentrations that are less than the RCS boron concentration. The addition of this allowance (LCO 3.9.2 R.A. A.2, 3.9.4 LCO Note, 3.9.4 R.A. A.1, and 3.9.5 R.A. B.1) is acceptable, since controls are maintained to provide assurance that the minimum boron concentration, and thus a minimum SDM, is maintained as specified in the COLR. This change is consistent with NUREG-1430, as modified by generic change TSTF-286, Rev 2.

LESS RESTRICTIVE -- ADMINISTRATIVE DELETION OF REQUIREMENTS

LA1 This information has been moved to the Bases or TRM. This information provides details of design or process which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe an acceptable method of compliance. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The Bases will be controlled by the Bases Control Process in Chapter 5 of the proposed Technical Specifications. The details of performance of the surveillances have been relocated to the TRM. Changes to the TRM will be controlled by 10 CFR 50.59. This change is consistent with NUREG-1430.

CTS Location	New Location
3.1.1.1.B	Bases, 3.9.4 & 3.9.5 LCO
3.8.2	TRM
3.8.6 Note *	Bases, 3.9.3, Background, LCO

LA2 CTS 3.8.11 is being relocated to the TRM. This Specification places restrictions on the removal of irradiated fuel from the reactor to ensure that sufficient time will elapse to allow the radioactive decay of short-lived fission products.

Although the Specification satisfied Criterion 2 of 10 CFR 50.36, the time to perform necessary activities prior to commencing movement of irradiated fuel ensures that there will normally be greater than 100 hours of subcriticality before any movement of irradiated fuel. Hence, the Specification is relocated in accordance with a prior industry/NRC agreement in the generic split report. Changes to the TRM are controlled under 10 CFR 50.59. This change is consistent with NUREG-1430.

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the reactor coolant system.

Objective

To specify those limiting conditions for operation of the reactor coolant system which must be met to ensure safe reactor operations.

<LATER> (3.4A) — LATER

3.1.1 Operational Components

Specification

3.1.1.1 Reactor Coolant Pumps

A. Pump combinations permissible for given power levels shall be as shown in Table 2.3-1. Operation with one Reactor Coolant Pump operating in each loop is limited to 24 hours with the reactor critical.

AMD 243

& <LATER> (3.4A)

3.9.4 RA A.1

LA1 Bases

LATER

L8

B. The boron concentration in the reactor coolant system shall not be reduced unless at least one reactor coolant pump or one decay heat removal pump is circulating reactor coolant. With no ~~reactor coolant pumps or~~ decay heat removal pumps running, immediately suspend all operations involving a reduction of boron concentration in the reactor coolant system that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1

<LATER> (3.4A) — LATER

3.1.1.2 Steam Generator

A. Two steam generators shall be operable whenever the reactor coolant average temperature is above 280°F.

<LATER> (3.4B) — LATER

3.1.1.3 Pressurizer Safety Valves

A. Both pressurizer code safety valves shall be operable when the reactor is critical. With one pressurizer code safety valve inoperable, either restore the valve to operable status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

B. When the reactor is subcritical, at least one pressurizer code safety valve shall be operable if all reactor coolant system openings are closed, except for hydrostatic tests in accordance with ASME Boiler and Pressure Vessel Code, Section III. The provisions of Specification 3.0.3 are not applicable.

<LATER> (3.4A) — LATER

3.1.1.4 Reactor Internals Vent Valves

The structural integrity and operability of the reactor internals vent valves shall be maintained at a level consistent with the acceptance criteria in Specification 4.1. The provisions of Specification 3.0.3 are not applicable.

<LATER> (3.4A) — LATER

3.1.1.5 Reactor Coolant Loops

A. With the reactor coolant average temperature above 280°F, the reactor coolant loops listed below shall be operable:

3.6 REACTOR BUILDING

<LATER>
(3.6)

Applicability

Applies to the operability of the reactor building.

Objective

To assure reactor building operability.

Specification

3.6.1 The reactor building shall be operable whenever all three (3) of the following conditions exist: —LATER

a. Reactor coolant pressure is 300 psig or greater.

b. Reactor coolant temperature is 200°F or greater.

c. Nuclear fuel is in the core.

With the reactor building inoperable, restore the reactor building to operable status within one hour or be in at least Hot Standby within the next 6 hours and in Cold Shutdown within the following 30 hours.

3.6.2 Reactor building integrity shall be maintained when the reactor coolant system is open to the reactor building atmosphere and the requirements for a refueling shutdown are not met. The provisions of Specification 3.0.3 are not applicable. —L4

3.6.3 Positive reactivity insertions which would result in the reactor being subcritical by less than 1% Δk/k shall not be made by control rod motion or boron dilution whenever reactor building integrity is not in force. The provisions of Specification 3.0.3 are not applicable.

<LATER>
(3.6)

3.6.4 The reactor shall not be taken critical or remain critical if the reactor building internal pressure exceeds 3.0 psig or a vacuum of 5.5 inches Hg. With the reactor critical, restore the containment pressure to within its limits within one hour or be in at least Hot Standby within the next 6 hours and in Cold Shutdown within the following 30 hours. —LATER

3.6.5 Prior to criticality following a refueling shutdown, a check shall be made to confirm that all manual reactor building isolation valves which should be closed are closed and locked, as required. The provisions of Specification 3.0.3 are not applicable.

ANO-251

<Add 3.9.1 Applicability Note> A9
<Add 3.9.5 RA A.2> A7
<Add SR 3.9.2.1> M6
<Add SR 3.9.2.2 with Note> M6
<Add 3.9.2 RA B.2> M5
<Add 3.9.4 Appl> A6

3.9.1
3.9.2
3.9.4
3.9.5

3.8 ~~FUEL LOADING AND REFUELING~~

~~Applicability~~

~~Applies to fuel loading and refueling operations.~~

~~Objective~~

~~To assure that fuel loading, refueling and fuel handling operations are performed in a responsible manner.~~ A1

~~Specification~~

3.8.1 ~~Radiation levels in the reactor building refueling area shall be monitored by instrument RE-8017. Radiation levels in the spent fuel storage area shall be monitored by instrument RE-8009. If any of these instruments become inoperable, portable survey instrumentation, having the appropriate ranges and sensitivity to fully protect individuals involved in refueling operation, shall be used until the permanent instrumentation is returned to service.~~ R TRM

3.8.2 Core subcritical neutron flux shall be continuously monitored by at least two neutron flux monitors, ~~each with continuous indication available,~~ whenever core geometry is being changed. ~~When core geometry is not being changed,~~ at least one neutron flux monitor shall be in service. LA1 TRM A1

3.9.2 LCO b
3.9.2 Appl — MODE 6
3.9.2 LCO a

ANO-243

See <CTS INSERT 58>

3.8.3.a. At least one decay heat removal loop shall be in operation.* Otherwise, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the reactor coolant system, and close all containment penetrations providing access from the containment atmosphere to the outside atmosphere within 4 hours.

b. When the water level above the top of the irradiated fuel assemblies seated within the reactor pressure vessel is less than 23 feet, two decay heat removal loops shall be operable.**

Otherwise, immediately initiate corrective action to return the required loops to operable status as soon as possible.

3.8.4 ~~During reactor vessel head removal and while loading and unloading fuel from the reactor,~~ MODE 6 the boron concentration shall be maintained at not less than that required ~~for refueling shutdown~~ in the COLR. M4

3.9.1 Appl
3.9.1 LCO

3.8.5 ~~Direct communications between the control room and the refueling personnel in the reactor building shall exist whenever changes in core geometry are taking place.~~ R TRM

3.9.4 LCO Note — *The decay heat removal loop may be removed from operation for up to 1 hour per 8 hour period ~~during the performance of core alterations.~~ A4 provided no operations are permitted that would cause reduction in RCS boron concentration.

<LATER> (3.8) — **The normal or emergency power source may be inoperable for each shutdown cooling loop. — LATER

ANO-245
ANO-244

<Add 3.9.5 LCO Note 1> L6
<Add 3.9.5 LCO Note 2> L7

3.9.4
3.9.5

<CTS INSERT 58>

3.9.4 LCO Note

3.9.4 LCO
3.9.5 LCO

3.9.4 RA. A.2
3.9.4 RA A.1
3.9.5 RA B.1

3.9.4 RA. A.4
3.9.5 RA B.3

3.8.3.a. ~~At least~~ one decay heat removal loop shall be in operation.* Otherwise, suspend ~~all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the reactor coolant system,~~ and close all containment penetrations providing access from the containment atmosphere to the outside atmosphere within 4 hours.

A8

immediately

L8

3.9.5 Appl

3.9.5 LCO

b. When the water level above the top of the irradiated fuel assemblies seated within the reactor pressure vessel is less than 23 feet, two decay heat removal loops shall be operable.*

LATER (3.8)

3.9.4 RA A.3
3.9.5 RA A.1
3.9.5 RA B.2

Otherwise, immediately initiate corrective action to return the required loops to operable status ~~as soon as possible.~~

M3

A1

Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.

AND-243

3.9.1
3.9.2
3.9.3
3.9.6

<ADD SR 3.9.3.2 with Note> M7
<ADD SR 3.9.6.1> M10
<ADD SR 3.9.3.1> M8

3.9.6 APPL
3.9.3 APPL

3.9.3 LCO b
3.9.3 LCO a
3.9.6 LCO

3.8.6 During the handling of irradiated fuel in the reactor building, at least one door on the personnel and emergency hatches shall be capable* of being closed. The equipment hatch cover shall also be capable* of being closed. At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated within the reactor pressure vessel. LA1 Bases

3.9.3 LCO c.1
3.9.3 LCO c.2

3.8.7 Isolation valves in lines containing automatic containment isolation valves shall be operable, or at least one shall be closed. L1

ANO-243

3.8.8 When two irradiated fuel assemblies are being moved simultaneously by the bridges within the fuel transfer canal, a minimum of 10 feet separation shall be maintained between the assemblies at all times. R TRM

See <CTS INSERT 59>

3.9.1 Cond A
3.9.2 Cond A
3.9.2 RA B.1
3.9.6 Cond A
3.9.3 Cond A

3.8.9 If any of the above specified limiting conditions for fuel loading and refueling are not met, movement of fuel into the reactor core shall cease; action shall be initiated immediately to correct the conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be made. The provisions of Specification 3.0.3 are not applicable. A3 + R TRM A5

3.9.3 LCO c.3
SR 3.9.3.3

3.8.10 The reactor building purge isolation system, including the radiation monitors shall be tested and verified to be operable ~~within 7 days prior to refueling operations~~ every 18 months. The provisions of Specification 3.0.3 are not applicable. M9 A5 L3 M13

3.8.11 ~~Irradiated fuel shall not be removed from the reactor until the unit has been subcritical for at least 100 hours. In the event of a complete core offload, a full core to be discharged shall be subcritical a minimum of 175 hours prior to discharge of more than 70 assemblies to the spent fuel pool. The provisions of Specification 3.0.3 are not applicable.~~ LA2 TRM

3.8.12 ~~All fuel handling in the Auxiliary Building shall cease upon notification of the issuance of a tornado watch for Pope, Yell, Johnson, or Logan counties in Arkansas. Fuel handling operations in progress will be completed to the extent necessary to place the fuel handling bridge and crane in their normal parked and locked position. The provisions of Specification 3.0.3 are not applicable.~~ R TRM

3.8.13 ~~No loaded spent fuel shipping cask shall be carried above or into the Auxiliary Building equipment shaft unless atmospheric dispersion conditions are equal to or better than those produced by Pasquill Type D stability accompanied by a wind velocity of 2 m/sec. In addition, the railroad spur door of the Turbine Building shall be closed and the fuel handling area ventilation system shall be in operation. The provisions of Specification 3.0.3 are not applicable.~~ R TRM

3.8.14 ~~Loads in excess of 2000 pounds shall be prohibited from travel over fuel assemblies in the storage pool. The provisions of Specification 3.0.3 are not applicable.~~ R TRM

* ~~Administrative controls shall ensure that appropriate personnel are aware that both personnel airlock doors and/or equipment hatch are open, a specific individual(s) is designated and available to close an airlock door and the equipment hatch cover following a required evacuation of the reactor building, and any obstruction(s) (e.g. cables and hoses) that could prevent closure of an airlock door and the equipment hatch cover be capable of being quickly removed.~~ LA1 Bases (3.9.3)

3.9.4
3.9.5

<CTS INSERT 59>

ANO-243

3.8.9 If any of the above specified limiting conditions for fuel loading and refueling are not met, movement of fuel into the reactor core shall cease; ~~action shall~~ be initiated to correct the conditions so that the specified limits are met, ~~and~~ no operations ~~which may increase the reactivity of the core shall be made. The provisions of Specification 3.0.3 are not applicable.~~

immediately

L5 A3 L8 A5

3.9.4 R.A. A.2
3.9.4 R.A. A.3
3.9.5 R.A. A.1
3.9.5 R.A. B.2
3.9.4 R.A. A.1
3.9.5 R.A. B.1

Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.

3.9.1 3.9.4
3.9.2 3.9.5
3.9.3 3.9.6

<LATER> (4.0)

3.8.15 Storage in the spent fuel pool shall be restricted to fuel assemblies having initial enrichment less than or equal to 4.1 w/o U-235. The provisions of Specification 3.0.3 are not applicable.

LATER

<LATER> (3.7 + 4.0)

3.8.16 Storage in Region 2 (as shown on Figure 3.8.1) of the spent fuel pool shall be further restricted by burnup and enrichment limits specified in Figure 3.8.2. In the event a checkerboard storage configuration is deemed necessary for a portion of Region 2, vacant spaces adjacent to the faces of any fuel assembly which does not meet the Region 2 burnup criteria (non-restricted) shall be physically blocked before any such fuel assembly may be placed in Region 2. This will prevent inadvertent fuel assembly insertion into two adjacent storage locations. The provisions of Specification 3.0.3 are not applicable.

LATER

<LATER> (3.7)

3.8.17 The boron concentration in the spent fuel pool shall be maintained (at all times) at greater than 1600 parts per million.

3.8.18 During the handling of irradiated fuel, the control room emergency air conditioning system and the control room emergency ventilation system shall be operable as required by Specification 3.9.

LATER

Bases

Detailed written procedures will be available for use by refueling personnel. These procedures, the above specifications, and the design of the fuel handling equipment as described in Section 9.6 of the FSAR incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety. If no change is being made in core geometry, one flux monitor is sufficient. This permits maintenance on the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition.

The requirement that at least one decay heat removal loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel at the refueling temperature (normally 140°F), and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.(1)

A2

The requirement to have two decay heat removal loops operable when there is less than 23 feet of water above the core, ensures that a single failure of the operating decay heat removal loop will not result in a complete loss of decay heat removal capability. With the reactor vessel head removed and 23 feet of water above the core, a large heat sink is available for core cooling, thus in the event of a failure of the operating decay heat removal loop, adequate time is provided to initiate emergency procedures to cool the core.

The shutdown margin indicated in Specification 3.8.4 will keep the core subcritical, even with all control rods withdrawn from the core.(2) Although the refueling boron concentration is sufficient to maintain the core $k_{eff} \leq 0.99$ if all the control rods were removed from the core, only a few control rods will be removed at any one time during fuel shuffling and

3.9.1
3.9.3
3.9.6

replacement. The keff with all rods in the core and with refueling boron concentration is approximately 0.9. Specification 3.8.5 allows the control room operator to inform the reactor building personnel of any impending unsafe condition detected from the main control board indicators during fuel movement.

The specification requiring testing reactor building purge termination is to verify that these components will function as required should a fuel handling accident occur which resulted in the release of significant fission products.

Because of physical dimensions of the fuel bridges, it is physically impossible for fuel assemblies to be within 10 feet of each other while being handled.

Per specification 3.8.6, the reactor building personnel and/or emergency airlock doors and the equipment hatch may be open during movement of irradiated fuel in the reactor building provided at least one door of each airlock and the equipment hatch are capable of being closed in the event of a fuel handling accident and the plant is in REFUELING SHUTDOWN with 23 feet of water above the fuel seated within the reactor pressure vessel. Should a fuel handling accident occur inside the reactor building, at least one of the personnel and/or emergency airlock doors and the equipment hatch will be closed following evacuation of the reactor building. For closure, the equipment hatch cover will be in place with a minimum of four bolts securing the cover to the sealing surface.

A2 + R TRM

Specification 3.8.11 is required as: 1) the safety analysis for the fuel handling accident was based on the assumption that the reactor had been shutdown for 100 hours [3]; and, 2) to assure that the maximum design heat load of the spent fuel pool cooling system will not be exceeded during a full core offload.

Specification 3.8.14 will assure that damage to fuel in the spent fuel pool will not be caused by dropping heavy objects onto the fuel. Administrative controls will prohibit the storage of fuel in locations adjoining the walls at the north and south ends of the pool, in the vicinity of cask storage area and fuel tilt pool access gates.

Specifications 3.8.15 and 3.8.16 assure fuel enrichment and fuel burnup limits assumed in the spent fuel safety analyses will not be exceeded.

Specification 3.8.17 assures the boron concentration in the spent fuel pool will remain within the limits of the spent fuel pool accident and criticality analyses.

REFERENCES

(1) FSAR, Section 9.5

(2) FSAR, Section 14.2.2.3

(3) FSAR, Section 14.2.2.3.3

Table 4.1-3

MINIMUM SAMPLING AND ANALYSIS FREQUENCY

Item	Test	Frequency	
1. Reactor Coolant Samples	a. Gamma Isotopic Analysis	a. Bi-weekly (7)	LATER
	b. Gross Activity Determination	b. 3 times/week and at least every third day (1)(6)(	
LATER (3.4B)	c. Gross Radioiodine Determination	c. Weekly (3)(6)(7)	
<LATER> (3.1)	d. Dissolved Gases	d. Weekly (7)	LATER
<LATER> (3.4A)	e. Chemistry (Cl, F, and O_2)	e. 3 times/week (8)	LATER
SR 3.9.1.1	f. Boron Concentration	f. ~~3 times/week~~ 72 hours	M12, L2 & R TRM
LATER (3.4B)	g. Radiochemical Analysis for $\bar{E}$ Determination (2) (4)	g. Monthly (7)	LATER
<LATER> (3.5) 2. Borated Water Storage Tank Water Sample	Boron Concentration	Weekly and after each makeup	LATER
3. Core Flooding Tank Sample	Boron Concentration	Monthly and afte each makeup	
<LATER> (3.7) 4. Spent Fuel Pool Water Sample	Boron Concentration	Monthly and afte each makeup (9)	LATER
5. Secondary Coolant Samples	a. Gross Radioiodine Concentration	a. Weekly (5)(7)(10)	R TRM
	b. Isotopic Radioiodine Concentration (4)	b. Monthly (7)(10)	
<LATER> (3.6) 6. Sodium Hydroxide Tank Sample	Sodium Hydroxide Concentration	Quarterly and af each makeup	LATER

<LATER> (3.4B)

Notes:

(1) A gross radioactivity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units µCi/gm. The total primary coolant activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities 15 minutes after the primary system is sampled. Whenever t gross radioactivity concentration exceeds 10% of the limit specified the Specification 3.1.4.1 or increases by 10 µCi/gm from the previous measured level, the frequency of sampling and analyzing shall be increased to a minimum of once/day until a steady activity level is established. LATER

3.9.4
3.9.5

4.27 DECAY HEAT REMOVAL

APPLICABILITY

Applies to surveillance of the decay heat removal system and to the reactor coolant loops and associated reactor coolant pumps as needed for decay heat removal. — LATER

<LATER> (3.4A)

OBJECTIVE

To assure the operability of the decay heat removal system and the reactor coolant loops as needed for decay heat removal.

SPECIFICATION

<LATER> (3.4A) 4.27.1 The required reactor coolant pumps shall be determined operable once per seven (7) days by verifying correct breaker alignments and indicated power availability. — LATER

<LATER> (5.6) 4.27.2 The required decay heat removal loop(s) shall be determined operable per Specification 4.2.2. — LATER

<LATER> (3.4A) 4.27.3 The required steam generator(s) shall be determined operable by verifying the secondary side water level to be ≥20 inches on the startup range at least once per 12 hours. — LATER

4.27.4 The required reactor coolant loop(s) shall be determined operable by verifying the required loop(s) to be in operation and circulating reactor coolant at least once per 12 hours.

3.9-07 SR 3.9.4.1 SR 3.9.5.1 + <LATER> (3.4A) 4.27.5 The required decay heat removal loop shall be determined to be in operation at least once per 12 hours. — & LATER

3.9-07 <Add SR 3.9.5.2> — M1

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

"R" - Relocation of requirements:

Relocating requirements which do not meet the Technical Specification selection criteria to documents with an established control program allows the Technical Specifications to be reserved only for those conditions or limitations upon reactor operation which are necessary to adequately limit the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety, thereby focusing the scope of Technical Specifications.

Therefore, requirements which do not meet the Technical Specification selection criteria in 10 CFR 50.36 have been relocated to other controlled license basis documents. This regulation addresses the scope and purpose of Technical Specifications. In doing so, it establishes a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in Technical Specifications. These criteria are as follows:

Criterion 1: Installed instrumentation that is used to detect and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2: A process variable that is an initial condition of a design basis accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3: A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission barrier.

Criterion 4: A structure, system or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

The application of these criteria is provided in the "Application of Selection Criteria to the ANO-1 Technical Specifications." Requirements which met the criteria have been included in the proposed improved Technical Specifications. Entergy Operations proposes to remove the requirements which do not meet the criteria from the Technical Specifications and relocate the requirements to a suitable owner controlled document. The requirements in the relocated Specifications will not be affected by this Technical Specification change. Entergy Operations will initially continue to perform the required operation and maintenance to assure that the requirements are satisfied. Relocating specific requirements for systems or variables will have no impact on the system's operability or the variable's maintenance, as applicable.

License basis document control mechanisms, such as 10 CFR 50.59, 10 CFR 50.54(a)(3), and ITS Section 5, "Administrative Controls," will be utilized for the relocated Specifications as they will be placed in other controlled license basis documents. This would allow Entergy Operations to make changes to these requirements, without NRC approval, as allowed by the applicable regulatory requirements. These controls are considered adequate for assuring structures, systems and components in the relocated Specifications are maintained operable and variables in the relocated Specifications are maintained within limits.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as indicated below:

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change relocates requirements and surveillances for structures, systems, components or variables which did not meet the criteria for inclusion in Technical Specifications as identified in the Application of Selection Criteria to the ANO-1 Technical Specifications. The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected structures, systems, components or variables will be relocated from the Technical Specifications to an appropriate administratively controlled license basis document and maintained pursuant to the applicable regulatory requirements. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or change in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the affected requirement will be relocated to an owner controlled license basis document for which future changes will be evaluated pursuant to the requirements of the applicable regulatory requirements. Therefore, this change does not involve a significant reduction in a margin of safety.

"A" - Administrative changes to requirements:

Reformatting and rewording the remaining requirements in accordance with the style of the improved Babcock & Wilcox Standard Technical Specifications in NUREG-1430 will make the Technical Specifications more readily understandable to plant operators and other users. Application of the format and style will also assure consistency is achieved between specifications. As a result, the reformatting and rewording of the Technical Specifications has been performed to make them more readily understandable by plant operators and other users. During this reformatting and rewording process, no technical changes (either actual or interpretational) to the Technical Specifications were made unless they were identified and justified.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as indicated below:

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change involves reformatting and rewording of the existing Technical Specifications. The reformatting and rewording process involves no technical changes to existing requirements. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   The proposed change will not significantly reduce the margin of safety because it has no impact on any safety analysis assumptions. This change is administrative in nature. As such, there is no technical change to the requirements and therefore, there is no significant reduction in the margin of safety.

"LA" - Less restrictive, Administrative deletion of requirements:

Portions of some Specifications provide information that is descriptive in nature regarding the equipment, system(s), actions or surveillances. This information is proposed to be deleted from the specification and relocated to other license basis documents which are under licensee control. These documents include the TS Bases, Safety Analysis Report (SAR), Technical Requirements Manual, and Programs and Manuals identified in ITS Section 5, “Administrative Controls.” The removal of descriptive information is permissible, because the documents containing the relocated information will be controlled through the applicable process provided by the regulatory requirements, e.g., 10 CFR 50.59, 10 CFR 50.54(a)(3), and ITS Section 5, “Administrative Controls.” This will not impact the actual requirements but may provide some flexibility in how the requirement is conducted. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as indicated below:

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change relocates requirements from the Technical Specifications to other license basis documents which are under licensee control. The documents containing the relocated requirements will be maintained using the provisions of applicable regulatory requirements. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be transposed from the Technical Specifications to other license basis documents, which are under licensee control, are the same as the existing Technical Specifications. The documents containing the relocated requirements will be maintained using the provisions of applicable regulatory requirements. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

"M" - More restrictive changes to requirements:

The ANO-1 Technical Specifications are proposed to be modified in some areas to impose more stringent requirements than previously identified. These more restrictive modifications are being imposed to be consistent with the improved Babcock & Wilcox Standard Technical Specifications. Such changes have been made after ensuring the previously evaluated safety analysis was not affected. Also, other more restrictive technical changes have been made to achieve consistency, correct discrepancies, and remove ambiguities from the specification.

The modification of the ANO-1 Technical Specifications and the changes made to achieve consistency within the specifications have been performed in a manner such that the most stringent requirements are imposed, except in cases which are individually evaluated.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change provides more stringent requirements for the ANO-1 Technical Specifications. These more stringent requirements are not assumed to be initiators of analyzed events and will not alter assumptions relative to mitigation of accident or transient events. The change has been confirmed to ensure no previously evaluated accident has been adversely affected. The more stringent requirements are imposed to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes do not impact the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated for ANO-1.

3. Does this change involve a significant reduction in a margin of safety?

The imposition of more stringent requirements prevents a reduction in the margin of plant safety by:

a) Increasing the analytical or safety limit,
b) Increasing the scope of the specification to include additional plant equipment,
c) Increasing the applicability of the specification,
d) Providing additional actions,
e) Decreasing restoration times,
f) Imposing new surveillances, or
g) Decreasing surveillance intervals.

The change is consistent with the safety analysis and licensing basis. Therefore, this change does not involve a reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

ANO-1 ITS SECTION 3.9: REFUELING OPERATIONS

Entergy Operations has evaluated these proposed Technical Specification changes and has determined that they involve no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92(c) as indicated below:

NSHC 3.9 L1

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The reduction in scope of the number of reactor building penetrations requiring OPERABLE automatic isolation valves does not affect the postulated initiator for any evaluated MODE 6 accident. Therefore, no significant increase in probability of any previously evaluated accident will occur. Further, the reduction in scope of the number of reactor building penetrations requiring OPERABLE automatic isolation valves will not significantly increase the consequences of an evaluated accident. This is because these penetrations are associated with closed loop systems that did not have direct access to either the reactor building atmosphere or the outside atmosphere. Without assuming a failure which resulted in a break in these systems, these penetrations were not previously a credible pathway for the release of radioactivity to the outside atmosphere should a fuel handling accident have occurred. The assumption of this additional failure resulting in the breach of these systems is not consistent with the assumptions of the analysis. Therefore, this change does not involve a significant increase in the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

An appropriate scope of applicability has been determined based on the safety analysis function that the reactor building penetrations maintain. The reduction in scope of the number of reactor building penetrations requiring OPERABLE automatic isolation valves does not result in a reduction in a margin of safety associated with any postulated MODE 6 accident. Because the leakage of radioactive materials via these penetrations was not previously credible, their exclusion from the ITS LCO requirements does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

NSHC 3.9 L2

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

A less frequent performance of a Surveillance Requirement does not result in any hardware changes. The Frequency of performance also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment does not change (and therefore any initiation scenarios are not changed) and the proposed Frequency has been determined to be adequate to demonstrate that the refueling canal boron concentration is within its limits. Further, the Frequency of performance of a surveillance does not significantly increase the consequences of an accident because a change in Frequency does not change the assumed response of the equipment in performing its specified mitigation functions from that considered with the original Frequency. Therefore, this change does not involve a significant increase in the consequences of any accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper surveillances are required for the parameters considered in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The proposed change does not impact any physical mechanism or process that would allow the refueling canal boron concentration to change in an undetected manner such that any resultant increase in core reactivity would occur. Therefore, a change in the Surveillance Frequency does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

NSHC 3.9 L3

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

A less frequent performance of a Surveillance Requirement does not result in any hardware changes. The Frequency of performance also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment does not change (and therefore any initiation scenarios are not changed) and the proposed Frequency has been determined to be adequate to demonstrate the reactor building purge isolation valves are OPERABLE. Further, the Frequency of performance of a surveillance does not significantly increase the consequences of an accident because a change in Frequency does not change the assumed response of the equipment in performing its specified mitigation functions from that considered with the original Frequency. Therefore, this change does not involve a significant increase in the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper surveillances are required for all equipment considered in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not impact the mitigatory function of the reactor building isolation valves such that any resultant increase in offsite exposure would occur. Therefore, a change in the Surveillance Frequency does not involve a significant reduction in the margin of safety.

NSHC 3.9 L4

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

While in MODES 5 or 6, the elimination of the requirement to establish reactor building integrity, when the reactor coolant system is open to the reactor building atmosphere with the required degree of subcriticality specified for a refueling shutdown not met, will not alter the assumed initiation, and hence, will not significantly increase the probability of any evaluated event. The ITS provides specific requirements for SHUTDOWN MARGIN (MODE 5) or degree of subcriticality (MODE 6). The ITS will establish Required Actions that initiate the restoration of the required SHUTDOWN MARGIN while in MODE 5. And while in MODE 6, ITS Required Actions will terminate activities that may result in the possibility of a Fuel Handling Accident which results in fission product release to the reactor building atmosphere, or otherwise affect the core reactivity condition through fuel loading errors or moderator dilution events. These ITS actions are the appropriate mitigatory actions to re-establish the initial conditions assumed in the analyses. Because these Required Actions re-establish the initial conditions assumed in the safety analyses, the consequences of a postulated event from this condition would not be significantly increased.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still result in the ITS establishing the proper control of the required SHUTDOWN MARGIN (MODE 5) or required degree of subcriticality (MODE 6) considered in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not involve a significant reduction in the margin of safety. During MODE 5, existing margins of safety would be preserved through the ITS 3.1.1, SHUTDOWN MARGIN (SDM), Required Actions. During MODE 6, three possible events could be postulated. The three are the fuel handling accident, fuel loading error and moderator dilution. Reactor building closure requirements exist, independent of the subject of this change, that would maintain the reactor building's mitigatory function as previously assumed in the Fuel Handling Accident analysis. The fuel loading error event is not expected to occur due to stringent administrative controls; and should it occur, the event is expected to manifest itself during power operations. Specific administrative controls are in place to limit the source, rate and total quantity of dilution available. Because of the administrative controls, this event would occur at a slow rate with observable indications of the abnormal condition; thus, the operator could then initiate mitigatory measures.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

<u>NSHC 3.9 L5</u>

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The elimination of the requirement to suspend the addition of non-irradiated fuel assemblies to the reactor core when a required decay heat removal (DHR) loop was inoperable will not alter the assumed initiation, and hence, will not significantly increase the probability of any evaluated event. The ITS will establish Required Actions that: 1) maintain a minimum reactor coolant boron concentration thus preserving the necessary degree of subcriticality and mixing of the reactor coolant during dilution, 2) suspend the loading of irradiated fuel assemblies in the core thus stopping an increase in the decay heat magnitude present in the core, 3) initiate action to restore the required DHR loop to operation, and 4) provide closure of all reactor building penetrations providing direct access from the reactor building atmosphere to the outside atmosphere. These actions are all consistent with the requirements of the CTS. ITS Required Actions will terminate activities that may result in increased levels of decay heat within the core, the possibility of a Fuel Handling Accident which results in fission product release to the reactor building atmosphere, or otherwise affect the core reactivity condition through a moderator dilution event. These ITS actions are the appropriate mitigatory actions to re-establish the initial conditions assumed in the analyses. Because these Required Actions re-establish the initial conditions assumed in the safety analyses, prevent the occurrence of evaluated events, and preserve the mitigatory response mechanisms should an event occur, the consequences of a postulated event from this condition would not be significantly increased.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still result in the ITS establishing the proper control of refueling activities considered in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not involve a significant reduction in the margin of safety. The allowance to continue to add non-irradiated fuel bundles to the reactor core during a period in which the required DHR loop was inoperable does not result in an increase in the decay heat magnitude of the core. Thus, any margins present during the period when the required DHR loop was inoperable, would continue to be present with the non-irradiated fuel bundles present. ITS Required Actions will terminate activities that may result in the possibility of a Fuel Handling Accident which results in fission product release to the reactor building atmosphere, or otherwise affect the core reactivity condition through fuel loading errors or moderator dilution events. These ITS Actions preserve the appropriate mitigatory actions in response to the inoperability of the required decay heat removal loop.

ANO-244

NSHC 3.9 L6

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

In MODE 6 with the water level < 23 feet above the top of the irradiated fuel seated in the reactor pressure vessel, two DHR loops are required to be operable, with one loop in operation. The proposed change will allow both DHR pumps to be de-energized for ≤ 15 minutes when switching from one train to another. The DHR pumps are not considered to be the initiator of any previously analyzed accident in the ANO-1 Safety Analysis Report. Although the proposed change will allow a 15 minute time frame with no forced circulation for cooling and mixing of boron concentration, the unit must maintain core outlet temperature > 10 degrees F below the saturation temperature, and will not be allowed to conduct any draining operation to further reduce the RCS water level or permit any operation that would cause a reduction of the RCS boron concentration. This ensures that an adequate heat sink will be available for the irradiated fuel without dependence on the DHR pumps and that mixing to distribute boron concentration changes will not be required. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration to the plant (no new or different type of equipment will be installed) or changes to parameters governing normal plant operation. The proposed change will continue to ensure that adequate capacity for heat removal is available. Therefore, this change does not result in a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

Although the proposed change will result in an allowance for the plant to be in MODE 6 with < 23 feet of water over the irradiated fuel with no forced DHR flow, this change is acceptable due to the limited time allowed, and the requirements that ensure that adequate capacity for heat removal is available and that boron concentration will not be reduced. Therefore, this change does not result in a significant reduction in a margin of safety.

ANO-245

NSHC 3.9 L7

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

In MODE 6 with the water level < 23 feet above the top of the irradiated fuel seated in the reactor pressure vessel, two DHR loops are required to be operable, with one loop in operation. The proposed change will allow one required DHR pump to be inoperable for up to 2 hours for surveillance testing. The DHR pumps are not considered to be the initiator of any previously analyzed accident in the ANO-1 Safety Analysis Report. Although the proposed change will allow one DHR pump to be inoperable for up to 2 hours, the remaining DHR pump must be operable and in operation. In addition, since the time to boil may be short, no draining operation to further reduce the RCS level is allowed, and the plant must be capable of injecting borated water into the reactor vessel. This provides assurance that an adequate heat sink remains available in the event the operating pump becomes inoperable. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration to the plant (no new or different type of equipment will be installed) or changes to parameters governing normal plant operation. The proposed change will continue to ensure that adequate capacity for heat removal is available. Therefore, this change does not result in a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

Although the proposed change will result in an allowance for the plant to be in MODE 6 with < 23 feet of water over the irradiated fuel with one required DHR pump inoperable, this change is acceptable due to the limited time allowed, and the requirements that ensure that adequate capacity for heat removal is available. Therefore, this change does not result in a significant reduction in a margin of safety.

ANO-243

NSHC 3.9 L8

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Allowing positive reactivity additions, in MODE 6 with no DHR flow available, that will not reduce RCS boron concentration below a minimum concentration specified in the Core Operating Limits Report, and thus maintain the minimum required SDM, will not alter the assumed initiation, and hence, will not significantly increase the probability of any evaluated event. The ITS contains actions that maintain the initial conditions assumed in the analyses. Because these Required Actions maintain the initial conditions assumed in the safety analyses, prevent the occurrence of evaluated events, and preserve the mitigatory response mechanisms should an event occur, the consequences of a postulated event from this condition would not be significantly increased.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration to the plant (no new or different type of equipment will be installed) or changes to parameters governing normal plant operation. The proposed change will continue to ensure that adequate boron concentration is maintained. Therefore, this change does not result in a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will allow positive reactivity changes in MODE 6 with no DHR flow. However, the ITS Required Actions limit such positive reactivity additions to provide assurance that the minimum boron concentration specified in the Core Operating Limits Report, and thus the minimum required SDM, are maintained. Therefore, this change does not result in a significant reduction in a margin of safety.

1. NUREG 3.9.3, 3.9.4, and 3.9.5 - At numerous locations, the ITS and ITS Bases have been marked to reflect the ANO-1 unit specific terminology for its "reactor building;" rather than the NUREG-1430 term "containment." This change has been annotated at each occurrence in the ITS. This editorial change is made to retain conformity to the current license basis documents.

2. NUREG 3.9.3 Bases - An insert to the Bases for Specification 3.9.3 clarifies that a temporary equipment hatch that is securely held in place may satisfy the requirement that the equipment hatch be closed and held in place by four bolts. ANO-1 has a steel temporary equipment hatch for the purpose of providing a secure reactor building closure. This insertion clarifies that it is acceptable for ANO-1 to continue to use the temporary equipment hatch structure as provided in Unit 1 SAR, Section 5.2.2.1.3. This change is consistent with current license basis.

3. 3.9-05 The ANO-1 fuel handling accident analysis credits no mitagatory actions with respect to reactor building closure, and therefore does not credit automatic closure of the reactor building purge valves. This was discussed in the ANO submittals related to Amendments 184 and 195, dated September 20, 1996 and April 16, 1999, respectively.

   NUREG 3.9.3 - LCO 3.9.3.c.2 makes reference to "an OPERABLE Containment Purge and Exhaust Isolation System." Several aspects of this LCO require modification in order to reflect the ANO-1 purge system configuration and operational capability. CTS 3.8.7 allows the isolation valves for reactor building penetrations to be open provided an isolation valve associated with those penetrations is OPERABLE. Further, the accident analyses do not credit reactor building purge isolation automatic isolation on high radiation levels. Valve closure is manually initiated by the operator. Lastly, CTS 3.8.10 requires that the reactor building purge isolation valves, and the associated purge exhaust radiation monitor be OPERABLE. Therefore, ITS 3.9.3 was modified to reflect the current license requirements and system configuration. ITS 3.9.3.c.2 was modified to reflect that the penetration must be "capable of being closed by an OPERABLE reactor building isolation valve, except for reactor building purge isolation valves." This preserves the CTS 3.8.7 requirements. This change is consistent with current license basis.

   ITS 3.9.3.c.3 was inserted to require the reactor building purge isolation valves be capable of being closed and the associated purge exhaust radiation monitoring channel be OPERABLE. This LCO requirement preserves the CTS 3.8.10 requirements. This separate LCO requirement was provided to specifically differentiate the requirement for an OPERABLE purge exhaust radiation monitor for this penetration flowpath from the requirements for the ITS 3.9.3.c.2 penetration flowpaths. This change is consistent with current license basis.

   Because of the ANO-1 reactor building purge system configuration and the CTS 3.8.7 allowance incorporated into ITS 3.9.3.c.2, SR 3.9.3.2 was modified to remove reference to closure initiation "on an actual or simulated actuation signal." Valve closure would be as a result of operator initiated action. Automatic closure of these valves on high radioactivity levels is not credited in the accident analyses. In addition, no requirement for an OPERABLE Engineered Safeguards (ES) actuation capability exists in MODE 6 as stated in the Bases. SR 3.9.3.2 and its Note (Ref. DOD 4) were further modified to specifically include the ITS 3.9.3.c.3 reactor building isolation valves. This change does not alter the intent of the SR, which is to verify that the isolation valves will close when required to do so. This change is consistent with current license basis.

ITS SR 3.9.3.3 was added to require a CHANNEL CALIBRATION of the reactor building purge exhaust radiation monitor with an 18 month Frequency. This SR presents the equivalent requirements of CTS 3.8.10. This SR ensures the OPERABILITY of the radiation monitoring instrumentation used to alert the operator of the need to isolate the reactor building purge release path. This change is consistent with current license basis.

The Bases description for ITS 3.9.3 required several modifications to reflect the LCO and SR changes. First, all reference to a "mini-purge" system was eliminated. ANO-1 has no such system. Second, reference to automatic isolation capability for the reactor building purge system penetrations was removed. These valves may be closed by an operator from the control room following receipt of indication that a high radiation level exists in the reactor building purge exhaust stream. No automatic closure interlock based on high radioactivity levels exists for these isolation valves. Third, all reference to ESAS functional capability was removed from the Bases supporting the OPERABILITY requirement while in MODE 6 (during Refueling) because these requirements are not pertinent. Fourth, the text was revised to reflect current license requirements that allow other reactor building penetrations to be open provided they are capable of being closed by an OPERABLE isolation valve. Lastly, the new LCO and SR requirements were incorporated into the Bases. These changes are consistent with current license basis.

3.9-05 The Bases description for SR 3.9.3.1 has been revised to delete requirements not specifically contained in SR 3.9.3.1, such as a demonstration that the valves are not blocked from closing, and demonstration that each valve operator has motive power. These changes are consistent with the current license basis, and are consistent with the assumptions in the fuel handling accident analysis, as discussed above.

ANO-241 4. NUREG 3.9.3 - Incorporates TSTF-284, Rev. 3.

5. NUREG 3.9.2 - Incorporates TSTF-096, Rev. 1.

6. NUREG 3.9.2 Bases - The Background section was modified to accurately describe the ANO-1 source range monitors. Gamma-Metrics fission chambers were installed as a post-TMI commitment that required environmentally qualified nuclear instrumentation. These instruments have superseded the original Bailey Model 880, BF_3 source range

instrumentation. The Bases were modified to reflect that the fission chamber units are the principle nuclear instruments and that the original BF_3 instruments are not used for satisfying source range nuclear instrumentation monitoring requirements during shutdown conditions. This change is consistent with current license basis.

As discussed in the first paragraph of the Background Section, portable source range instruments may be used to satisfy the LCO requirements. For clarification, the word portable was replaced with the word temporary. Although no change in intent exists with this change, it does eliminate the connotation that the substitute instrument would have to posses some degree of mobility. Further, temporary better describes the type of instrument that may be used in this context. To further clarify the usage of temporary source range monitors, an additional paragraph was inserted into the LCO that establishes the requirement that the temporary instrument be functionally equivalent to the installed instrumentation.

7. NUREG 3.9.4 - The Applicability was changed to cover operation in MODE 6 with the water level ≥ 23 ft above the top of the irradiated fuel seated in the reactor pressure vessel. This change preserves the Applicability implied by CTS 3.8.3.a. Implied because CTS 3.8.3.b addresses the decay heat removal requirements when the water level is less than 23 feet above the irradiated fuel seated in the reactor vessel. And, CTS 3.8.3.b is premised on already having a requirement for one DHR loop being OPERABLE and in operation. This Applicability preserves the large inventory requirement that is capable of providing decay heat removal for an extended period of time. This change is consistent with current license basis. 3.9-07

8. NUREG 3.9.5 - The Applicability for LCO 3.9.5 was changed to cover operation in MODE 6 with the water level less than 23 feet above the top of the irradiated fuel assemblies seated in the reactor pressure vessel. This change in Applicability replicates that established in CTS 3.8.3.b. Associated with the change in LCO 3.9.5 Applicability, Required Action A.2 was modified to provide consistent Actions for exiting the Applicability as one of the options available to the operator. This change is consistent with current license basis.

   The Bases were modified as necessary to reflect these changes.

9. NUREG 3.9.5 Bases - The Bases discussion for LCO 3.9.5 was modified by an inserted sentence that clarifies that the DHR loops may be considered OPERABLE when aligned to the Borated Water Storage Tank (BWST). This provision is necessary to support filling of the refueling canal or the performance of required testing of the DHR trains. Further, this clarification is necessary because of the explicit discussion in the LCO Bases of what constitutes a DHR flow path. This change to the Bases acknowledges these special operational conditions. This change is consistent with current license basis.

10. NUREG 3.9.3 - ITS 3.9.3.a and ITS 3.9.3.b were modified to reflect the CTS 3.8.6 requirements regarding the personnel and emergency air locks. The CTS requires that one door in each air lock be capable of being closed while moving irradiated fuel within the reactor building. The CTS also requires that the equipment hatch be capable of being closed while moving irradiated fuel within the reactor building. Associated with this requirement are administrative controls that ensure that personnel are capable of closing the airlock door and equipment hatch cover at the appropriate time. These administrative controls are discussed in the ITS 3.9.3 Bases. This change reflects current license basis.

11. NUREG 3.9.4 - SR 3.9.4.1 was modified to remove reference to a minimum decay heat removal system volumetric flowrate. The ANO-1 CTS does not establish a minimum flow requirement. The actual minimum flow rate is administratively controlled in operating procedures. Operation of the system is sufficient to ensure adequate mixing of the coolant to prevent boron stratification. Adequate heat removal is a function of a number of system parameters in addition to a minimum volumetric flowrate. As such, the operator has direct indication of the adequacy of the decay heat removal system in removing decay heat and adjustments would be made based on the trended indications. Although not done for this reason, this change establishes consistency between this SR and numerous ITS Section 3.4 SRs requiring verification of DHR operation. ANO-1 continues to employ administrative and procedural controls to ensure adequate DHR flow, which have been acceptable for operation under CTS. This change is consistent with current license basis.

    The SR 3.9.4.1 Bases were modified as necessary to reflect this change.

3.9-07 The SR 3.9.5.1 Bases were also revised to incorporate this change. SR 3.9.5.1 provides a requirement to verify one DHR loop is in operation. The NUREG Bases would require a flow determination as was discussed above. As stated above, ANO-1 continues to employ administrative and procedural controls to ensure adequate DHR flow, which have been acceptable for operation under CTS. This change is consistent with current license basis.

3.9-07 12. Not used.

13. NUREG 3.9.1 – Incorporates TSTF-214.

14. NUREG 3.9.2 - CTS 3.8.2 established the requirements for (source range) neutron flux monitoring in MODE 6. This Specification required one OPERABLE monitor when "core geometry is not being changed," and two OPERABLE monitors "whenever core geometry is being changed." NUREG-1430 has been modified to reflect these CTS requirements. ITS 3.9.2.a requires one source range neutron flux monitor be OPERABLE during the LCO Applicability (MODE 6). ITS 3.9.2.b requires one additional source range neutron flux monitor be OPERABLE during CORE ALTERATIONS. This change is consistent with current license basis.

Condition A was modified to establish a Condition that was entered when one of the required source neutron flux monitors was inoperable "during CORE ALTERATIONS." NUREG-1430 Required Actions A.1 and A.2 replicate CTS 3.8.9 requirements for this entry Condition. With this change, the Condition A entry condition matches the Applicability of the ITS 3.9.2.b requirements and provides the appropriate Required Actions for this Condition.

Condition B was modified to establish a Condition that is entered when there are no OPERABLE source range neutron flux monitors. Required Action B.1, in addition to Required Actions A.1 and A.2 if during CORE ALTERATIONS, replicates the required CTS 3.8.9 requirements for this condition.

These changes maintain the requirements of the CTS while providing adequate monitoring capability of changes in the core's neutron flux. When core reactivity conditions are stable, i.e., no CORE ALTERATIONS are in progress, one neutron source range monitor is adequate. During the conditions when the core's reactivity condition is subject to change, i.e., during CORE ALTERATIONS, two monitors are required to provide independent and redundant monitoring capability of the reactivity changes in the core. This change is consistent with current license basis.

15. NUREG 3.9.3 - CTS 3.8.6 established the Applicability for reactor building closure penetrations as "during the handling of irradiated fuel in the reactor building." This CTS requirement will be retained as the Applicability for ITS 3.9.3. The NUREG Applicability of "during CORE ALTERATIONS" will not be adopted in the ITS. Retention of the CTS Applicability results in NUREG 3.9.3 Required Action A.1 not being adopted because it presents requirements that are inconsistent with the LCO Applicability. These changes maintain the current license requirements.

16. Not used.

17. NUREG 3.9.5 - SR 3.9.5.2 was revised to clarify that the surveillance is applicable to each required pump regardless of its operating status since both pumps may be operating. The Bases are also revised to indicate that if a pump is verified to be in operation, this is also sufficient to verify the correct breaker alignment and indicated power availability. This change is consistent with changes made to NUREG LCO 3.4.5, LCO 3.4.6, LCO 3.4.7, and LCO 3.4.8 - NUREG SR 3.4.5.2, SR 3.4.6.2, SR 3.4.7.3 and SR 3.4.8.2.

    The Bases are also revised to reflect this change.

18. Not used

19. NUREG 3.9.2 Bases - The Applicability discussion for ITS 3.9.2 covered MODES 2 through 6. An editorial change added a paragraph describing the lack of applicability in

MODE 1. This editorial change preserves the unit specific configuration and functional capabilities and was made only for completeness. This change is consistent with current license basis.

20. Bases ITS 3.9.3 - Additional guidance on what constitutes a "direct access" path from the reactor building to the outside atmosphere was provided. This is intended to assist the operator in determining the scope of the LCO and assist in determining the acceptability of temporary closures. This avoids the need for future interpretation of what constitutes "direct access." This change preserves the interpretations allowed under the current license basis.

21. NUREG 3.9.6 - Incorporates TSTF-020.

22. NUREG 3.9.6 - The LCO was revised to reflect the CTS 3.8.6 requirement that the refueling canal level be maintained greater than or equal to 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel. This change preserves the initial conditions of the ANO-1 Fuel Handling Accident in the reactor building. This change is consistent with current license basis as recently approved in Amendment 184.

    NUREG 3.9.6 Bases - The Applicable Safety Analyses discussion has been revised to describe the initial assumptions of the ANO-1 Fuel Handling Accident in the reactor building. This change maintains consistency with the ANO-1 license basis.

    CTS 3.8.6 defined the Applicability for the refueling canal water level requirements as "during the handling of irradiated fuel in the reactor building." This Applicability is preserved in ITS 3.9.6. The NUREG-1430 requirements of during CORE ALTERATIONS, except during latching and unlatching or CONTROL ROD drive shafts, is not adopted. The assumed initiator of the Fuel Handling Accident is the accidental drop of an irradiated fuel assembly with its subsequent fall to a horizontal position. Protective requirements for this assumed initiation condition are preserved by limiting the Applicability to "during movement of irradiated fuel assemblies ...." This change is consistent with current license basis.

    NUREG 3.9.6 Required Action A.1 is not adopted because it established Required Actions contrary to the ITS 3.9.6 Applicability. NUREG Required Action A.2 (ITS 3.9.6 RA A.1) that requires the suspension of the movement of irradiated fuel assemblies is sufficient to prevent the occurrence of a Fuel Handling Accident should the refueling canal water level drop below 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel. This change is consistent with current license basis.

    NUREG SR 3.9.6.1 was modified to reflect the LCO required level of greater than or equal to 23 feet above the top of irradiated fuel assemblies seated with the reactor pressure vessel. This change is consistent with current license basis.

23. NUREG Bases - ANO-1 was designed and licensed to the AEC's General Design Criteria (GDC) which was published in the Federal Register on July 11, 1967 [32FR10213]. Appendix A to 10 CFR 50 effective in 1971 [36FR3256] and subsequently amended, is somewhat different from the proposed 1967 criteria. SAR Section 1.4 includes an evaluation of ANO with respect to the 1967 criteria. The NUREG statement concerning the GDC criteria is modified in the ITS to reference the current licensing basis description in the SAR.

24 NUREG Bases - The Criterion statement at the conclusion of the Applicable Safety Analysis section was modified at each occurrence to refer to 10 CFR 50.36 instead of the NRC Policy Statement. This is an editorial change associated with the implementation of the 10 CFR 50.36 rule changes after NUREG-1430, Revision 1 was issued.

The 10 CFR 50.36 Criterion satisfied by the ITS LCOs was modified to preserve consistency with the ANO-1 license basis. The NUREG Criterion specified were modified to be consistent with the analysis assumptions regarding equipment availability and operating condition (i.e., MODE).

ANO-246 The Criterion statements for the NUREG 3.9.4 and 3.9.5 Bases were revised to incorporate TSTF-367, except for the reference to 10 CFR 50.36, as discussed above.

3.9-07 25 Not used.

26 NUREG 3.9.1 Bases - The NUREG states that the refueling boron concentration is intended to ensure an overall core reactivity of $K_{eff} \leq 0.95$ during fuel handling with control rods and fuel assemblies assumed to be in the most adverse condition. These statements have been modified to reflect the current ANO-1 license basis. The ANO-1 analysis assumptions for the boron dilution accident are based on an initial K_{eff} of ≤ 0.99. The Bases for CTS 3.8.4 states: "The shutdown margin indicated in Specification 3.8.4 will keep the core subcritical, even with all control rods withdrawn from the core. Although the refueling boron concentration is sufficient to maintain the core $K_{eff} \leq 0.99$ if all control rods were removed from the core, only a few control rods will be removed at any one time during fuel shuffling and replacement. The K_{eff} with all rods in the core and with refueling boron concentration is approximately 0.9." Therefore, the required overall core reactivity has been changed from $K_{eff} \leq 0.95$ to $K_{eff} \leq 0.99$ for consistency with the current license basis.

ANO-251 27. Incorporates TSTF-272, Rev 1, except as noted in DOD-28.

3.9-01 28. NUREG LCO 3.9.1 and Bases - The NUREG requires that the boron concentration of the refueling cavity be maintained within the limit specified in the Core Operating Limits Report. However, ANO does not use this terminology. A search of the Safety Analysis Report has shown no instance of "refueling cavity." Therefore, this term has been deleted in the ITS as a plant specific difference.

ANO-244 29. Incorporates TSTF-349, Rev 1. Editorial changes were made to allow the incorporation of TSTF-361, Rev 2.

ANO-245 30. Incorporates TSTF-361, Rev 2. Editorial changes were made to allow the incorporation of TSTF-349, Rev 1.

ANO-243 31. Incorporates TSTF-286, Rev 2. Editorial changes have been incorporated to improve the readability of the 3.9.4 LCO Bases.

3.9-07 32. NUREG 3.9.5 Bases discussion for Required Action B.2 contains a discussion of two methods of alternate decay heat removal. This information has been deleted in the ITS. The Required Action discussion includes a reference to alternate decay heat removal methods as specified in the unit's Abnormal and Emergency Operating Procedures. This provides a ready reference to the methods to be used. Retaining two examples in the Bases could result in confusion since it does not present a complete list. In this condition, the Operator would be using his Abnormal and Emergency Operating Procedures, as well as the Technical Specifications, therefore, this information does not provide guidance that would already be in use.

CTS

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

3.9-01

LCO 3.9.1 ~~a.~~ Boron concentrations of the Reactor Coolant System, and the refueling canal ~~, and the refueling cavity~~ shall be maintained within the limit specified in the COLR. edit 3.8.4 28

~~b. Boron concentration shall not be reduced unless reactor coolant is circulating.~~ 13

28 3.8.4

AND-251 3.9-01

APPLICABILITY: MODE 6.

NOTE
Only applicable to the refueling canal ~~and refueling cavity~~ when connected to the RCS. 27

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME	
A. Boron concentration not within limit.	A.1	Suspend CORE ALTERATIONS.	Immediately	3.8.9
	AND			
	A.2	Suspend positive reactivity additions.	Immediately	3.8.9
	AND			
	A.3	Initiate action to restore boron concentration to within limit.	Immediately	3.8.9

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.9.1.1	Verify boron concentration is within the limit specified in the COLR.	72 hours	Table 4.1-3 Item 1.f.

CTS

3.9 REFUELING OPERATIONS

3.9.2 Nuclear Instrumentation

LCO 3.9.2 a. ~~Two~~ One source range neutron flux ~~monitors~~ monitor shall be OPERABLE~~.~~, and (14) 3.8.2
b. One additional source range neutron flux monitor shall be OPERABLE during CORE ALTERATIONS. 3.8.2

APPLICABILITY: MODE 6. 3.8.2

ANO-243

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One ~~(required)~~ source range neutron flux monitor inoperable ~~.~~ during CORE ALTERATIONS	A.1 Suspend CORE ALTERATIONS.	Immediately	3.8.9 (14)
	AND		
	A.2 ~~Suspend positive reactivity additions.~~ Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately	(31) 3.8.9
B. ~~Two (required)~~ No OPERABLE source range neutron flux ~~monitors inoperable.~~ monitor	B.1 Initiate action to restore one source range neutron flux monitor to OPERABLE status.	Immediately	(14) 3.8.9
	AND		
	B.2 Perform SR 3.9.1.1.	~~4 hours~~ ~~AND~~ Once per 12 hours ~~thereafter~~	N/A (5)

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.9.2.1	Perform CHANNEL CHECK.	12 hours	N/A
SR 3.9.2.2	------------------NOTE------------------ Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION.	~~[~~18~~]~~ months	N/A N/A

CTS

3.9 REFUELING OPERATIONS (1)

3.9.3 ~~Containment~~ Reactor Building Penetrations

LCO 3.9.3 The ~~containment~~ reactor building penetrations shall be in the following status:

a. The equipment hatch is capable of being closed ~~and held in place by four bolts~~; 3.8.6 (10)

b. One door in each air lock closed; and 3.8.6

c. Each penetration providing direct access from the ~~containment~~ reactor building atmosphere to the outside atmosphere either: (1)

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or 3.8.7

2. capable of being closed by an OPERABLE ~~Containment Purge and Exhaust~~ reactor building Isolation ~~System~~ valve, except reactor building purge isolation valves, or (2) 3.8.7 (3) 3.8.10

<INSERT 3.9-4A> —>> (15)

APPLICABILITY: ~~During CORE ALTERATIONS,~~ (15) 3.8.6
During movement of irradiated fuel assemblies within ~~containment~~ the reactor building. (1)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more ~~containment~~ reactor building penetrations not in required status.	~~A.1 Suspend CORE ALTERATIONS.~~	~~Immediately~~ (15) (1)
	~~AND~~	
	~~A.2~~ A.1 Suspend movement of irradiated fuel assemblies within ~~containment~~ the reactor building.	Immediately 3.8.9 (1)

<INSERT 3.9-4A>

3. capable of being closed by an OPERABLE reactor building purge isolation valve with the purge exhaust radiation monitoring channel OPERABLE.

(1)

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	CTS
SR 3.9.3.1	Verify each required reactor building ~~containment~~ penetration is in the required status.	7 days	N/A
SR 3.9.3.2	Verify each required reactor building ~~containment purge and exhaust~~ isolation valve and each reactor building purge isolation valve actuates to the isolation position~~, on an actual or simulated actuation signal~~.	~~[~~18~~]~~ months	N/A (3)
	------- NOTE ------- Not required to be met for ~~containment purge and exhaust valves~~ reactor building isolation valves and reactor building purge isolation valves in penetrations closed to comply with LCO c.1.		(4) (3)
SR 3.9.3.3	Perform CHANNEL CALIBRATION of reactor building purge exhaust radiation monitor.	18 months	3.8.10 (3)

ANO-241

INSERT

3.9-07

CTS

3.9 REFUELING OPERATIONS

3.9-07

3.9.4 Decay Heat Removal (DHR) and Coolant Circulation—High Water Level

LCO 3.9.4 One DHR loop shall be OPERABLE and in operation. 3.8.3.a

ANO-243

---NOTE---
The required DHR loop may be removed from operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause ~~reduction of~~ introduction into the Reactor Coolant System, coolant with ~~boron concentration~~ boron concentration less than that required to meet the minimum required boron concentration of LCO 3.9.1. 3.8.3.a Note * (31)

3.9-07

APPLICABILITY: MODE 6 with the water level ≥ 23 ft above the top of ~~reactor vessel flange~~ the irradiated fuel seated in the reactor pressure vessel. N/A (7)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME	
A. DHR loop requirements not met.	A.1	~~Suspend operations involving a reduction in reactor coolant boron concentration.~~ Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1. (31)	Immediately	3.8.3.a 3.1.1.1.B 3.8.9
	AND			
	A.2	Suspend loading irradiated fuel assemblies in the core.	Immediately	3.8.3.a 3.8.9
	AND			
	A.3	Initiate action to satisfy DHR loop requirements.	Immediately	3.8.9 3.8.3.b (2ND ¶)
	AND			

(continued)

ANO-243

3.9-07

CTS

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME	CTS
A. (continued)	A.4	Close all reactor building ~~containment~~ penetrations providing direct access from ~~containment~~ the reactor building atmosphere to ~~outside~~ atmosphere.	4 hours	3.8.3.a 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	CTS
SR 3.9.4.1	Verify one DHR loop is in operation. ~~and circulating reactor coolant at ≥ flow rate of ≥ [2800] gpm.~~	12 hours	4.27.5 11

CTS

3.9 REFUELING OPERATIONS

3.9.5 Decay Heat Removal (DHR) and Coolant Circulation—Low Water Level

ANO-245
ANO-244 3.9-07

LCO 3.9.5 Two DHR loops shall be OPERABLE, and one DHR loop shall be in operation. 3.8.3.a 3.8.3.b 29 30 edit

<INSERT 3.9-8A>

APPLICABILITY: MODE 6 with the water level < 23 ~~ft~~ feet above the top of ~~reactor vessel flange~~ the irradiated fuel seated in the reactor pressure vessel. 3.8.3.b 8

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. Less than required number of DHR loops OPERABLE.	A.1 Initiate action to restore DHR loop to OPERABLE status.	Immediately	3.8.9 3.8.3.b (2ND ¶)
	OR		
	A.2 Initiate action to establish ≥ 23 ~~ft~~ feet of water above the top of ~~reactor vessel flange~~ the irradiated fuel seated in the reactor pressure vessel.	Immediately	N/A edit 8
B. No DHR loop OPERABLE or in operation.	B.1 ~~Suspend operations involving a reduction in reactor coolant boron concentration.~~ Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately	31 3.8.3.a
	AND		
	B.2 Initiate action to restore one DHR loop to OPERABLE status and to operation.	Immediately	3.8.3.b (2ND ¶)
	AND		
		(continued)	

ANO-243 3.9-07

<INSERT 3.9-8A>

ANO-244

------------------------------------NOTES------------------------------------

1. All DHR pumps may be de-energized for ≤ 15 minutes when switching from one train to another provided:

   a. The core outlet temperature is maintained > 10 degrees F below saturation temperature;

   b. No operations are permitted that would cause a reduction of the Reactor Coolant System boron concentration; and

   c. No draining operations to further reduce RCS water volume are permitted.

ANO-245

2. One required DHR loop may be inoperable for up to 2 hours for surveillance testing, provided that the other DHR loop is OPERABLE and in operation.

3.9-07

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.3 Close all ~~containment~~ reactor building penetrations providing direct access from ~~containment~~ the reactor building atmosphere to outside atmosphere.	4 hours

3.8.3.a

1

3.9-07

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.5.1 Verify one DHR loop is in operation.	12 hours
SR 3.9.5.2 Verify correct breaker alignment and indicated power available to ~~the~~ each required DHR pumps ~~that is not in operation~~.	7 days

4.27.5

N/A

17

CTS

3.9 REFUELING OPERATIONS

3.9.6 Refueling Canal Water Level

feet

3.8.6

LCO 3.9.6 Refueling canal water level shall be maintained ≥ 23 ~~ft~~ above the top of ~~the reactor vessel flange.~~ irradiated fuel assemblies seated within the reactor pressure vessel. 22

APPLICABILITY: ~~During CORE ALTERATIONS, except during latching and unlatching of CONTROL ROD drive shafts,~~ 22
During movement of irradiated fuel assemblies within ~~containment.~~ the reactor building. 3.8.6 1

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME	
A. Refueling cavity water level not within limit.	~~A.1~~	~~Suspend CORE ALTERATIONS.~~	~~Immediately~~	22
	~~AND~~			
	A.~~2~~ 1	Suspend movement of irradiated fuel assemblies within ~~containment.~~ the reactor building.	Immediately	edit 3.8.9 1
	~~AND~~			
	~~A.3~~	~~Initiate action to restore refueling cavity water level to within limit.~~	~~Immediately~~	21

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.9.6.1	Verify refueling canal water level is ≥ 23 ft feet above the top of ~~reactor vessel flange~~ irradiated fuel assemblies seated within the reactor pressure vessel.	24 hours	N/A

22

3.9-01

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

28 edit

The limit on the boron concentrations of the Reactor Coolant System (RCS), and the refueling canal, ~~and the refueling cavity~~ during refueling ensures that the reactor remains subcritical during MODE 6. The Refueling boron concentration is ~~the soluble boron concentration in~~ Specified for the coolant in each of these volumes ~~having~~ since each volume has direct access to the reactor core during refueling.

edit edit edit

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit ~~is~~ specified in the COLR~~. Unit procedures~~ ensures ~~the specified boron concentration in order to maintain~~ an overall core reactivity of $k_{eff} \leq 0.9$~~5~~ ~~during fuel handling~~, with all control rods OUT ~~and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by unit procedures~~.

9 26

SAR, Section 1.4, GDC 26 ~~of 10 CFR 50, Appendix A~~, requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The ~~Chemical Addition~~ Makeup and Purification System ~~serves as the system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration~~ has the ability to initiate and maintain a cold shutdown condition in the reactor.

23

INSERT From Applicable Safety Analyses (B 3.9-2)

~~The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling cavity. The refueling canal and the refueling cavity are then flooded with borated water from the borated water storage tank into the open reactor vessel by gravity feeding or by the use of the Decay Heat Removal (DHR) System pumps.~~

edit

~~The pumping action~~ Operation of the ~~DHR~~ Decay Heat Removal (DHR) System in the RCS~~, and the natural circulation due to thermal driving heads in the reactor vessel and the refueling cavity~~ mixes the added concentrated boric acid with the water in the refueling canal. The DHR System is in operation during refueling (see

edit edit edit

(continued)

Decay Heat Removal (DHR)

BASES

3.9-01

BACKGROUND (continued)

LCO 3.9.4, "~~DHR~~ and Coolant Circulation ~~High Water Level~~," 25 edit
and LCO 3.9.5, "~~DHR~~ and Coolant Circulation – Low Water Level") to provide forced circulation in the RCS and assist edit
in maintaining the boron concentrations in the RCS, the and
refueling canal ~~and the refueling cavity~~ above the COLR
limit. 28

APPLICABLE SAFETY ANALYSES

During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident in the accident analysis ~~and is conservative for MODE 6~~. The boron concentration limit specified in the COLR is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance. edit

9

The required boron concentration and the unit refueling procedures ~~that demonstrate the correct fuel loading plan (including full core mapping)~~ ensure the k_{eff} of the core ~~will remain ≤ 0.95~~ during the refueling operation. ~~Hence, at least a 5% Dk/k margin of safety is established during refueling.~~ edit 26 edit

MOVE to BACKGROUND (B 3.9-1)

3.9-01

tube

During refueling, ~~the water volume in~~ the spent fuel pool, the transfer ~~canal~~, the refueling canal ~~the refueling cavity~~, and the reactor vessel ~~form a single mass~~ are connected. As a result, the soluble boron concentration is relatively the same in each of these volumes. edit 28 edit

The RCS boron concentration satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10CFR 50.36 (Ref. 2). 24

3.9-01

LCO

The LCO requires that a minimum boron concentration be maintained ~~in the RCS~~ and the refueling canal ~~and the refueling cavity~~ while in MODE 6. The boron concentration limit specified in the COLR ensures a core k_{eff} of ≤ ~~0.95~~ 0.99 is maintained during fuel handling operations with CONTROL RODS and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity allowed by unit procedures.) 28 26 edit

~~This LCO also requires that coolant be circulated during any boron dilution. Providing forced coolant circulation during changes in boron concentration ensures mixing of the coolant, eliminating the potential for pockets of diluted, unmixed coolant, which may cause loss of required SDM.~~ 13

(continued)

BASES

LCO (continued)

~~Adequate mixing prevents stratification to ensure that dilution induced reactivity changes are gradual, as well as recognizable and controllable by the operator. Forced circulation will also ensure that the boron concentration determined by chemical analysis is representative of the entire coolant volume.~~ 13

Violation of the LCO ~~could lead to~~ provides a potential for an inadvertent criticality during MODE 6. edit

---

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. ~~The required boron concentration ensures a $k_{eff} \leq 0.95$.~~ edit

(New paragraph) 9

Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," LCO 3.1.5, "Safety Rod Insertion Limits," and LCO 3.2.1, "Regulating Rod Insertion Limits," ~~and LCO 3.1.2, "Reactivity Balance"~~ ensure that an adequate amount of negative reactivity is available to shut down the reactor and to maintain it subcritical. edit

27 28

ACTIONS

ANO-251 3.9-01 <INSERT B 3.9-3A>

A.1 and A.2

Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO. If the boron concentration of ~~any coolant volume in~~ the RCS, or the refueling canal ~~, or the refueling cavity~~ is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately. edit 28

3.9-01

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position. 31

ANO-243 <INSERT B 3.9-3A>

A.3

In addition to immediately suspending CORE ALTERATIONS ~~or~~ and positive reactivity additions, action to restore the concentration must be initiated immediately. edit

~~In determining the required combination of boration flow rate and concentration,~~ there is no unique Design Basis edit edit

(continued)

<INSERT B3.9-3A>

ANO-251

The Applicability is modified by a Note. The Note states that the limits on boron concentration are only applicable to the refueling canal when that volume is connected to the Reactor Coolant System. When the refueling canal is isolated from the RCS, no potential path for boron dilution exists.

<INSERT B3.9-3B>

ANO-243

Operations that add limited positive reactivity (e.g., temperature fluctuations from inventory addition or temperature control fluctuations), but when combined with all other operations affecting core reactivity (e.g., intentional boration) result in overall net negative reactivity addition, are not precluded by this action.

BASES

ACTIONS

A.3 (continued)

analysis that requires a specific rate of boration

event ~~that must be satisfied~~. The only requirement is to restore the boron concentration to its required value as soon as possible. ~~In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.~~ edit edit

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

ANO-251 3.9-01 3.9-01

SR 3.9.1.1 27

This SR ensures the coolant boron concentration in the RCS, 28 the refueling canal, and connected portions of ~~and the refueling cavity~~ is within the COLR limits. The boron concentration of the coolant in each volume is determined ~~periodically~~ every 72 hours by chemical analysis. 27

<INSERT B3.9-4A> required

~~A minimum Frequency of once every 72 hours is therefore a reasonable amount of time to verify the boron concentration of representative samples.~~ The Frequency is based on ~~operating~~ industry experience, which has shown 72 hours to be adequate. edit edit edit

REFERENCES

1. ~~10 CFR 50, Appendix A~~ SAR, Section 1.4, GDC 26. 23
2. 10 CFR 50.36. 24

<INSERT B3.9-4A>

ANO-251

Prior to re-connecting portions of the refueling canal to the RCS, this SR must be met per SR 3.0.4. If any dilution activity has occurred while the cavity was disconnected from the RCS, this SR ensures the correct boron concentration prior to communication with the RCS.

B 3.9 REFUELING OPERATIONS

B 3.9.2 Nuclear Instrumentation

BASES

BACKGROUND

The source range neutron flux monitors are used during refueling operations to monitor the core reactivity condition. The installed source range neutron flux monitors are part of the Nuclear Instrumentation System ~~(NIS)~~ (NI). These detectors are located external to the reactor vessel and detect neutrons leaking from the core. The use of ~~portable~~ temporary detectors is permitted, provided the LCO requirements are met. edit 6

The installed source range neutron flux monitors ~~are BF3 detectors operating in the proportional region of the gas filled detector characteristic curve~~. Channels include fission chamber. The detectors monitor the neutron flux in counts per second. The instrument range covers six decades of neutron flux ~~(1E+6 cps) with a [5]% instrument accuracy~~. Instrumentation. The ~~detectors~~ also provide continuous visual indication in the control room ~~and an audible alarm~~ to alert operators to a ~~possible dilution accident~~ significant change in neutron flux. The NIS System is designed in accordance with the criteria presented in Reference 1. ~~If used, portable detectors should be functionally equivalent to the installed NIS source range monitors.~~ edit 6 edit

APPLICABLE SAFETY ANALYSES

~~Two~~ An OPERABLE source range neutron flux monitor~~s~~ ~~are~~ is required to provide ~~a signal~~ indication to alert the operator to unexpected changes in core reactivity, such as may be caused by a boron dilution accident or an improperly loaded fuel assembly (Ref. 1). ¶ The safety analysis of the uncontrolled boron dilution accident is described in Reference 2. The analysis of the uncontrolled boron dilution accident shows that the ~~normally available SDM would not be lost~~ reactor remains subcritical, ~~and there is sufficient time for the operator to take corrective action~~. edit edit edit edit edit

The source range neutron flux monitors are not credited for boron dilution event mitigation in the safety analyses.

The source range neutron flux monitors satisfy Criterion ~~3~~ 4 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 3). 24

(continued)

BASES (continued)

LCO

This LCO requires ~~two~~ one source range neutron flux monitors OPERABLE to ensure that ~~redundant~~ monitoring capability is available to detect changes in core reactivity. 14 6

<INSERT B 3.9-6A>

<INSERT B 3.9-6B>

APPLICABILITY

In MODE 6, the source range neutron flux monitors must be OPERABLE to determine changes in core reactivity. There is no other direct means available to check core reactivity levels. In MODES 2, 3, 4, and 5, ~~these same installed~~ source range detectors and circuitry are also required to be OPERABLE by LCO 3.3.9, "Source Range Neutron Flux." edit 19

<INSERT B 3.9-6C>

ACTIONS

A.1 and A.2

With only one [required] source range neutron flux monitor OPERABLE during CORE ALTERATIONS, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and ~~positive reactivity additions~~ must be suspended immediately. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position. edit 14 31

<INSERT B 3.9-6D>

<INSERT B 3.9-6E>

ANO-243

B.1

With no [required] source range neutron flux monitor OPERABLE, action to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, action shall be continued until a source range neutron flux monitor is restored to OPERABLE status or until the Applicability is exited. edit edit

B.2

With no [required] source range neutron flux monitor OPERABLE, there is no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and positive reactivity additions are not to be made in accordance with Required Actions A.1 and A.2, the core reactivity condition is stabilized until the source range neutron flux monitors are ~~OPERABLE~~ restored to an OPERABLE status. This stabilized condition is ~~determined~~ verified by performing SR 3.9.1.1 to ensure that the required boron concentration exists. edit edit edit edit

(continued)

<INSERT B3.9-6A>

One additional source range neutron flux monitor shall be OPERABLE during CORE ALTERATIONS. This additional requirement ensures redundant monitoring capability when positive reactivity changes are being made to the core.

<INSERT B3.9-6B>

The use of temporary detectors is permitted for purposes of complying with this LCO. If used, the temporary detectors should be functionally equivalent to the installed source range monitors and satisfy applicable Surveillance Requirements.

<INSERT B3.9-6C>

In MODE 1, the neutron flux level is above the indicated range of the monitors. Thus, they are no longer relied upon for reactivity or power level monitoring. Hence, there are no requirements on source range neutron flux monitors in MODE 1.

<INSERT B3.9-6D>

ANO-243

introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1

<INSERT B3.9-6E>

ANO-243

Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

BASES

ACTIONS

B.2 (continued)

once per 12

The Completion Time of 4 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration. ~~The Frequency of once per 12 hours~~ ~~ensures that unplanned changes in boron concentration would be identified.~~ The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this time period.

5 edit

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

normally

SR 3.9.2.1 is the performance of a CHANNEL CHECK, which is a comparison of the parameter indicated on one channel to a similar parameter on other channels. ~~It is based on the assumption that the two indication channels should be consistent with core conditions.~~ Changes in fuel loading and core geometry can also result in significant differences between source range channels, but each channel should be consistent with its local conditions.

<Insert B3.9-7A>

<Insert B 3.9-7B>

edit edit edit

The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified ~~similarly~~ for the same instruments in LCO 3.3.9.

edit

SR 3.9.2.2

SR 3.9.2.2 is the performance of a CHANNEL CALIBRATION every ~~[18]~~ 18 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the source range nuclear instrument is a complete check and re-adjustment of the channel~~s~~, from the pre-amplifier input to ~~the~~ indicator~~s~~. The 18 month Frequency is based on ~~the need to perform this Surveillance during the conditions that apply during a plant outage.~~ ~~Operating~~ industry experience which has shown these components usually pass the Surveillance when performed at the ~~[18]~~ 18 month Frequency.

edit edit edit edit edit

REFERENCES

1. ~~10 CFR 50, Appendix A~~ SAR, Section 1.4, GDC 13, GDC 26, GDC 28, and GDC 29.

23

2. ~~F~~SAR, Section ~~[ ]~~ 14.1.2.4.

edit

3. 10 CFR 50.36.

24

<INSERT B3.9-7A>

It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious.

<INSERT B3.9-7B>

When in MODE 6 with only one channel OPERABLE, a CHANNEL CHECK is still required. However, in this condition, a redundant source range instrument may not be available for comparison. The CHANNEL CHECK provides verification that the OPERABLE source range channel is energized and indicating a value consistent with current unit status.

B 3.9 REFUELING OPERATIONS

B 3.9.3 ~~Containment~~ Reactor Building Penetrations

1

BASES

15

BACKGROUND

During ~~CORE ALTERATIONS or~~ the movement of irradiated fuel assemblies within ~~containment~~ the reactor building ~~with irradiated fuel in containment~~, a release of fission product radioactivity within ~~containment~~ the reactor building will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, ~~this~~ the containment of fission products is accomplished by maintaining ~~containment~~ reactor building OPERABLE as described in LCO 3.6.1, "~~Containment~~ Reactor Building." In MODE 6, the potential for ~~containment~~ reactor building pressurization as a result of an accident is not likely; therefore, requirements to isolate ~~the containment~~ the reactor building from the outside atmosphere can be less stringent. ~~The LCO~~ In order to make this distinction, the penetration requirements are referred to as "~~containment~~ reactor building closure" rather than "~~containment~~ reactor building OPERABILITY." ~~Containment~~ Reactor building closure means that all potential ~~escape~~ direct release paths are closed or capable of being closed. Since there is no significant potential for ~~containment~~ reactor building pressurization, the Appendix J leakage criteria and tests are not required.

edit

The ~~containment~~ reactor building serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR 100. Additionally, the ~~containment~~ reactor building provides radiation shielding from the fission products that may be present in the ~~containment~~ reactor building atmosphere following accident conditions.

edit

edit

edit

The ~~containment~~ reactor building equipment hatch, which is part of the ~~containment~~ reactor building pressure boundary, provides a means for moving large equipment and components into and out of ~~containment~~ the reactor building. During ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within ~~containment~~ the reactor building, the equipment hatch must be ~~held in place by at least four bolts~~ capable of being closed. ~~Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.~~

15 edit edit 10 2

<<INSERT B3.9-8A>>

The ~~containment~~ reactor building air locks, which are also part of the ~~containment~~ reactor building pressure boundary, provide a means for personnel access ~~during~~ in MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "~~Containment~~ Reactor Building Air Locks." Each air lock has a door at ~~both~~ each end~~s~~. The doors are normally interlocked to prevent simultaneous opening when ~~containment~~ the reactor building OPERABILITY is required. During ~~periods of~~ unit shutdown

15

edit edit edit edit edit

(continued)

<INSERT B3.9-8A>

During the movement of irradiated fuel assemblies within the reactor building, administrative controls shall ensure that appropriate personnel are aware that the equipment hatch is open, that a specific individual(s) is designated and available to close the equipment hatch cover following a required evacuation of the reactor building, and that any obstruction(s) (e.g., cables and hoses) that could prevent closure of the equipment hatch cover be capable of being quickly removed (Ref. 1). Should a fuel handling accident occur in the reactor building, the equipment hatch will be closed following evacuation of the reactor building For closure, the equipment hatch cover will be in place with a minimum of four bolts securing the cover to the sealing surface. During outages, a temporary equipment hatch cover may be used in lieu of the permanent equipment hatch cover (Ref. 2).

BASES

BACKGROUND (continued)

OPERABILITY

when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods ~~when frequent containment entry is necessary~~. During ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within ~~containment~~, ~~containment~~ closure ~~is required; therefore~~ the door interlock mechanism may remain disabled, ~~but one air lock door must always remain closed~~.

reactor building

the

the

<INSERT B3.9-9A>

requires that one door in each air lock be capable of being closed.

The requirements on containment penetration closure ensure that a release of fission product radioactivity within ~~containment~~ will be restricted ~~from escaping to the environment. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling.~~

from

the

to within regulatory limits.

The ~~Containment~~ Reactor Building Purge ~~and Exhaust~~ System includes ~~two~~ a Supply ~~subsystems. The normal subsystem includes a [42] inch purge~~ penetration and ~~a [42] inch~~ exhaust penetration. ~~The second subsystem, or minipurge system, includes an [8] inch purge penetration and an [8] inch exhaust penetration.~~ During MODES 1, 2, 3, and 4, the ~~two~~ valves in ~~each of~~ the ~~normal purge~~ Supply and exhaust penetrations are secured in the closed position. ~~The two valves in each of the two minipurge penetrations can be opened intermittently but are closed automatically by the Engineered Safety Feature Actuation System (ESFAS). Neither of~~ the ~~sub~~systems is not subject to a Specification in MODE 5.

Temperature Control

In MODE 6, ~~large air exchangers are necessary to conduct refueling operations~~. The ~~normal [42] inch~~ purge system is may be used for ~~this purpose~~, and all four valves ~~are~~ closed on ~~a~~ an indication of ~~reactor building (RB) high radiation signal in accordance with LCO 3.3.15, "Reactor Building (RB) Purge Isolation High Radiation."~~ by an operator based

<INSERT B3.9-9B>

~~The~~ the other ~~containment~~ penetrations that provide direct access from ~~containment~~ atmosphere to outside atmosphere must be isolated on at least one side. ~~Isolation may be achieved by~~ an OPERABLE ~~automatic~~ isolation valve, ~~or~~ by a closed manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, atmospheric pressure ventilation barrier for the other ~~containment~~ penetrations during fuel movements (Ref. 1).

Move

or capable of being isolated by

or automatic

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edit
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3
3

(continued)

<INSERT B3.9-9A>

During the movement of irradiated fuel assemblies within the reactor building, administrative controls shall ensure that appropriate personnel are aware that both personnel airlock doors are open, that a specific individual(s) is designated and available to close an airlock door following a required evacuation of the reactor building, and any obstruction(s) (e.g., cables and hoses) that could prevent closure of an airlock door be capable of being quickly removed (Ref. 3). Should a fuel handling accident occur inside the reactor building, at least one of the personnel and/or emergency air lock doors will be closed following evacuation of the reactor building.

<INSERT B3.9-9B>

This LCO requires that an OPERABLE radiation monitor be present on the purge exhaust flow path to provide the necessary indication to the operator.

BASES (continued)

APPLICABLE SAFETY ANALYSES 15

During ~~CORE ALTERATIONS or~~ the movement of ~~fuel assemblies~~ irradiated fuel assemblies within ~~containment~~ ~~with irradiated fuel in containment~~ the reactor building, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel. ~~(Ref. 2). Fuel handling accidents, analyzed in Reference 3, include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies.~~ 4 The requirements of LCO 3.9.6, "Refueling Canal Water Level," and the minimum decay time of [100] hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity subsequent to a fuel handling accident results in doses that are within the requirements specified in ~~10 CFR 100. The acceptance limits for offsite radiation exposure are contained in~~ Reference ~~2~~ 4. 1 edit edit edit edit

3.9-06

Reactor building ~~Containment~~ penetrations satisfy Criterion ~~3~~ 4 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref 5). 24

LCO 2

This LCO limits the consequences of a fuel handling accident in ~~containment~~ the reactor building by limiting the potential escape paths for fission product radioactivity from ~~containment~~ the reactor building. The LCO requires any penetration providing direct access from the ~~containment~~ reactor building atmosphere to the outside atmosphere to be closed or capable of being closed by an isolation valve. ~~except for the OPERABLE containment purge and exhaust penetrations. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by the RB purge isolation signal.~~ 3 ~~The OPERABILITY requirements for this LCO ensure that the automatic purge and exhaust valve closure times specified in the FSAR can be achieved and therefore meet the assumptions used in the safety analysis to ensure releases through the valves are terminated such that radiological doses are within the acceptance limit.~~ 3 10

<< INSERT B3.9-10C >>

¶ << INSERT B 3.9-10A >>

¶ << INSERT B 3.9-10B >> 20

APPLICABILITY 15

The reactor building ~~containment~~ penetration requirements are applicable during ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within ~~containment~~ the reactor building because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, ~~containment~~ the reactor building penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within ~~containment~~ the reactor building

(continued)

<INSERT B3.9-10A>

The reactor building personnel airlock doors and/or the equipment hatch may be open during movement of irradiated fuel in the reactor building provided that one door is capable of being closed in the event of a fuel handling accident. Administrative controls shall ensure that appropriate personnel are aware that both personnel airlock doors and/or equipment hatch are open, that a specific individual(s) is designated and available to close an airlock door and the equipment hatch cover following a required evacuation of the reactor building, and any obstruction(s) (e.g. cables and hoses) that could prevent closure of an airlock door and the equipment hatch cover be capable of being quickly removed (Ref. 1 and 3). For closure, the equipment hatch cover will be in place with a minimum of four bolts securing the cover to the sealing surface. During outages, a temporary equipment hatch cover may be used in lieu of the permanent equipment hatch cover (Ref. 2).

<INSERT B3.9-10B>

The definition of "direct access from the reactor building atmosphere to the outside atmosphere" is any path that would allow for the transport of reactor building atmosphere to any atmosphere located outside of the reactor building structure. This includes the Auxiliary Building. As a general rule, closed systems do not constitute a direct path between the reactor building and the outside environments. All permanent and temporary penetration closures should be evaluated to assess the possibility for a release path to the outside environment. For the purpose of determining what constitutes a "direct access" path, no failure mechanisms should be applied to create a scenario which results in a "direct access" path. For example, line breaks, valve failures, power losses or natural phenomena should not be postulated as part of the evaluation process.

<INSERT B3.9-10C>

This LCO requires the reactor building purge isolation valves and the purge exhaust flow path radiation monitor be OPERABLE.

edit

BASES

1 edit

edit

APPLICABILITY (continued)

is ~~are~~ not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on ~~containment~~ reactor building penetration status.

ACTIONS

A.1 ~~and A.2~~ 15

With the ~~containment~~ reactor building equipment hatch, air locks, or any ~~containment~~ reactor building penetration that provides direct access from the ~~containment~~ reactor building atmosphere to the outside atmosphere not in the required status, ~~including the Containment Purge and Exhaust Isolation System not capable of automatic actuation when the purge and exhaust valves are open,~~ the unit must be placed in a condition in which the isolation function is not needed. This is accomplished by immediately suspending ~~CORE ALTERATIONS and~~ movement of irradiated fuel assemblies within ~~containment~~ the reactor building. Performance of ~~these~~ this action~~s~~ shall not preclude moving a component to a safe position.

3 edit edit edit

15 edit edit

<<INSERT B3.9-11A>>

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This Surveillance demonstrates that each of the ~~containment~~ reactor building penetrations required to be in its closed position is in that position. ~~The Surveillance on the open purge and exhaust valves will demonstrate that the valves are not blocked from closing. Also the Surveillance will demonstrate that each valve operator has motive power, which will ensure each valve is capable of being closed by an OPERABLE automatic RB purge isolation signal.~~

edit

3

The Surveillance is performed every 7 days during ~~CORE ALTERATIONS or~~ the movement of irradiated fuel assemblies within the ~~containment~~ reactor building. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. ~~A surveillance before the start of refueling operations will provide two or three surveillance verifications during the applicable period for this LCO.~~

15 edit edit

~~As such,~~ This Surveillance ensures that a postulated fuel handling accident that releases fission product

edit

(continued)

<INSERT B3.9-11A>

These actions remove the potential for an event which may require reactor building closure to prevent a significant radioactivity release.

edit

(1)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1 (continued)

radioactivity within the reactor building ~~containment~~ will not result in a release of fission product radioactivity to the environment in excess of that recommended by Standard Review Plan Section 15.7.4 (Ref. 1, 3 and 6).

edit

SR 3.9.3.2

This Surveillance demonstrates that each reactor building ~~containment purge and exhaust~~ isolation valve actuates to its isolation position on manual initiation ~~or on an actual or simulated high radiation signal~~. ~~The 18 month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements.~~ ~~In LCO 3.3.15, "RB Purge Isolation High Radiation," the isolation instrumentation requires a CHANNEL CHECK every 12 hours and a CHANNEL FUNCTIONAL TEST every 92 days to ensure the channel OPERABILITY during refueling operations. Every 18 months a CHANNEL CALIBRATION is performed. The system actuation response time is demonstrated every 18 months, during refueling, on a STAGGERED TEST BASIS.~~ ~~SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements.~~ This ~~These~~ Surveillance ~~s performed during MODE 6~~ will ensure that the isolation valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the reactor building ~~containment~~.

reactor building isolation found in Section 3.6.

edit

edit

(3)

edit

edit

edit

<INSERT B3.9-12A>

(4)

<INSERT B3.9-12B>

(3)

REFERENCES

~~1. GPU Nuclear Safety Evaluation SE-0002000-001, Rev. 0, May 20, 1988.~~

1. Safety Evaluation Report related to ANO-1 Amendment No. 195, April 16, 1999.

edit

~~2. FSAR, Section [ ].~~

edit

~~3.~~ 6. NUREG-0800, Section 15.7.4, Rev. 1, July 1981.

edit

4. SAR, Section 14.2.2.3.

edit

5. 10 CFR 50.36.

(24)

2. SAR, Section 5.2.2.1.3.

(2)

3. Safety Evaluation Report related to ANO-1 Amendment No. 184, September 20, 1996.

<INSERT B3.9-12A>

ANO-241

The SR is modified by a Note stating that this surveillance is not required to be met for valves in isolated penetrations. The LCO provides the option to close penetrations in lieu of requiring isolation capability.

<INSERT B 3.9-12B>

SR 3.9.3.3

This SR requires a CHANNEL CALIBRATION of the reactor building purge exhaust radiation monitor. The CHANNEL CALIBRATION is a complete check of the instrument loop and sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. The CHANNEL CALIBRATION is performed consistent with the setpoint requirements. The 18 month Frequency is based on operating experience and is consistent with the typical operating cycle.

3.9-07

B 3.9 REFUELING OPERATIONS

B 3.9.4 Decay Heat Removal (DHR) and Coolant Circulation—High Water Level

BASES

23

BACKGROUND

The purposes of the DHR System (Ref. 1) in MODE 6 are to remove decay heat and sensible heat from the reactor Reactor Coolant System (RCS), as required by GDC 34, and to provide mixing of ~~borated~~ coolant, ~~to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and~~ to prevent boron stratification (Ref. 2). Heat is removed from the RCS by circulating reactor coolant through the DHR heat exchanger(s), where the heat is transferred to the ~~Component Cooling~~ Service Water System ~~via the DHR heat exchanger(s)~~. The coolant is then returned to the ~~RCS via the RCS cold leg(s)~~ reactor vessel via the core flood tank injection nozzles. Operation of the DHR System for ~~normal cooldown or decay~~ heat removal is manually accomplished from the control room. The heat removal rate is adjusted by control of the flow of reactor coolant through the DHR heat exchanger(s), ~~and~~ bypassing the heat exchanger(s), and throttling of Service Water through the heat exchanger(s). Mixing of the reactor coolant is ~~maintained~~ provided by ~~this~~ the continuous ~~circulation~~ operation of ~~reactor coolant through~~ the DHR System.

edit
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APPLICABLE SAFETY ANALYSES

~~If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to inadequate cooling of the reactor fuel as a result of a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to boron plating out on components near the areas of the boiling activity, and because of the possible addition of water to the reactor vessel with a lower boron concentration than is required to keep the reactor subcritical. The loss of reactor coolant and the reduction in boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. One train of the DHR System is required to be operational in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to prevent this challenge. The LCO does permit de-energizing the DHR pump for short durations under the condition that the boron concentration is not diluted. This conditional de-energizing of the DHR pump does not result in a challenge to the fission product barrier.~~

<INSERT B3.9-13A>

edit

(continued)

<INSERT B3.9-13A>

Without a DHR loop in operation, the reactor coolant temperature may not be maintained below the boiling point. This could lead to inadequate cooling of the reactor fuel as a result of a loss of coolant in the reactor vessel due to boiling. The loss of reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Operation of one train of the DHR System in MODE 6 is sufficient to prevent this challenge. The LCO does permit de-energizing the DHR pump for short durations under the condition that the boron concentration is not reduced. This conditional de-energizing of the DHR pump does not result in a challenge to the fission product barrier.

3.9-07

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~Although~~ the DHR System ~~does not meet a specific criterion of the NRC Policy Statement, it was identified in the NRC Policy Statement as an important contributor to risk reduction. Therefore, the DHR System is retained as a Specification.~~ Satisfies Criterion 4 of 10CFR50.36 (Ref.3). 24

LCO

Only one DHR loop is required for decay heat removal in MODE 6, with a water level ≥ 23 ft above the top of the ~~reactor vessel flange~~ fuel assemblies seated in the reactor pressure vessel. ~~Only one DHR loop is required to be OPERABLE because the volume of water above the reactor vessel flange provides backup decay heat removal capability. At least one DHR loop must be OPERABLE and in operation to provide:~~ The operating DHR loop provides: 7 EDIT.

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

To be considered ~~An~~ OPERABLE, ~~DHR~~ a DHR loop includes a DHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the ~~low end~~ temperature. The flow path starts in ~~one of~~ the 'A' ~~RCS~~ hot leg~~s~~ and is returned to the reactor vessel via the core flood tank injection nozzles. ~~RCS cold legs.~~ EDIT. EDIT. EDIT.

Additionally, to be considered OPERABLE, each DHR loop ~~is considered OPERABLE if it can be~~ must be capable of being manually aligned (remote or local) in the ~~shutdown cooling~~ decay heat removal mode ~~for removal of decay heat. Operation of one subsystem can maintain the reactor coolant temperature as required.~~ EDIT. EDIT. EDIT.

The LCO is modified by a Note that allows the required DHR loop to be removed from operation for up to 1 hour in an 8 hour period, provided no operations that would cause ~~reduction of the RCS boron concentration is in progress.~~ <INSERT B 3.9-14A> ~~Boron concentration reduction~~ <INSERT B 3.9-14B> is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This allowance permits operations such as core mapping or maintenance ~~or~~ alterations in the vicinity of the reactor vessel ~~hot leg~~ nozzles and RCS to DHR isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling ~~cavity~~ canal. 31 edit edit edit

ANO-243

(continued)

<INSERT B3.9-14A>

ANO-243

introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1

<INSERT B3.9-14B>

ANO-243

with coolant at boron concentrations less than required to assure the RCS boron concentration is maintained

BASES (continued)

APPLICABILITY

3.9-07

fuel assemblies seated in the reactor pressure vessel

fuel assemblies seated in the reactor vessel,

One DHR loop must be OPERABLE and in operation in MODE 6, with the water level ≥ 23 ft above the top of ~~the reactor vessel flange~~, to provide decay heat removal. ~~The 23 ft water level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.6, "Refueling Canal Water Level."~~ Requirements for the DHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). DHR loop requirements in MODE 6, with the water level < 23 ~~ft~~ feet above the top of the ~~reactor vessel flange~~ are located in LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation—Low Water Level."

7

EDIT.

ACTIONS

~~DHR loop requirements are met by having one DHR loop OPERABLE and in operation, except as permitted in the Note to the LCO.~~

EDIT.

A.1

If DHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. ~~Reduced boron concentrations can occur by adding water with a lower boron concentration than that contained in the RCS. Therefore, actions that reduce boron concentration shall be suspended immediately.~~

ANO-243

<INSERT B 3.9-15A>

31

A.2

If DHR loop requirements are not met, actions shall be taken immediately to suspend the loading of irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. ~~A minimum refueling~~ water level 23 ~~ft~~ feet above the ~~reactor vessel flange~~ fuel assemblies seated in the reactor vessel provides an adequate available heat sink. Suspending any operation that would increase decay heat load, such as loading ~~a~~ an irradiated fuel assembly, is prudent under this condition.

Canal

7

EDIT.

(continued)

<INSERT B3.9-15A>

ANO-243

Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

3.9-07

BASES

ACTIONS (continued)

A.3

If DHR loop requirements are not met, actions shall be initiated immediately in order to satisfy DHR loop requirements.

edit

<< INSERT B 3.9-16A >> —>

A.4

If DHR loop requirements are not met, all ~~containment~~ reactor building penetrations providing direct access from the ~~containment~~ atmosphere to outside atmosphere shall be closed within 4 hours.

1

edit

<< INSERT B 3.9-16B >> —>

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that the DHR loop is in operation and circulating reactor coolant. ~~The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core.~~ The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the DHR System.

<INSERT B 3.9-16C>

11

edit

REFERENCES

2. ~~F~~SAR, Section ~~[ ]~~ 9.5

edit

3. 10 CFR 50.36.

24

1. SAR, Section 1.4.

23

<INSERT B3.9-16A>

Restoration of one decay heat removal loop is required because this is the only active method of removing decay heat. Dissipation of decay heat through natural convection to the large inventory of water in the refueling canal should not be relied upon for an extended period of time. The immediate Completion Time reflects the importance of restoring an adequate decay heat removal loop.

<INSERT B3.9-16B>

If no means of decay heat removal can be restored, the core decay heat could raise temperatures and cause boiling in the core which could result in increased levels of radioactivity in the reactor building atmosphere. Closure of the penetrations providing access to the outside atmosphere will prevent the uncontrolled release of radioactivity to the environment.

<INSERT B3.9-16C>

Verification includes flow rate, temperature, or pump status monitoring, which help assure that forced flow is providing heat removal.

B 3.9 REFUELING OPERATIONS

B 3.9.5 Decay Heat Removal (DHR) and Coolant Circulation—Low Water Level

BASES

23

BACKGROUND

(Ref. 1) and

The purposes of the DHR System in MODE 6 are to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of ~~borated~~ coolant ~~to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and~~ to prevent boron stratification (Ref 2). Heat is removed from the RCS by circulating reactor coolant through the DHR heat exchanger(s), where the heat is transferred to the ~~Component Cooling~~ Water System ~~via the DHR heat exchanger~~. The coolant is then returned to the ~~RCS~~ via the ~~RCS cold leg(s)~~. Operation of the DHR System for normal cooldown/decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by control of the flow of reactor coolant through the DHR heat exchanger(s) ~~and~~ bypassing the heat exchanger(s). Mixing of the reactor coolant is ~~maintained~~ by ~~this~~ continuous ~~circulation~~ of ~~reactor coolant through~~ the DHR System.

the reactor

Service

reactor vessel

Core flood tank injection nozzles.

and by throttling of Service Water through the heat exchanger(s).

provided

the

operation

edit
edit
edit
edit
edit
edit

APPLICABLE SAFETY ANALYSES

<INSERT B 3.9-17A>

~~If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to inadequate cooling of the reactor fuel due to resulting loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to boron plating out on components near the areas of the boiling activity, and because of the possible addition of water to the reactor vessel with a lower boron concentration than is required to keep the reactor subcritical. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Two trains of the DHR System are required to be OPERABLE, and one is required to be in operation, to prevent this challenge.~~

edit

Satisfies Criterion 4

24

~~Although~~ The DHR System ~~does not meet a specific criterion of the NRC Policy Statement, it was identified in the NRC Policy Statement~~ as an important contributor to risk

10CFR 50.36 (Ref. 3),

(continued)

<INSERT B3.9-17A>

Without a DHR loop in operation, the reactor coolant temperature may not be maintained below the boiling point. This could lead to inadequate cooling of the reactor fuel as a result of a loss of coolant in the reactor vessel due to boiling. The loss of reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Operation of one train of the DHR System in MODE 6 is sufficient to prevent this challenge. However, without a large water inventory to provide a backup means of decay heat removal, an additional train of the DHR System is required to be OPERABLE in order to provide a backup

BASES

APPLICABLE SAFETY ANALYSES (continued) ~~reduction. Therefore, the DHR System is retained as a Specification.~~ edit

LCO

In MODE 6, with the water level < 23 ~~ft~~ feet above the top of the ~~reactor vessel flange~~ fuel assemblies seated in the reactor vessel, two DHR loops must be OPERABLE. Additionally, one DHR loop must be in operation to provide: edit 8

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and 29

c. Indication of reactor coolant temperature. 30

<INSERT B 3.9-18A> To be considered

~~An~~ a OPERABLE DHR loop ~~consists~~ must of a DHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the ~~low end~~ 'A' temperature. The flow path starts in ~~one of~~ the ~~RCS~~ hot legs and is returned to the ~~RCS cold legs~~ reactor vessel via the core flood tank injection nozzles. edit edit 9

<INSERT B3.9-18B>

APPLICABILITY

Two DHR loops are required to be OPERABLE, and one in operation in MODE 6, with the water level < 23 ~~ft~~ feet above the top of the ~~reactor vessel flange~~ fuel assemblies seated in the reactor vessel, to provide decay heat removal. Requirements for the DHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). DHR loop requirements in MODE 6 ~~with the water level ≥ 23 ft above the top of the reactor vessel flange, are located in LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation—High Water Level."~~ edit 8 edit

ACTIONS

A.1 and A.2

With fewer than the required loops OPERABLE, action shall be immediately initiated and continued until the DHR loop is restored to OPERABLE status or until ≥ 23 ~~ft~~ feet of water level is established above the ~~reactor vessel flange~~ fuel assemblies seated in the reactor vessel. When the water level is established at ≥ 23 ~~ft~~ feet above the ~~reactor~~ edit 8 edit

(continued)

ANO-245
ANO-244
ANO-358
3.9-07
ANO-358

<INSERT B3.9-18A>

ANO-244

This LCO is modified by two Notes. Note 1 permits the DHR pumps to be de-energized for ≤ 15 minutes when switching from one train to another. The circumstances for stopping both DHR pumps are to be limited to situations when the outage time is short and the core outlet temperature is maintained > 10 degrees F below saturation temperature. The Note prohibits boron dilution or draining operations when DHR forced flow is stopped.

ANO-245

The second Note allows one DHR loop to be inoperable for a period of 2 hours provided the other loop is OPERABLE and in operation. Prior to declaring the loop inoperable, consideration should be given to the existing plant configuration. This consideration should include that the core time to boil is short, there is no draining operation to further reduce RCS water level and that capability exists to inject borated water into the reactor vessel. This permits surveillance tests to be performed on the inoperable loop during a time when these tests are safe and possible.

<INSERT B3.9-18B>

Additionally, to be considered OPERABLE, each DHR loop must be capable of being manually aligned (remote or local) in the decay heat removal mode for removal of decay heat. Operation of one subsystem can maintain the reactor coolant temperature as required.

Both DHR pumps may be aligned to the Borated Water Storage Tank (BWST) to support filling of the refueling canal or the performance of required testing.

BASES

ACTIONS

A.1 and A.2 (continued)

~~vessel flange~~, the Applicability will change to that of LCO 3.9.4, and only one DHR loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary ~~for an operator to initiate corrective actions to restore the required forced circulation or water level.~~ due to the increased risk of operating without a large available heat sink for decay heat removal

edit 8

B.1

3.9-07

If no DHR loop is in operation or no DHR loop is OPERABLE, there will be no forced circulation to provide mixing to establish uniform boron concentrations. ~~Reduced boron concentration can occur by adding water with a lower boron concentration than that contained in the RCS. Therefore, actions that reduce boron concentration shall be suspended immediately.~~ 31

AND-243

<INSERT B 3.9-19A>

B.2

3.9-07

If no DHR loop is in operation or no DHR loop is OPERABLE, actions shall be initiated immediately and continued without interruption to restore one DHR loop to OPERABLE status and operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE DHR loops and one operating DHR loop should be accomplished expeditiously.

If no DHR loop is OPERABLE or in operation, alternate actions shall have been initiated immediately under Condition A to establish ≥ 23 ft of water above the top of the ~~reactor vessel flange~~ fuel assemblies seated in the reactor vessel. Furthermore, when the LCO cannot be fulfilled, alternate decay heat removal methods, as specified in the unit's Abnormal and Emergency Operating Procedures, should be implemented. ~~This includes decay heat removal using the charging or safety injection pumps through the Chemical and Volume Control System with consideration for the boron concentration.~~ The method used to remove decay heat should be the most prudent as well as the safest choice, based upon unit conditions. The choice could be different if the reactor vessel head is in place rather than removed. 8 32

(continued)

<INSERT B3.9-19A>

ANO-243

Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

BASES

ACTIONS (continued)

3.9-07

B.3 DHR

If no ~~RHR~~ DHR loop is in operation, all ~~containment~~ reactor building penetrations providing direct access from the ~~containment~~ reactor building atmosphere to the outside atmosphere must be closed within 4 hours. With the ~~RHR~~ DHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the ~~containment~~ reactor building atmosphere. Closing ~~containment~~ reactor building penetrations that are open to the outside atmosphere ensures that dose limits are not exceeded. edit edit 1

The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

3.9-07 <INSERT B 3.9-20A>

This Surveillance demonstrates that one DHR loop is in operation. ~~The flow rate is determined by the flow rate necessary to provide efficient decay heat removal capability and to prevent thermal and boron stratification in the core.~~ 11

~~In addition, during operation of the DHR loop with the water level in the vicinity of the reactor vessel nozzles, the DHR loop flow rate determination must also consider the DHR pump suction requirement.~~ The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator to monitor the DHR System in the control room.

3.9-07

SR 3.9.5.2

Verification that ~~the~~ each required pump is ~~OPERABLE~~ available ensures that an additional DHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. edit 17 17

<INSERT B3.9-20B> The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. SAR, Section 1.4. 23 edit
2. ~~1.~~ ~~F~~SAR, Section ~~[ ]~~ 9.5
3. 10 CFR 50.36. 24

<INSERT B3.9-20A>

3.9-07 Verification includes flow rate, temperature, or pump status monitoring, which help assure that forced flow is providing heat removal.

<INSERT B3.9-20B>

Alternatively, verification that a DHR pump is in operation as required by SR 3.9.4.1 also verifies proper breaker alignment and power availability.

B 3.9 REFUELING OPERATIONS

B 3.9.6 Refueling Canal Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies ~~or performance of CORE ALTERATIONS, except during latching and unlatching of CONTROL ROD drive shafts,~~ within ~~containment~~ the reactor building requires a minimum water level of 23 ~~ft~~ feet above the top of the ~~reactor vessel flange~~ irradiated fuel assemblies seated within the reactor pressure vessel. During refueling, this maintains sufficient water level in the containment, the refueling canal, the fuel transfer canal, the refueling cavity, and the spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident within 10 CFR 100 limits, as provided by the guidance of Reference 3.

edit

APPLICABLE SAFETY ANALYSES

During ~~CORE ALTERATIONS and during~~ movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of the fuel handling accident in ~~containment~~ the reactor building postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 ~~ft~~ feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain ~~10%~~ 12% of the total fuel rod iodine inventory (Ref. ~~1~~ 2).

The fuel handling accident analysis inside ~~containment~~ the reactor building is described in Reference 2. With a minimum water level of 23 ~~ft~~ feet, and a minimum decay time of ~~72~~ 100 hours prior to fuel handling, the analysis ~~and test programs~~ demonstrates that the iodine release due to a postulated fuel handling accident is adequately captured by the water, and offsite doses are maintained within allowable limits (Ref. 3).

Refueling canal water level satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10CFR50.36 (Ref. 4).

(continued)

BASES (continued)

LCO

A minimum refueling ~~cavity~~ canal water level of 23 ~~ft~~ feet above the ~~reactor vessel flange~~ top of the irradiated fuel assemblies seated in the reactor pressure vessel is required to ensure that the radiological consequences of a postulated fuel handling accident inside ~~containment~~ the reactor building are within acceptable limits as provided by 10 CFR 100. edit 22 1

APPLICABILITY

LCO 3.9.6 is applicable during ~~CORE ALTERATIONS, except during latching and unlatching of CONTROL ROD drive shafts, and when moving~~ movement of irradiated fuel assemblies within the ~~containment~~ the reactor building. The LCO minimizes the possibility of a fuel handling accident in ~~containment~~ the reactor building that is beyond the assumptions of the safety analysis. If irradiated fuel is not present in ~~containment~~ the reactor building, there can be no significant radioactivity release as a result of a postulated fuel handling accident in the reactor building. ~~Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.14, "Fuel Storage Pool Water Level."~~ 22 1 edit

ACTIONS

A.1 ~~and A.2~~

With a water level of < 23 ~~ft~~ feet above the top of the irradiated fuel assemblies seated with the reactor pressure vessel ~~flange~~, all operations involving ~~CORE ALTERATIONS or~~ the movement of irradiated fuel assemblies shall be suspended immediately to ensure that a fuel handling accident cannot occur. 22

The suspension of ~~CORE ALTERATIONS and~~ irradiated fuel movement shall not preclude completion of movement of a component to a safe position.

~~A.2~~

~~In addition to immediately suspending CORE ALTERATIONS or movement of irradiated fuel, action to restore refueling cavity water level must be initiated immediately.~~ 21

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 23 ~~ft~~ feet above the top of the irradiated fuel assemblies seated within the ~~reactor~~ pressure vessel ~~flange~~ ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. Water at the required level above the top of the ~~reactor~~ pressure vessel ~~flange~~ limits the consequences of damaged fuel rods that are postulated to result from a postulated fuel handling accident inside ~~containment~~ the reactor building (Ref. 2). edit 22 1

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls ~~of valve positions~~, which make significant unplanned level changes unlikely. edit.

REFERENCES

1. Regulatory Guide 1.25, March 23, 1972.
2. ~~F~~SAR Section ~~[ ]~~ 14.2.2.3. edit.
3. 10 CFR 100.10.
4. 10 CFR 50.36. 24

4.0 DESIGN FEATURES

4.1 Site Location

The site for Arkansas Nuclear One is located in Pope County, Arkansas on the north bank of the Dardanelle Reservoir (Arkansas River), approximately 6 miles west-northwest of Russellville, AR. The exclusion area boundary shall have a radius of 0.65 statute miles from the Unit 1 reactor building.

4.0 DESIGN FEATURES

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 177 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Assemblies

The reactor core shall contain 60 safety and regulating CONTROL ROD assemblies and 8 APSR assemblies. The CONTROL ROD assembly control material shall be a silver-indium-cadmium alloy and the APSR assembly control material shall be an Inconel alloy, as approved by the NRC.

DESIGN FEATURES

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 4.1 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.6.2.4.3 of the SAR;

c. A nominal 10.65 inch center to center distance between fuel assemblies placed in the storage racks;

d. New or partially spent fuel assemblies with a discharge burnup in the "acceptable range" of Figure 3.7.15-1 allowed unrestricted storage in either fuel storage rack Region 1 or Region 2; and

e. New or partially spent fuel assemblies with a discharge burnup in the "unacceptable range" of Figure 3.7.15-1 stored in Region 1, or in checkerboard configuration in Region 2.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 4.1 weight percent;

b. $k_{eff} \leq 0.95$ under normal conditions, which includes an allowance for uncertainties as described in Section 9.6.2.4.3 of the SAR;

c. $k_{eff} \leq 0.98$ with optimum moderation, which includes an allowance for uncertainties as described in Section 9.6.2.4.3 of the SAR;

d. A nominal 21 inch center to center distance between fuel assemblies placed in the storage racks; and

e. Ten interior storage cells, as shown in Figure 4.3.1.2-1, precluded from use during fuel storage.

DESIGN FEATURES

4.3 Fuel Storage

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 397 ft.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 968 fuel assemblies.

<-----NORTH

		NO			NO		
			NO	NO			
			NO	NO			
			NO	NO			
		NO			NO		

"NO" Indicates a location in which fuel loading is prohibited.

Figure 4.3.1.2-1 (page 1 of 1)

Fresh Fuel Storage Rack
Loading Pattern

CTS DISCUSSION OF CHANGES

ITS Section 4.0: Design Features

ADMINISTRATIVE

A1 The designated change represents a non-technical, non-intent change to the Arkansas Nuclear One, Unit 1 Current Technical Specifications (CTS) made to make the ANO-1 Improved Technical Specifications (ITS) consistent with the Babcock and Wilcox (B&W) revised Standard Technical Specification (RSTS), NUREG-1430, Revision 1. This change does not alter the requirements of the CTS or RSTS. Examples of this type of change include: wording preference; convention adoption; editorial, numbering and formatting changes; and hierarchy structure.

A2 The ANO-1 CTS Bases will be administratively deleted in their entirety in favor of the NUREG-1430 Bases. The CTS Bases will be reviewed for technical content that will be identified for retention in the ITS Bases.

A3 The "less than" requirements for k_{eff} ,in CTS 5.4.1.1, have been revised to ≤ in ITS 4.3.1.2. These are considered to be essentially equivalent since the parameter can be less than than the limit, but be so close as to be imperceptible. This change is consistent with design basis and with NUREG-1430.

A4 The statement regarding the applicability of the provisions of Specification 3.0.3 is not retained. This statement is no longer required since the Specification is moved to the Design Features section for which LCO 3.0.3 is not applicable. Since there is no change in the application of the requirements, this change is considered administrative.

A5 Not used.

A6 Not used.

TECHNICAL CHANGE -- MORE RESTRICTIVE

M1 CTS 5.4.2 is revised to include additional information to describe the nominal center to center distance between fuel assemblies placed in the spent fuel storage racks. This change provides a safe geometric spacing in the spent fuel storage racks. There are only high density spent fuel storage racks provided at ANO-1 as discussed in SAR Section 9.6.2.3. Therefore, there is no need to differentiate between high density and low density racks in ITS 4.3.1, nor to provide any information on low density storage racks pursuant to RSTS 4.3.1.1.d. This change is consistent with RSTS 4.3.1.1.c.

M2 CTS 5.4.2 is revised to include additional information described in NUREG 4.3.2 and
4.0-02 4.3.3 concerning the number of available storage containers and the minimum drainage level of the ANO-1 spent fuel pool. This change ensures the aforementioned pool designs are maintained and controlled within ITS and is consistent with NUREG-1430.

CTS DISCUSSION OF CHANGES

TECHNICAL CHANGE -- LESS RESTRICTIVE

L None

LESS RESTRICTIVE -- ADMINISTRATIVE DELETION OF REQUIREMENTS

LA1 This information has been moved to a licensee controlled document such as the Technical Requirements Manual (TRM), Safety Analysis Report (SAR), etc. This information provides details of the method of implementation which are not directly pertinent to the actual requirement. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The details relocated to the TRM will be controlled by 10 CFR 50.59. The details relocated to the SAR will be controlled by 10 CFR 50.59 and 50.71. This change is consistent with NUREG-1430.

CTS Location	New Location
5.1	SAR 1.2.1
5.1	SAR 2.2
5.2.1	SAR 5.2.1
5.2.1	SAR 14.2.2.5.5
5.2.2	SAR 5.2.5
5.2.3	SAR 6.5
5.3.1.2	SAR Table 3-2
5.3.1.2	SAR 3.2.2.1.1
5.3.1.3	SAR Table 3-2
5.3.1.4	SAR 3.2.1
5.3.1.4	SAR Fig. 3-60
5.3.1.4	SAR 3A.3
5.3.1.4	SAR Fig. 3A-4
5.3.1.5	SAR 3.2.4.2
5.3.1.5	SAR Fig. 3-2
5.3.1.6	TRM
5.3.2.1	SAR 4.1.3
5.3.2.2	SAR 4.1.2
5.3.2.3	TRM
5.4.1.1	TRM
5.4.1.2	SAR 9.6.2.1
5.4.2.2	SAR 5.1.2.1.2

4.3.1.1.a 3.8.15 Storage in the spent fuel pool shall be restricted to fuel assemblies having initial enrichment less than or equal to 4.1 w/o U-235. The provisions of Specification 3.0.3 are not applicable. A4

<LATER> (3.7) LATER

4.3.1.1.d 4.3.1.1.e 3.8.16 Storage in Region 2 (as shown on Figure 3.8.1) of the spent fuel pool shall be further restricted by burnup and enrichment limits specified in Figure 3.8.2. In the event a checkerboard storage configuration is deemed necessary for a portion of Region 2, vacant spaces adjacent to the faces of any fuel assembly which does not meet the Region 2 burnup criteria (non-restricted) shall be physically blocked before any such fuel assembly may be placed in Region 2. This will prevent inadvertent fuel assembly insertion into two adjacent storage locations. The provisions of Specification 3.0.3 are not applicable. A4 +LATER

<LATER> (3.7)

<LATER> (3.7) 3.8.17 The boron concentration in the spent fuel pool shall be maintained (at all times) at greater than 1600 parts per million. LATER

<LATER> (3.7) 3.8.18 During the handling of irradiated fuel, the control room emergency air conditioning system and the control room emergency ventilation system shall be operable as required by Specification 3.9. LATER

Bases

Detailed written procedures will be available for use by refueling personnel. These procedures, the above specifications, and the design of the fuel handling equipment as described in Section 9.6 of the FSAR incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety. If no change is being made in core geometry, one flux monitor is sufficient. This permits maintenance on the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. A2

The requirement that at least one decay heat removal loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel at the refueling temperature (normally 140°F), and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.(1)

The requirement to have two decay heat removal loops operable when there is less than 23 feet of water above the core, ensures that a single failure of the operating decay heat removal loop will not result in a complete loss of decay heat removal capability. With the reactor vessel head removed and 23 feet of water above the core, a large heat sink is available for core cooling, thus in the event of a failure of the operating decay heat removal loop, adequate time is provided to initiate emergency procedures to cool the core.

The shutdown margin indicated in Specification 3.8.4 will keep the core subcritical, even with all control rods withdrawn from the core.(2) Although the refueling boron concentration is sufficient to maintain the core keff ≤ 0.99 if all the control rods were removed from the core, only a few control rods will be removed at any one time during fuel shuffling and

4.1

5.0 **DESIGN FEATURES**

Specifications for design features are intended to cover characteristics of importance to each of the physical barriers, and to the maintenance of safety margins in the design.

A1

5.1 SITE

Applicability

Applies to the location and extent of the exclusion area.

Objective

To define the location and the size of the site area as pertains to safety.

Specification

4.1 Arkansas Nuclear One-Unit 1 is located on a site consisting of approximately 1100 acres which provides for 0.65 statute mile exclusion radius from the reactor building. This exclusion area includes certain portions of the bed and banks of the Dardanelle Reservoir which are owned by the Federal Government. An easement authorizes exclusion of all persons from these areas during periods of emergency. The site is approximately 6 statute miles WNW from the City of Russellville (Latitude 35°-18'-36" N, Longitude 93°-13'-53" W) in an area characterized by remoteness from population centers.

LA1

SAR

REFERENCE

FSAR, Section 2.2

4.0

LA1
SAR

5.2 REACTOR BUILDING

Applicability

Applies to those design features of the reactor building relating to operational and public safety.

Objective

To define the significant design features of the reactor building structure, reactor building isolation system, and penetration room ventilation system.

Specification

5.2.1 Reactor Building Structure

The reactor building completely encloses the reactor and the associated reactor coolant system. It is a fully continuous reinforced concrete structure in the shape of a cylinder with a shallow domed roof and a flat foundation slab. The cylindrical portion is prestressed by a post tensioning system consisting of horizontal and vertical tendons. The dome has a three-way post tensioning system. The foundation slab is conventionally reinforced with high strength reinforcing steel. The entire structure is lined with 1/4" welded steel plate to provide vapor tightness.

The internal net free volume of the reactor building is approximate 1.81 x 10^6 cu. ft. The approximate inside dimensions are: diameter 116'; height--207'. The approximate thickness of the concrete form the buildings are: cylindrical wall--3-3/4'; dome--3-1/4'; and the foundation slab--9'.

The concrete reactor building structure provides adequate shielding for both normal operation and accident situations. Design pressure and temperature are 59 psig and 286 F, respectively.

The reactor building is designed for an external atmospheric pressure of 3.0 psi greater than the internal pressure. This corresponds to a margin of 0.5 psi above the differential pressure that could be developed if the building is sealed with an internal temperature of 110 F and it is subsequently cooled to an internal temperature of less than 50 F. Since the building is designed for this pressure differential, vacuum breakers are not required.

The principal design basis for the structure is that it be capable of withstanding the internal pressure resulting from a loss of coolant accident, as defined in FSAR Section 14 with no loss of integrity. In this event the total energy contained in the water of the reactor coolant system is

4.0

LA1
SAR

assumed to be released into the reactor building through a break in the reactor coolant piping. Subsequent pressure behavior is determined by the building volume, engineered safety features, and the combined influence of energy sources and heat sinks. ([1])

5.2.2 Reactor Building Isolation System

Leakage through all fluid penetrations not serving accident-consequence-limiting systems is to be minimized by a double barrier so that no single, credible failure or malfunction of an active component can result in loss-of-isolation or intolerable leakage. The installed double barriers take the form of closed piping systems, both inside and outside the reactor building and various types of isolation valves. ([2])

<LATER> (3.7)

LATER

5.2.3 Penetration Room Ventilation System

This system is designed to collect, control, and minimize the release of radioactive material from the reactor building to the environment in post-accident conditions. It may also operate intermittently during normal conditions as required to maintain satisfactory temperature in the penetrations rooms. When the system is in operation, a slightly negative pressure will be maintained in the penetration room to assure inleakage. ([3])

REFERENCES:

(1) FSAR Section 5.1

(2) FSAR Section 5.2.5

(3) FSAR Section 6.5

LA1
SAR

<LATER> (3.7)

LATER

4.2

5.3 REACTOR

Specification

5.3.1 Reactor Core

4.2.1

5.3.1.1 The reactor shall contain 177 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide ~~pellets.~~ (UO_2) as fuel material. Limited substitutions of stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions. A1

5.3.1.2 ~~The reactor core approximates a right circular cylinder with an equivalent diameter of 128.9 inches and an active height of 144 inches. The active fuel length is approximately 142 inches.(2)~~ LA1 SAR

5.3.1.3 ~~The average enrichment of the initial core is a nominal 2.62 weight percent of U-235. Three fuel enrichments are used in the initial core. The highest enrichment is less than 3.5 weight percent U-235.~~ LA1 SAR

4.2.2

5.3.1.4 There are 60 ~~full-length~~ control rod assemblies (CRA) and 8 axial power shaping rod assemblies (APSRAs) ~~distributed in the reactor core as shown in FSAR Figure 3-60. The full-length CRAs contain a 134-inch length of~~ control material of silver-indium-cadmium alloy, and the ~~clad with stainless steel. Each~~ APSRAs contain ~~a 63-inch length of~~ Inconel-600 alloy ~~(3)~~, as approved by the NRC. LA1 SAR A1

5.3.1.5 ~~The initial core had 68 burnable poison spider assemblies with similar dimensions as the full-length control rods. The cladding is Zircaloy-4 filled with alumina-boron and placed in the core as shown in FSAR Figure 3-2.~~ LA1 SAR

5.3.1.6 ~~Reload fuel shall conform to the design and evaluation described in FSAR and shall not exceed an enrichment of 4.1 weight percent of U-235.~~ LA1 SAR

5.3.2 ~~Reactor Coolant System~~

5.3.2.1 ~~The reactor coolant system is designed and constructed in accordance with code requirements.(4)~~ LA1 SAR

4.2

LA1
SAR

5.3.2.2 The reactor coolant system and any connected auxiliary systems exposed to the reactor coolant conditions of temperature and pressure, are designed for a pressure of 2500 psig and a temperature of 650 F. The pressurizer and pressurizer surge line are designed for a temperature of 670 F.(5)

LA1
TRM

5.3.2.3 The reactor coolant system volume is less than 12,200 cubic feet.

A1

REFERENCES:

(1) FSAR, Section 3.2.1

(2) FSAR, Section 3.2.2

(3) FSAR, Section 3.2.4.2

(4) FSAR, Section 4.1.3

(5) FSAR, Section 4.1.2

4.3

5.4 NEW AND SPENT FUEL STORAGE FACILITIES

A1

~~Applicability~~

~~Applies to storage facilities for new and spent fuel assemblies.~~

~~Objective~~

~~To assure that both new and spent fuel assemblies will be stored in such a manner that an inadvertent criticality could not occur.~~

Specification

5.4.1 New Fuel Storage

4.3.1.2.a-e

1. New fuel assemblies may be stored in the Fresh Fuel Storage Rack (FFSR). The FFSR consists of ~~a nine by eight array of~~ storage cells on nominal center to center distance of 21 inches in both directions. Ten interior storage cells, as shown in Figure 5.4-1, are precluded from use ~~and will be physically blocked prior to any storage in the fresh fuel rack~~. This configuration is sufficient to maintain a K_{eff} of less than 0.98 with optimum moderation and 0.95 under normal conditions, based on fuel with an enrichment of 4.1 weight percent U-235.

LA1 TRM

LA1 TRM

A3

which includes an allowance for uncertainties as described in Section 9.6.2.4.3 of the SAR.

4.0-01

A1

4.3.1.1.d
4.3.1.1.e

2. New fuel may also be stored in the spent fuel pool ~~or in its shipping containers~~.

LA1 SAR

5.4.2 Spent Fuel Storage

4.3.1.1.b

1. The spent fuel racks are designed and shall be maintained so that the calculated effective multiplication factor is no greater than 0.95 (including all known uncertainties) when the pool is flooded with unborated water as described in Section 9.6.2.4.3 of the SAR

A1

4.0-09

~~2. The spent fuel pool and the new fuel pool racks are designed as seismic Class I equipment.~~

LA1 SAR

~~REFERENCES~~

~~FSAR, Section 9.6~~

A1

<Add 4.3.1.1.c>

M1

<Add 4.3.2 & 4.3.3>

M2

4.0-02

Fig. 4.3.1.2-1 ~~FIGURE 3.4-1~~ ANO FFSR LOADING PATTERN A1

<-----NORTH

		NO			NO		
			NO	NO			
			NO	NO			
			NO	NO			
		NO			NO		

"NO" Indicates a location in which fuel loading is prohibited.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

"R" - Relocation of requirements:

Relocating requirements which do not meet the Technical Specification selection criteria to documents with an established control program allows the Technical Specifications to be reserved only for those conditions or limitations upon reactor operation which are necessary to adequately limit the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety, thereby focusing the scope of Technical Specifications.

Therefore, requirements which do not meet the Technical Specification selection criteria in 10 CFR 50.36 have been relocated to other controlled license basis documents. This regulation addresses the scope and purpose of Technical Specifications. In doing so, it establishes a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in Technical Specifications. These criteria are as follows:

Criterion 1: Installed instrumentation that is used to detect and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2: A process variable that is an initial condition of a design basis accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3: A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission barrier.

Criterion 4: A structure, system or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

The application of these criteria is provided in the "Application of Selection Criteria to the ANO-1 Technical Specifications." Requirements which met the criteria have been included in the proposed improved Technical Specifications. Entergy Operations proposes to remove the requirements which do not meet the criteria from the Technical Specifications and relocate the requirements to a suitable owner controlled document. The requirements in the relocated Specifications will not be affected by this Technical Specification change. Entergy Operations will initially continue to perform the required operation and maintenance to assure that the requirements are satisfied. Relocating specific requirements for systems or variables will have no impact on the system's operability or the variable's maintenance, as applicable.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

License basis document control mechanisms, such as 10 CFR 50.59, 10 CFR 50.54(a)(3), and ITS Section 5, "Administrative Controls," will be utilized for the relocated Specifications as they will be placed in other controlled license basis documents. This would allow Entergy Operations to make changes to these requirements, without NRC approval, as allowed by the applicable regulatory requirements. These controls are considered adequate for assuring structures, systems and components in the relocated Specifications are maintained operable and variables in the relocated Specifications are maintained within limits.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as indicated below:

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change relocates requirements and surveillances for structures, systems, components or variables which did not meet the criteria for inclusion in Technical Specifications as identified in the Application of Selection Criteria to the ANO-1 Technical Specifications. The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected structures, systems, components or variables will be relocated from the Technical Specifications to an appropriate administratively controlled license basis document and maintained pursuant to the applicable regulatory requirements. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or change in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the affected requirement will be relocated to an owner controlled license basis document for which future changes will be evaluated pursuant to the requirements of the applicable regulatory requirements. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

"A" - Administrative changes to requirements:

Reformatting and rewording the remaining requirements in accordance with the style of the improved Babcock & Wilcox Standard Technical Specifications in NUREG-1430 will make the Technical Specifications more readily understandable to plant operators and other users. Application of the format and style will also assure consistency is achieved between specifications. As a result, the reformatting and rewording of the Technical Specifications has been performed to make them more readily understandable by plant operators and other users. During this reformatting and rewording process, no technical changes (either actual or interpretational) to the Technical Specifications were made unless they were identified and justified.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as indicated below:

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change involves reformatting and rewording of the existing Technical Specifications. The reformatting and rewording process involves no technical changes to existing requirements. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   The proposed change will not significantly reduce the margin of safety because it has no impact on any safety analysis assumptions. This change is administrative in nature. As such, there is no technical change to the requirements and therefore, there is no significant reduction in the margin of safety.

"LA" - Less restrictive, Administrative deletion of requirements:

Portions of some Specifications provide information that is descriptive in nature regarding the equipment, system(s), actions or surveillances. This information is proposed to be deleted from the specification and relocated to other license basis documents which are under licensee control. These documents include the TS Bases, Safety Analysis Report (SAR), Technical Requirements Manual, and Programs and Manuals identified in ITS Section 5, "Administrative Controls." The removal of descriptive information is permissible, because the documents containing the relocated information will be controlled through the applicable process provided by the regulatory requirements, e.g., 10 CFR 50.59, 10 CFR 50.54(a)(3), and ITS Section 5, "Administrative Controls." This will not impact the actual requirements but may provide some flexibility in how the requirement is conducted. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as indicated below:

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change relocates requirements from the Technical Specifications to other license basis documents which are under licensee control. The documents containing the relocated requirements will be maintained using the provisions of applicable regulatory requirements. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be transposed from the Technical Specifications to other license basis documents, which are under licensee control, are the same as the existing Technical Specifications. The documents containing the relocated requirements will be maintained using the provisions of applicable regulatory requirements. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

"M" - More restrictive changes to requirements:

The ANO-1 Technical Specifications are proposed to be modified in some areas to impose more stringent requirements than previously identified. These more restrictive modifications are being imposed to be consistent with the improved Babcock & Wilcox Standard Technical Specifications. Such changes have been made after ensuring the previously evaluated safety analysis was not affected. Also, other more restrictive technical changes have been made to achieve consistency, correct discrepancies, and remove ambiguities from the specification.

The modification of the ANO-1 Technical Specifications and the changes made to achieve consistency within the specifications have been performed in a manner such that the most stringent requirements are imposed, except in cases which are individually evaluated.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change provides more stringent requirements for the ANO-1 Technical Specifications. These more stringent requirements are not assumed to be initiators of analyzed events and will not alter assumptions relative to mitigation of accident or transient events. The change has been confirmed to ensure no previously evaluated accident has been adversely affected. The more stringent requirements are imposed to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes do not impact the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated for ANO-1.

3. Does this change involve a significant reduction in a margin of safety?

The imposition of more stringent requirements prevents a reduction in the margin of plant safety by:

a) Increasing the analytical or safety limit,
b) Increasing the scope of the specification to include additional plant equipment,
c) Increasing the applicability of the specification,
d) Providing additional actions,
e) Decreasing restoration times,
f) Imposing new surveillances, or
g) Decreasing surveillance intervals.

The change is consistent with the safety analysis and licensing basis. Therefore, this change does not involve a reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

ITS Section 4.0: Design Features

Entergy Operations has evaluated these proposed Technical Specification changes and has determined that they involve no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92(c) as indicated below:

No unit specific "Less Restrictive" changes identified.

ITS DISCUSSION OF DIFFERENCES
ITS Section 4.0: Design Features

1 NUREG 4.2.1 - Minor revisions are incorporated into the Improved Technical Specifications (ITS) description of fuel assemblies pursuant to the Revised Standard Technical Specification (RSTS) 4.2.1. The use of zircaloy is clarified as cladding material by the addition of the term "clad." ZIRLO is omitted since it is not intended to be used as cladding material for this unit. This change is consistent with the requirements of 10 CFR 50.46, which allows the use of either cladding material. The allowance for "limited substitutions of zirconium alloy filler rods for fuel rods" is currently not approved for use in ANO-1 and is omitted in the ITS. These changes are consistent with current license basis.

2 NUREG 4.2.2 - Incorporates TSTF-123, Rev 1.

3 NUREG 4.2.2 - The plant specific "control material" in the CONTROL RODS is silver indium cadmium and Inconel in the APSRs as identified in CTS 5.3.1.4. This change is consistent with current license basis.

4 NUREG 4.3.1.1 - There are only high density spent fuel storage racks provided at ANO-1. Therefore, there is no need to differentiate between high density and low density racks in ITS 4.3.1.1, nor to provide any information on low density storage racks pursuant to NUREG 4.3.1.1.d. This change is consistent with current license basis.

5 NUREG 4.3.1.2 - The CTS 5.4.1.1 plant specific controls which preclude storage in ten of the interior new (fresh) fuel storage rack locations are retained. These controls are necessary to assure the margin to criticality required by ITS 4.3.1.2.c is maintained as discussed in the submittal documents and the Safety Evaluation Report related to Amendment No. 166. This change is consistent with current license basis.

4.0-02 6 Not Used.

7 NUREG 4.3.1.2 - Details of reactivity conditions of the fuel storage racks were revised to reflect requirements from CTS 5.4. The ANO-1 SAR does not provide sufficient detail to support adoption of requirements as presented in NUREG-1430. This change is consistent with current license basis.

4.0-01 8 Not Used.

9 NUREG 4.3.3 – The ANO-1 spent fuel pool does not include storage spaces
4.0-02 specifically designated as failed fuel containers. Therefore, this information is not retained in ITS 4.3.3. This change is consistent with current license basis. ANO-1 specific values are inserted within other bracketed spaces in NUREG 4.3.2 and NUREG 4.3.3.

CTS

4.0 DESIGN FEATURES

4.1 Site Location ~~[Text description of site location.]~~ INSERT 4.0-1A 5.1

4.2 Reactor Core

4.2.1 Fuel Assemblies 5.3.1.1

The reactor shall contain ~~[177]~~ 177 fuel assemblies. Each assembly shall consist of a matrix of ~~[Zircaloy or ZIRLO]~~ clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of ~~zirconium alloy or~~ stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions. 1 edit

4.2.2 ~~CONTROL RODS~~ Control Assemblies 5.3.1.4

The reactor core shall contain ~~[60]~~ 60 safety and regulating and ~~[8]~~ 8 APSR assemblies. ~~axial power shaping CONTROL RODS.~~ The CONTROL ROD assembly CONTROL ROD assemblies control material shall be ~~a~~ silver-indium-cadmium ~~, boron carbide, or hafnium metal~~ alloy and the APSR assembly control material shall be an Inconel alloy, as approved by the NRC. a 2 3

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of ~~[4.5]~~ 4.1 weight percent; 3.8.15

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in [Section ~~9.1~~ 9.6.2.4.3 of the ~~F~~SAR]; 5.4.2.1

4.0-01

(continued)

<INSERT 4.0-1A>

The site for Arkansas Nuclear One Unit 1 is located in Pope County, Arkansas on the bank of the Dardanelle Reservoir (Arkansas River), approximately 6 miles west-northwest of Russellville, AR. The exclusion area boundary shall have a radius of 0.65 statute miles from the Unit 1 reactor building.

CTS

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

NA

~~c.~~ b. A nominal ~~[ ]~~ 10.65 inch center to center distance between fuel assemblies placed in ~~[the high density fuel~~ storage racks~~];~~ ~~]~~

~~[d. A nominal [ ] inch center to center distance between fuel assemblies placed in [the low density fuel storage racks];]~~ 4

~~e.~~ d. New or partially spent fuel assemblies with a discharge burnup in the "acceptable range" of Figure ~~[3.7.17-1]~~ 3.7.15-1 ~~may be~~ allowed unrestricted storage in ~~[either]~~ fuel storage rack~~(s)~~ Region 1 or Region 2; ~~and]~~ 5.4.1.2 3.8.16 edit

ANO-356

~~f.~~ e. New or partially spent fuel assemblies with a discharge burnup in the "unacceptable range" of Figure ~~[3.7.17-1]~~ ~~will be~~ stored in ~~compliance with the NRC approved [specific document containing the analytical methods, title, date, or specific configuration or figure].]~~ Region 1, or in a checkerboard configuration in Region 2. 3.8.16 edit

edit

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of ~~[4.5]~~ 4.1 weight percent; 5.4.1.1

b. $k_{eff} \leq 0.95$ ~~is fully flooded with unborated water,~~ under normal conditions which includes an allowance for uncertainties as described in [Section ~~9.1~~ 9.6.2.4.3 of the ~~F~~SAR]; 5.4.1.1 7

c. $k_{eff} \leq 0.98$ ~~if moderated by aqueous foam,~~ with optimum moderation, which includes an allowance for uncertainties as described in [Section ~~9.1~~ 9.6.2.4.3 of the ~~F~~SAR]; ~~and~~ 5.4.1.1

4.0-01 4.0-01 4.0-01

d. A nominal ~~[21.125]~~ 21 inch center to center distance between fuel assemblies placed in the storage racks; and 5.4.1.1

e. Ten interior storage cells, as shown in Figure 4.3.1.2-1, precluded from use during fuel storage. 5.4.1.1 5

(continued)

CTS

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

4.3.2 Drainage NA

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation [~~138~~ 397 ft ~~4 inches~~].

4.0-02

4.3.3 Capacity NA

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than [~~1357~~ 968] fuel assemblies ~~[and six failed fuel containers]~~. 9

<Insert 4.0-3A> 5

<INSERT 4.0-3A>

<----North

		NO			NO		
			NO	NO			
			NO	NO			
			NO	NO			
		NO			NO		

"NO" indicates a location in which fuel loading is prohibited.

Figure 4.3.1.2-1 (page 1 of 1)
Fresh Fuel Storage Rack Loading Pattern

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The ANO-1 plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

5.1.2 An individual with an active Senior Reactor Operator (SRO) license shall be designated as responsible for the control room command function while the unit is in MODE 1, 2, 3, or 4. With the unit not in MODES 1, 2, 3, or 4, an individual with an active SRO or Reactor Operator license shall be designated as responsible for the control room command function.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power unit.

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the unit specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the Safety Analysis Report (SAR);

b. The ANO-1 plant manager shall be responsible for overall safe operation of the unit and shall have control over those onsite activities necessary for safe operation and maintenance of the unit;

c. A specified corporate executive shall have corporate responsibility for overall unit nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the unit to ensure nuclear safety. The specified corporate executive shall be identified in the SAR; and

d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

5.2.2 Unit Staff

a. A non-licensed operator shall be on site when fuel is in the reactor and an additional non-licensed operator shall be on site when the reactor is in MODES 1, 2, 3, or 4.

b. The minimum shift crew composition for licensed operators shall meet the minimum staffing requirements of 10 CFR 50.54(m)(2)(i) for one unit, one control room.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

c. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) for one unit, one control room, and 5.2.2.a and 5.2.2.g for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

e. The amount of overtime worked by unit staff members performing safety related functions shall be limited and controlled in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12).

f. The operations manager or assistant operations manager shall hold an SRO license.

g. In MODES 1, 2, 3, or 4, an individual shall provide advisory technical support for the operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI ANS 3.1 - 1978 for comparable positions, except for the designated radiation protection manager, who shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, September 1975.

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Section 7.1 of Generic Letter 82-33;

c. Fire Protection Program implementation; and

d. All programs specified in Specification 5.5.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and

The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports.

Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

   2. a determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after the approval of the ANO general manager; and

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made effective. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at least once per 18 months. The provisions of SR 3.0.2 are applicable.

5.5.3 Post Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive iodine, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:

a. Training of personnel;

b. Procedures for sampling and analysis; and

c. Provisions for maintenance of sampling and analysis equipment.

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in Appendix B, Table 2, Column 2, to 10 CFR 20.1001 – 20.2402;

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be in accordance with the following:

   1. For noble gases: a dose rate ≤ 500 mrem/yr to the whole body and a dose rate ≤ 3000 mrem/yr to the skin, and

   2. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrem/yr to any organ;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public beyond the site boundary due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.5 (Not Used).

5.5.6 (Not Used).

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel. Surface and volumetric examination of the reactor coolant pump flywheels will be conducted coincident with refueling or maintenance shutdowns such that during 10 year intervals all four reactor coolant pump flywheels will be examined. Such examinations will be performed to the extent possible through the access ports, i.e., those areas of the flywheel accessible without motor disassembly. The surface and volumetric examination may be accomplished by Acoustic Emission Examination as an initial examination method. Should the results of the Acoustic Emission Examination indicate that additional examination is necessary to ensure the structural integrity of the flywheel, then other appropriate NDE methods will be performed on the area of concern.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Reactor Coolant Pump Flywheel Inspection Program inspection frequencies.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.8 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

ASME Code terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Monthly	At least once per 31 days
Every 6 weeks	At least once per 42 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

5.5.9 Steam Generator (SG) Tube Surveillance Program

This program provides controls to ensure integrity of the steam generator tubing through a defined inservice surveillance program, and to minimize exposure of personnel to radiation during performance of the surveillance program.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the SG Tube Surveillance Program inspection frequencies.

a. The first steam generator tubing inspection performed in accordance with 5.5.9.b and 5.5.9.c.1 shall be considered as constituting the baseline condition for subsequent inspections.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

b. Examination Methods:

1. Inservice inspection of steam generator tubing shall include non-destructive examination by eddy-current testing or other equivalent techniques. The inspection equipment shall provide a sensitivity that will detect defects with a penetration of 20 percent or more of the minimum allowable as-manufactured tube wall thickness except for a sleeved tube at the lower sleeve end.

2. For examination of the sleeved steam generator tubing at the lower sleeve end, the indications will be compared to those obtained during the baseline sleeved tube inspection. Significant deviations between these indications will be considered sufficient evidence to warrant designation as a degraded tube. Direct quantification of the 40 percent through-wall plugging limit is available with eddy-current testing.

c. Selection and Testing:

The steam generator sample size is specified in Table 5.5.9-1. The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.5.9-2. The inservice inspection of steam generator tubes shall be performed at the frequencies as specified in 5.5.9.d and the inspected tubes shall be verified acceptable per the acceptance criteria of 5.5.9.e. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in both steam generators; the tubes selected for these inspections shall be selected on a random basis except:

1. The first sample inspection during each inservice inspection (subsequent to the baseline inspection) of each steam generator shall include:

   i. All nonplugged tubes that previously had detectable wall penetrations (>20%), except tubes in which the wall penetration has been spanned by a sleeve, and

   ii. At least 50% of the tubes inspected shall be in those areas where experience has indicated potential problems, except where specific groups are inspected per 5.5.9.c.1.iii.

   A tube inspection (pursuant to 5.5.9.e.1.ix) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

iii. Tubes in the following groups may be excluded from the first random sample if all tubes in a group in both steam generators are inspected. The inspection may be concentrated on those portions of the tubes where imperfections were previously found. No credit will be taken for these tubes in meeting minimum sample size requirements. Where only a portion of the tube is inspected, the remainder of the tube will be subjected to the random inspection.

(1) Group A-1: Tubes within one, two or three rows of the open inspection lane.

(2) Group A-2: Unplugged tubes with sleeves installed.

(3) Group A-3: Tubes in the wedge-shaped group on either side of the lane region (Group A-1) as defined by Figure 5.5.9-1.

iv. Tubes with axially-oriented tube end cracks (TEC) which have been left inservice for the previous cycle shall be inspected with a rotating coil eddy current technique in the area of the TEC and characterized in accordance with topical report BAW-2346P, Rev.0, during all subsequent SG inspection intervals pursuant to 5.5.9.d. The results of this examination may be excluded from the first random sample. Tubes with axial TECs identified during previous inspections, which meet the criteria to remain in service, will not be included when calculating the inspection category of the OTSG.

v. Implementation of the upper tubesheet ODIGA alternate repair criteria requires a 100% bobbin coil inspection of the non-plugged and non-sleeved tubes, spanning the defined region of the upper tubesheet, during all subsequent SG inspection intervals pursuant to 5.5.9.d. Tubes with ODIGA identified during previous inspections, which meet the criteria to remain in service, will not be included when calculating the inspection category for the OTSG. The defined region begins one inch above the upper tubesheet secondary face and ends at the nearest tube roll transition. ODIGA indications detected by the bobbin coil probe shall be characterized using rotating coil probes in accordance with ANO Engineering Report No. 00-R-1005-01.

2. All tubes which have been repaired using the reroll process will have the new roll area inspected during the inservice inspection.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

3. The second and third sample inspections during each inservice inspection as required by Table 5.5.9-2 may be less than a full tube inspection by concentrating the inspection on those areas of the tube sheet array and on those portions of the tubes where tubes with imperfections were previously found.

4. The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected, are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

NOTES:

(1) In all inspections, previously degraded tubes whose degradations have not been spanned by a sleeve must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

(2) Where special inspections are performed pursuant to 5.5.9.c.1.iii, defective or degraded tubes found as a result of the inspection shall be included in determining the Inspection Results Category for that special inspection but need not be included in determining the Inspection Results Category for the general steam generator inspection.

(3) Where special inspections are performed pursuant to 5.5.9.c.2, defective or degraded tube indications found in the new roll area as a result of the inspection and any indications found above the new roll area, are not included in the determination for the inspection results category of a general steam generator inspection.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

d. The above-required inservice inspections of steam generator tubes shall be performed at the following frequencies:

1. The baseline inspection shall be performed during the first refueling shutdown. Subsequent inservice inspections shall be performed at intervals of not less than 10 nor more than 24 calendar months after the previous inspection. If the results of two consecutive inspections for a given group* of tubes following service under all volatile treatment (AVT) conditions fall into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval for that group may be extended to a maximum of 40 months.

2. If the results of the inservice inspection of a steam generator performed in accordance with Table 5.5.9-2 at 40-month intervals for a given group* of tubes fall in Category C-3, subsequent inservice inspections shall be performed at intervals of not less than 10 nor more than 20 calendar months after the previous inspection. The increase in inspection frequency shall apply until a subsequent inspection meets the conditions specified in 5.5.9.d.1 and the interval can be extended to 40 months.

3. Additional unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 5.5.9-2 during the shutdown subsequent to any of the following conditions:

i. Primary-to-secondary leakage in excess of the limits of Specification 3.4.13 (inservice inspection not required if leaks originate from tube-to-tubesheet welds). If the leaking tube is from either Group A-1 or A-3 as defined in Specification 5.5.9.c.1.iii, all of the tubes in the affected group in this steam generator may be inspected in lieu of the first sample inspection specified in Table 5.5.9-2. If the degradation mechanism which caused the leak is limited to a specific portion of the tube length, the inspection per this paragraph may be limited to the affected portion of the tube length. If the results of this inspection fall into the C-3 category, all of the tubes in the same group in the other steam generator will also be similarly inspected.

*A group of tubes means:
(a) All tubes inspected pursuant to 5.5.9.c.1.iii, or
(b) All tubes in a steam generator less those inspected pursuant to 5.5.9.c.1.iii.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

If the leaking tube has been repaired by the reroll process and is leaking in the new roll area, all of the tubes in the steam generator that have been repaired by the reroll process will have the new roll area inspected. If the results of this inspection fall into the C-3 category, all of the tubes with rerolled areas in the other steam generator will also be similarly inspected. This inspection will be in lieu of the first sample inspection specified in Table 5.5.9-2.

ii. A seismic occurrence greater than the Operating Basis Earthquake,

iii A loss-of-coolant accident requiring actuation of the engineered safeguards, or

iv. A main steam line or feedwater line break.

e. Acceptance Criteria:

1. Terms as used in this program:

i. Tubing or Tube means that portion of the tube or sleeve which forms the primary system to secondary system pressure boundary.

ii. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

iii. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either the inside or outside of a tube.

iv. Degraded Tube means a tube containing imperfections ≥ 20% of the nominal wall thickness caused by degradation, except where all degradation has been spanned by the installation of a sleeve or repaired by a rerolled joint.

The reroll repair process will be used to repair tubes with defects in the upper and lower tubesheet areas as described in topical report, BAW-2303P, Revision 4.

v. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

vi. Defect means an imperfection of such severity that it exceeds the plugging limit except where the imperfection has been spanned by the installation of a sleeve. A tube containing a defect in its pressure boundary is defective.

vii. Plugging Limit means the imperfection depth at or beyond 40% of the nominal tube wall thickness for which the tube shall be sleeved, rerolled, or removed from service because it may become unserviceable prior to the next inspection. This does not apply to ODIGA indications within the defined region of the upper tubesheet. These indications shall be assessed for continued plant operation in accordance with ANO Engineering Report No. 00-R-1005-01, Rev. 1.

Axially-oriented TEC indications in the tube that do not extend beyond the adjacent cladding portion of the tube sheet into the carbon steel portion are not included in this definition. These indications shall be assessed for continued plant operation in accordance with topical report BAW-2346P, Rev. 0.

viii. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 5.5.9.d.3.

ix. Tube Inspection means an inspection of the steam generator tube from the point of entry completely to the point of exit. For tubes that have been repaired by the reroll process within the tubesheets, that portion of the tube outboard of the new roll can be excluded from future periodic inspection requirements because it is no longer part of the pressure boundary once the repair roll is installed.

2. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug, reroll, or sleeve all tubes exceeding the plugging limit and all tubes containing non-TEC through-wall cracks) required by Table 5.5.9-2.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

TABLE 5.5.9-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No
No. of Steam Generators per Unit	Two
First Inservice Inspection	Two
Second & Subsequent Inservice Inspection	One[1]

Table Notation:

[1] The inservice inspection may be limited to one steam generator on alternating schedule encompassing 3N% of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

TABLE 5.5.9-2

STEAM GENERATOR TUBE INSPECTION [2,3]

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.[1]	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug, reroll, or sleeve defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug, reroll, or sleeve Defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug, reroll, or sleeve defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G. plug, reroll, or sleeve defective tubes and inspect 2S tubes in other S.G.	Other S.G. is C-1	None	N/A	N/A
			Other S.G. is C-2	Perform action for C-2 result of second sample	N/A	N/A
			Other S.G. is C-3	Inspect all tubes in each S.G. and plug, reroll, or sleeve defective tubes.	N/A	N/A

NOTES:

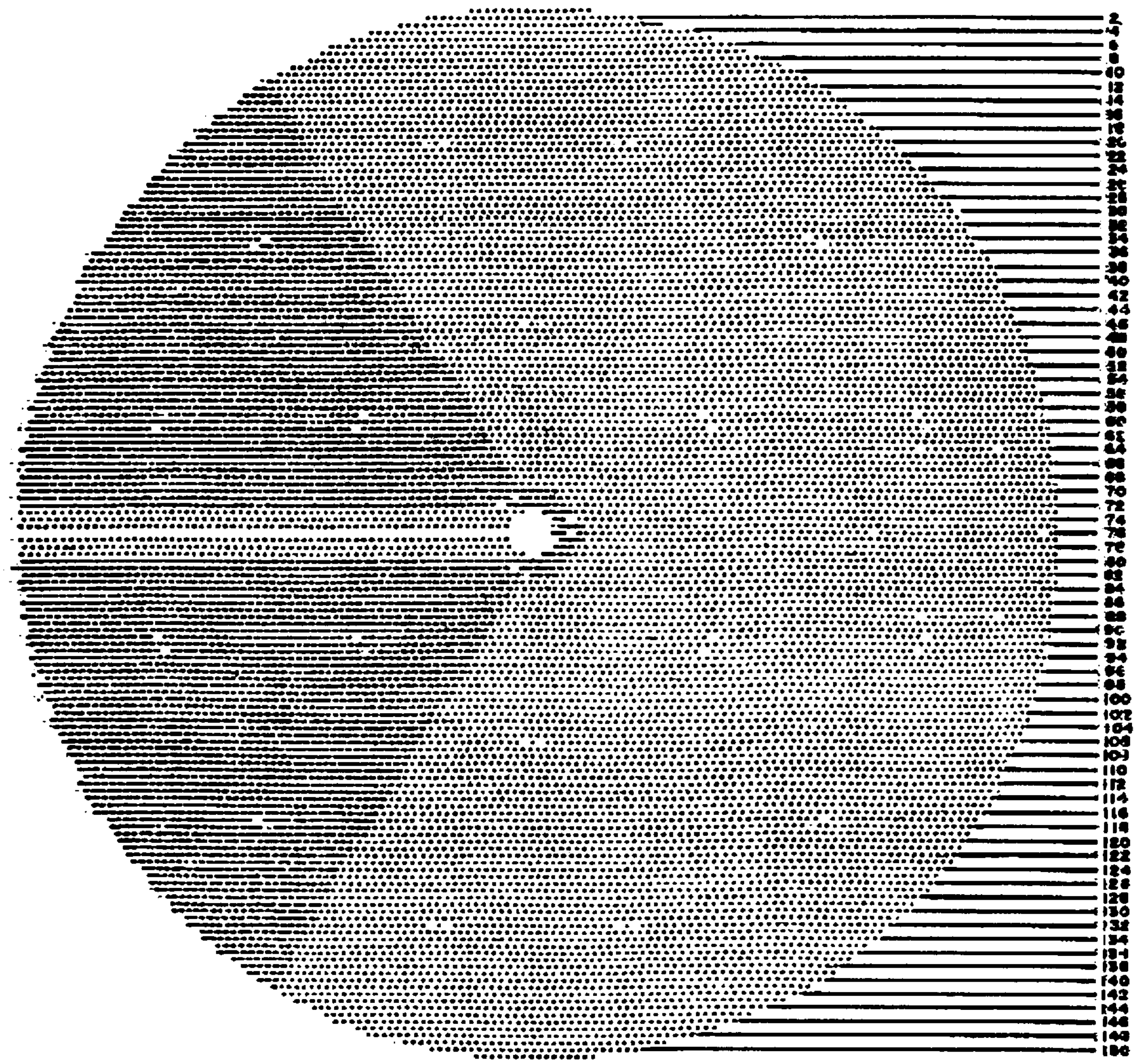
[1] $S = \frac{3N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

[2] For tubes inspected pursuant to 5.5.9.c.1.iii: No action is required for C-1 results. For C-2 results in one or both steam generators plug, reroll, or sleeve defective tubes. For C-3 results in one or both steam generators, plug, reroll, or sleeve defective tubes and provide a report to NRC pursuant to 5.6.7.

[3] No more than ten thousand (10,000) sleeves may be installed in both ANO-1 steam generators combined.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals



DESCRIPTION	TUBE COUNT
Group A-1: Lane region tubes as defined in 5.5.9.c.1.iii(1)	382
Group A-3: Wedge shaped group depicted by darkened region of figure	4880

FIGURE 5.5.9-1 (page 1 of 1)

Upper Tube Sheet View of Wedge Shaped Group (Group A-3) per 5.5.9.c.1.iii

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.10 Secondary Water Chemistry

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events required to initiate corrective action.

5.5.11 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safeguards (ES) ventilation systems filters at the frequencies specified in Regulatory Guide 1.52, Revision 2. The VFTP is applicable to the Penetration Room Ventilation System (PRVS), the Fuel Handling Area Ventilation System (FHAVS), and the Control Room Emergency Ventilation System (CREVS).

a. Demonstrate that an inplace cold DOP test of the high efficiency particulate (HEPA) filters shows:

   1. $\geq$ 99% DOP removal for the PRVS when tested at the system design flowrate of 1800 scfm $\pm$ 10% and the FHAVS when tested at the system design flowrate of 39000 cfm $\pm$ 10%; and

   2. $\geq$ 99.95% DOP removal for the CREVS when tested in accordance with Regulatory Guide 1.52, Revision 2, at the system design flowrate of 2000 cfm $\pm$ 10%.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

b. Demonstrate that an inplace halogenated hydrocarbon test of the charcoal adsorbers shows:

   1. ≥ 99% halogenated hydrocarbon removal for the PRVS when tested at the system design flowrate of 1800 cfm ± 10% and FHAVS when tested at the system design flowrate of 39000 cfm ± 10%; and

   2. ≥ 99.95% halogenated hydrocarbon removal for the CREVS when tested in accordance with Regulatory Guide 1.52, Revision 2, at the system design flowrate of 2000 cfm ± 10%.

c. Demonstrate that a laboratory test of a sample of the charcoal adsorber meets the laboratory testing criteria of ASTM D3803-1989 when tested at 30°C and 95% relative humidity for a methyl iodide penetration of:

   1. < 5% for the PRVS;

   2. < 5% for the FHAVS; and

   3. when obtained as described in Regulatory Guide 1.52, Revision 2, for CREVS

      i. ≤ 2.5% for 2 inch charcoal adsorber beds; and

      ii. ≤ 0.5% for 4 inch charcoal adsorber beds.

d. Demonstrate for the PRVS, FHAVS, and CREVS, that the pressure drop across the combined HEPA filters, other filters in the system, and the charcoal adsorbers is < 6 inches of water when tested at the following system design flowrates ± 10%:

PRVS	1800 cfm
FHAVS	39000 cfm
CREVS	2000 cfm

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas System, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected temporary outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure." The liquid radwaste quantities shall be determined in accordance with the ODCM.

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Waste Gas System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents;

c. A surveillance program to ensure that the quantity of radioactivity contained in all temporary outdoor liquid radwaste tanks: 1) that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents; and 2) that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations equal to the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.13 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

   1. an API gravity or an absolute specific gravity within limits,

   2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

   3. water and sediment within limits;

b. Within 31 days following addition of new fuel oil to storage tanks, verify that the properties of the new fuel oil, other than those addressed in a. above, are within limits for ASTM 2D fuel oil;

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days based on ASTM D-2276, Method A-2 or A-3; and

d. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program surveillance Frequencies.

5.5.14 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

   1. A change in the TS incorporated in the license; or

   2. A change to the updated SAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

   Proposed changes that do not meet these criteria shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the SAR.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, and assuming no concurrent loss of offsite power or loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.16 Reactor Building Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the reactor building as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

In addition, the reactor building purge supply and exhaust isolation valves shall be leakage rate tested once prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days.

The peak calculated reactor building internal pressure for the design basis loss of coolant accident, P_a, is 54 psig.

The maximum allowable reactor building leakage rate, L_a, shall be 0.20% of containment air weight per day at P_a.

Reactor Building leakage rate acceptance criteria is $\leq 1.0L_a$. During the first unit startup following each test performed in accordance with this program, the leakage rate acceptance criteria are $< 0.60L_a$ for the Type B and Type C tests and $< 0.75L_a$ for Type A tests.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Reactor Building Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Reactor Building Leakage Rate Testing Program.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

5.6.1 Occupational Radiation Exposure Report

----------NOTE----------
A single submittal may be made for ANO. The submittal should combine sections common to both units.

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent > 100 mrems and the associated collective deep dose equivalent (reported in person-rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket ionization chamber, thermoluminescence dosimeter (TLD), electronic dosimeter, or film badge measurements. Small exposures totaling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year.

5.6.2 Annual Radiological Environmental Operating Report

----------NOTE----------
A single submittal may be made for ANO. The submittal should combine sections common to both units.

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

--NOTE--
A single submittal may be made for ANO. The submittal shall combine sections common to both units . The submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

2.1.1 Variable Low RCS Pressure – Temperature Protective Limits
3.1.1 SHUTDOWN MARGIN (SDM)
3.1.8 PHYSICS TESTS Exceptions – MODE 1
3.1.9 PHYSICS TEST Exceptions - MODE 2
3.2.1 Regulating Rod Insertion Limits
3.2.2 AXIAL POWER SHAPING RODS (APSR) Insertion Limits
3.2.3 AXIAL POWER IMBALANCE Operating Limits
3.2.4 QUADRANT POWER TILT (QPT)
3.2.5 Power Peaking
3.3.1 Reactor Protection System (RPS) Instrumentation
3.4.1 RCS Pressure, Temperature, and Flow DNB limits
3.4.4 RCS Loops – MODES 1 and 2
3.9.1 Boron Concentration

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following document:

Babcock & Wilcox Topical Report BAW-10179-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses" (the approved revision at the time the reload analyses are performed). The approved revision number shall be identified in the COLR.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

5.6.6 Reactor Building Inspection Report

Any degradation exceeding the acceptance criteria of the containment structure detected during the tests required by the Containment Inspection Program shall undergo an engineering evaluation within 60 days of the completion of the inspection surveillance. The results of the engineering evaluation shall be reported to the NRC within an additional 30 days of the time the evaluation is completed. The report shall include the cause of the condition that does not meet the acceptance criteria, the applicability of the conditions to the other unit, the acceptability of the concrete containment without repair of the item, whether or not repair or replacement is required and, if required, the extent, method, and completion date of necessary repairs, and the extent, nature, and frequency of additional examinations.

5.6.7 Steam Generator Tube Surveillance Reports

a. Following each inservice inspection of steam generator tubes, the complete results of the inspection shall be reported to the NRC. This report, to be submitted within 90 days of inspection completion, shall include:

   1. Number and extent of tubes inspected;

   2. Location and percent of wall-thickness penetration for each indication of an imperfection;

   3. Identification of tubes plugged and tubes sleeved;

   4. Number of tubes repaired by rerolling and number of indications detected in the new roll area of the repaired tubes;

   5. Summary of the condition monitoring and operational assessment results when applying TEC alternate repair criteria; and

   6. Summary of the condition monitoring and the operational assessment results (including growth) when applying the upper tubesheet ODIGA alternate repair criteria.

b. In addition, the Commission shall be notified of the results of steam generator tube inspections which fall into Category C-3 as denoted in Table 5.5.9-2 prior to resumption of plant operation. The written report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP), or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area; or

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

   (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

   (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

   1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designee.

   2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP, or equivalent, while in the area by means of closed circuit television, or personnel qualified in radiation protection procedures responsible for controlling personnel radiation exposure in the area and with the means to communicate with individuals in the area who are covered by such surveillance.

4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

CTS DISCUSSION OF CHANGES

ITS Section 5.0: Administrative Controls

ADMINISTRATIVE

A1 The designated change represents a non-technical, non-intent change to the Arkansas Nuclear One, Unit 1 Current Technical Specifications (CTS) made to make the ANO-1 Improved Technical Specifications (ITS) consistent with the Babcock and Wilcox (B&W) revised Standard Technical Specification (RSTS), NUREG-1430, Revision 1 and 10 CFR Part 20. This change does not alter the requirements of the CTS or RSTS. Examples of this type of change include: wording preference; convention adoption; editorial, numbering and formatting changes; and hierarchy structure.

A2 The ANO-1 CTS Bases will be administratively deleted in their entirety in favor of the NUREG-1430 Bases. The CTS Bases will be reviewed for technical content that will be identified for retention in the ITS Bases.

A3 A statement regarding the Applicability of SR 3.0.2 and SR 3.0.3 is added for clarification that the allowances provided by these general Surveillance Requirements are applicable to the identified program. This is an administrative change since the CTS 4.0.2 and 4.0.3 are currently applicable to the requirements being moved to the program that will be identified in the Administrative Controls (Section 5). This change is applicable for CTS 4.2.6 which is to be incorporated into the Reactor Coolant Pump Flywheel Inspection Program, ITS 5.5.7, and to CTS 4.10, 3.13, and 3.15 which are to be incorporated into the Ventilation Filter Testing Program, ITS 5.5.11. This change is also applicable for CTS 3.24, 3.25.1 and 3.25.2 which are to be incorporated into the Explosive Gas and Storage Tank Radioactivity Monitoring Program, ITS 5.5.12, and to CTS 4.6.1.4.e which is to be incorporated into the Diesel Fuel Oil Testing Program, ITS 5.5.13. Additionally, this change is applicable for CTS 4.0.5 which is to be incorporated into the Inservice Testing Program, ITS 5.5.8.

A4 CTS 4.18.6 and Table 4.18-2 reference to a Special Report are removed from the markup to show the editorial removal of cross references in the ITS. This is considered an administrative change because ITS 5.6.7 will continue to have the additional reporting requirements prescribed in the "special" report. This is considered editorial and no change in requirements are associated with this change. This change is consistent with NUREG-1430.

CTS DISCUSSION OF CHANGES

A5 This information has been removed from the ITS since it duplicates requirements provided in the regulations. Such duplication is unnecessary and results in additional administrative burden to revise the duplicate TS when these regulations are revised. Since removal of the duplication results in no actual change in the requirements, removal of the duplicative information is considered an administrative change. Further, changes to the requirements are controlled by the NRC. This change is consistent with NUREG-1430.

CTS Location	Duplicated Regulation
4.0.5	10 CFR 50.55a(f) and 50.55a(g)
4.2.2	10 CFR 50.55a(g)
4.2.3	10 CFR 50.55a(g)
4.3.1 & 4.3.3	10 CFR 50.55a(g)
4.27.2	10 CFR 50.55a(g)
Table 6.2-1 Note *	10 CFR 50.54(m)(2)(iv)
Table 6.2-1 Add. Req. 1	10 CFR 50.54(m)(2)(iii)
Table 6.2-1 Add. Req. 2	10 CFR 50.54(m)(2)(iii)
Table 6.2-1 Add. Req. 4	10 CFR 50.54(m)(2)(iv)
6.10	10 CFR 20
6.12.1	10 CFR 50.4
6.12.3.4	10 CFR 50.4
6.12.5	10 CFR 50.4

The following CTS sections also detail requirements duplicated in the referenced Regulation. However, since 10 CFR 55.4 infers a requirement for Technical Specifications to reference these specifics, the CTS requirement is editorially revised to reflect the Regulation:

6.2.2	10 CFR 50.54(m)(2)(i)
Table 6.2-1 SOL	10 CFR 50.54(m)(2)(i)
Table 6.2-1 OL	10 CFR 50.54(m)(2)(i)

A6 The CTS 4.2.1 pre-operational requirements have been previously completed. Therefore, this surveillance is no longer required, and it deletion is an administrative change.

A7 NUREG 5.5.8, Inservice Testing Program, includes "every 9 months" and "biennially or every 2 years" as ASME test frequencies, and provides a specific number of days (276 and 731 days respectively) by which to interpret these frequencies. Since these frequencies are already provided in the ASME Code and/or NUREG-1430, and the interpretation is simply an obvious editorial clarification, this change is administrative.

ANO-347

A8 The presentation of the requirements for ventilation filter testing is revised for consistency. All frequencies and methods are replaced by a reference to perform the testing at the frequencies specified in Regulatory Guide 1.52, Rev. 2. Since there is no actual change in the Frequencies, this change is considered to be one of presentation only, and therefore, administrative in nature.

A9 Not used.

ANO-335 A10 Not used.

A11 The "≤0.60 L_a" and "≤0.75 L_a" limits for acceptable reactor building leakage in CTS 6.8.4 have been revised to "<0.60 L_a" and "<0.75 L_a" for consistency with the acceptance criteria provided in 10 CFR 50, Appendix J. Therefore, this change has no impact on application of the regulations and is considered administrative.

ANO-341

A12 CTS markup Insert 110jA shows adoption of a statement that the ITS SR 3.0.2 and ITS SR 3.0.3 allowances are applicable to the ITS SG Tube Inspection Program. This is necessary in the ITS to clearly establish that the Section 3.0 allowance is applicable to the Section 5.0 requirements regarding SG tube inspection. The CTS did not require this statement because the SG tube inspection requirements were located within the Surveillance Requirements Section of the CTS and was clearly subject to the CTS 4.0.2 and CTS 4.0.3 allowances. This change is consistent with NUREG-1430 as modified by TSTF-118 with the addition of ITS SR 3.0.3, consistent with the ANO-1 current licensing basis.

A13 CTS 6.12.2.2 is revised to reflect the correct 10 CFR 20 terminology for the units of occupational exposure. A statement limiting the report scope to those persons monitored was added as a statement of the obvious. Lastly, the pocket dosimeter was revised to refer to a pocket ionization chamber and the electronic dosimeter was specified as an additional means of collecting the exposure data. These changes are considered purely administrative since they result in no relaxation of requirements, result in compliance with 10 CFR 20, more accurately reflect the principal of operation of the pocket dosimeter, and acknowledge industry usage of advanced dosimetry devices. These changes are consistent with 10 CFR 20 and NUREG-1430 as revised by TSTF-152.

A14 CTS 6.12.2.6 is revised to reflect the reporting requirements consistent with 10 CFR 20 and minor editorial changes. These changes are considered purely administrative since they result in no relaxation of requirements and result in compliance with 10 CFR 20. These changes are consistent with 10 CFR 20 and NUREG-1430 as revised by TSTF-152.

ANO-340 A15 Not used.

A16 CTS 4.18.5.b, 2nd paragraph, was added by amendment 203 as a one-time, temporary change -- only applicable through Cycle 16. Since ANO-1 will complete Cycle 16 prior to implementation of ITS, this provision can be deleted. This is an administrative change.

CTS DISCUSSION OF CHANGES

A17 CTS 6.8.5 is updated to reflect the latest changes to 10 CFR Part 20. The changes maintain the same overall level of effluent control while retaining the operational flexibility that currently exists. The Specification continues to provide reasonable assurance that acceptable limits will be maintained and eliminate possible confusion or improper implementation of the revised 10 CFR Part 20 requirements. Additionally, consistent with the intent of performing periodic surveillances, a statement regarding the Applicability of SR 3.0.2 and SR 3.0.3 is added. Since no change to the regulatory requirements is made this change is considered administrative.

ANO-345 A18 This page is not yet approved in its current form. Therefore, this markup is dependent on the expected NRC approval of the September 28, 2000 license amendment request related to revision of the SG tube reroll process.

ANO-354 A19. This page is not yet approved in its current form. Therefore, this markup is dependent on the expected NRC approval of the August 29, 2000 license amendment request related to revision of the SG ODIGA requirements.

CTS DISCUSSION OF CHANGES

TECHNICAL CHANGE -- MORE RESTRICTIVE

M1 CTS 6.3.1 is updated to reflect the latest changes to the QAPM approved by the NRC on November 6, 1998 (TAC No. M97893). Unit staff qualifications are revised to reflect commitments to ANSI ANS 3.1-1978 (in lieu of ANSI N18.1-1971). Additional experience and education requirements are imposed for certain positions due to this change. This change is an additional restriction on unit operation.

M2 Not used.

M3 CTS 6.8.1 is revised to incorporate additional procedure requirements. The reference to Regulatory Guide 1.33, Appendix A, is updated from November 1972, to reference Revision 2 of the guidance, dated February 1978. This updated reference is consistent with the current reference in the ANO-2 CTS, and with the RSTS. An additional item is incorporated to require emergency operating procedures for implementation of the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Section 7.1 of Generic Letter 82-33. This is consistent with the CTS requirements prior to Amendment 179 and with the RSTS. Finally, additional requirements are included to provide procedures for each of the programs identified in proposed ITS 5.5. Of these, only two programs are totally new: the Technical Specification Bases Control Program and the Safety Function Determination Program (see DOC M7 below). The remaining programs are based on requirements in the CTS. This change is also consistent with the RSTS and is an additional restriction on unit operation.

M4 Not used.

M5 CTS 3.13.1.d, CTS 3.15.1.d, CTS 4.10.2.d.1, CTS 4.11.1, and CTS 4.17.1 are revised to include the prefilters and "roughing" filters in the ventilation system differential pressure testing requirements. The revision is shown as "other filters in the system" to accommodate system specific nomenclature and system design variances. These filters are part of the system and obviously do contribute to the system pressure drop and capability of the system to perform its function. Therefore, inclusion of the prefilters in this testing is appropriate. This change is an additional restriction on unit operation.

M6 Not used.

M7 Two new programs are proposed for inclusion in the ITS. These are ITS 5.5.14, "Technical Specification Bases Control Program," and ITS 5.5.15, "Safety Function Determination Program." Both of these programs are necessary for proper implementation of the ITS, and are consistent with NUREG-1430. These new programs are an additional restriction on unit operation.

M8 CTS 4.26.2 requires the reactor building purge supply and exhaust isolation valves to be local leak rate tested in accordance with the 10 CFR 50, Appendix J requirements, but on a Frequency which is more restrictive than the Appendix J frequency. The CTS Frequency is related to reactor building integrity, and the ITS Frequency will also require the testing on a Frequency similarly related to the Applicability for reactor building OPERABILITY. However, the Applicability for reactor building integrity/OPERABILITY has been revised (see ITS Section 3.6) in a manner which is more restrictive than CTS. This change in Applicability is also reflected in this Surveillance Requirement Frequency. This is an additional restriction on unit operation.

M9 CTS 4.6.1.4.e is revised to include testing of new fuel oil. Immediate confirmation of fuel oil quality (by monitoring for specific gravity, viscosity, and appearance/color) as well as follow up confirmatory testing within 30 days after adding new fuel oil to the bulk storage tank will provide added assurance of acceptable fuel oil. This broad spectrum testing will not be routinely performed (refer to DOC L6) since this initial verification provides the necessary confirmation of fuel oil quality. Additionally, this testing is in accordance with NUREG-1430. This is an additional restriction on unit operation.

M10 By deleting specific Regulatory Guide (RG) 1.52 section references from CTS 4.10.2.b.1, the associated ITS section (5.5.11) will ensure all applicable RG 1.52 filter testing frequencies and criteria are applied to the TS ventilation filter systems. This results in a more restrictive change to unit operation, although RG 1.52 testing not specifically detailed in the CTS has previously been incorporated within the ANO filter testing program. RG 1.52 criteria not contained within the CTS includes the air flow distribution test (when maintenance activities may have affected the air flow distribution) for the Control Room Emergency Ventilation System, and the charcoal absorber leak test following charcoal sampling activities (when the effectiveness of the charcoal absorber may have been affected) for all TS ventilation systems. These tests are currently performed, as applicable, under the filter testing program at ANO.

ANO-347

M11 The specific system design flow rate values for the Penetration Room Ventilation System and the Fuel Handling Area Ventilation system are incorporated in the ITS. Incorporation of these values is in accordance with NUREG-1430. This is an additional restriction on unit operation.

ANO-348

CTS DISCUSSION OF CHANGES

TECHNICAL CHANGE -- LESS RESTRICTIVE

L1 Not used.

L2 The CTS 6.11 requirements for high radiation areas are revised to include additional, previously approved methods for implementation of alternates to the "control device" or "alarm signal" requirements of 10 CFR 20. These alternatives provide adequate control of personnel in high radiation areas as evidenced by NRC issuance of NUREG-1430 and approval of generic change TSTF-258, Revision 4.

L3a CTS 6.12.2.2 is revised to require the submittal of the Occupational Exposure Data Report by April 30 of each calendar year. This change is consistent with the comprehensive revisions to 10 CFR 20. The date of submittal for the Annual Occupational Exposure Report is revised from March 1 to April 30. This report is provided to supplement the information required by 10 CFR 20.2206(b) which is filed on or before April 30 in accordance with 10 CFR 20.2206(c). The supplemental information report submittal date is therefore revised to correspond to the required submittal date of the report being supplemented.

L3b The CTS 6.12.2.4 requirements for reporting of all challenges to the pressurizer electromatic relief valves (ERVs) and the pressurizer safety valves is omitted. Reporting of these challenges was incorporated into the CTS in response to TMI Action Item II.K.3.3. This action plan item was originally implemented only to provide a venue for data gathering, and this requirement has been in effect since 1980. There is no plant specific safety basis for submitting routine information on the operation of this particular equipment. Finally, any challenges to these valves that result in a potential impact on safety would be evaluated for reportability under 10 CFR 50.73. See also: NUREG-0565, items 2.1.2.c & 2.1.2.e; NUREG-0611, items 3.2.4.h & 3.2.4.j; NUREG-0626, items F-2.5 & F-3.5; and NUREG-0635, item 3.2.4.d for background information on this report.

L4 CTS 3.7.3.A.1 provides a requirement for a redundant subsystem verification for the purpose of identifying a potential loss of safety function. CTS 3.7.3.B would require a shutdown if a potential loss of safety function were discovered. The ITS does not always require a shutdown if a loss of function is identified. Rather, it requires that both redundant components be declared inoperable and the corresponding ACTIONS of the LCO applicable for those components be entered. These ACTIONS may provide for other compensatory measures that have been determined to be appropriate for the condition. Therefore, this CTS requirement is more appropriately addressed with the added Safety Function Determination Program of ITS 5.5.15. This change is consistent with NUREG-1430.

ANO-346 L5 Not used.

CTS DISCUSSION OF CHANGES

L6 CTS 4.6.1.4.e is revised to require the periodic testing of the stored fuel oil only for particulates (replacing the periodic testing per ASTM-D975) once every 31 days per ITS 5.5.13 (refer to DOC M9 for added testing requirements). This change also relaxes CTS requirement that the sample and testing be in conjunction with the monthly DG run. These changes reflect industry-standard acceptable DG fuel oil testing programs reflected in NUREG-1430. Over the storage life of ANO-1 DG fuel oil, the properties tested by ASTM-D975 are not expected to change and performing these tests once on the new fuel oil (see DOC M9) provides adequate assurance of the proper quality fuel oil. The periodic testing for particulates monitors a parameter that reflects degradation of fuel oil and can be trended to provide increased confidence that the stored DG fuel oil will support DG operability.

CTS DISCUSSION OF CHANGES

LESS RESTRICTIVE -- ADMINISTRATIVE DELETION OF REQUIREMENTS

LA1 Where possible, plant specific management position titles in the CTS are replaced with generic titles as provided in ANSI/ANS 3.1. Personnel who fulfill these positions are still required to meet the qualifications detailed in proposed Specification 5.3. In addition, compliance details relating to the plant specific management position titles fulfilling the duties of these generic positions will continue to be defined, established, documented and updated in a plant controlled document, such as the Quality Assurance Program Manual (QAPM). This approach is consistent with the intent of Generic Letter 88-06 which recommended, as a line item improvement, relocation of the corporate and unit organization charts to licensee controlled documents. The intent of the Generic Letter, and of this proposed change, is to reduce the unnecessary burden on NRC and licensee resources being used to process changes due solely to personnel titles changes during reorganizations. Since this change does not eliminate any of the qualifications, responsibilities or requirements for these personnel or the positions, the change is considered to be a change in presentation only and is therefore administrative. The specific replacements are:

ITS 5.5.1	ANO general manager for General Manager, Plant Operations

LA2 The requirement for a third non-licensed operator is moved to a licensee controlled document such as the Safety Analysis Report (SAR). This information provides details of the method of implementation of other requirements (e.g., fire protection) which are not directly pertinent to the specific shift manning requirements, and which are no longer directly controlled by Technical Specifications (since they did not meet the inclusion criteria of 10 CFR 50.36). Since this detail is not necessary to adequately describe the actual regulatory requirement, it can be moved to a licensee controlled document without a significant impact on safety. Placing this detail in controlled documents provides adequate assurance that it will be maintained. The SAR will be controlled by 10 CFR 50.59 and 50.71(e).

LA3 This information has been moved to a licensee controlled document such as the Ventilation Filter Testing Program (VFTP), Diesel Fuel Oil Testing Program (DFOTP), or Reactor Building Leakage Rate Testing Program (RBLRTP), etc. A description of the Program is incorporated into the Administrative Controls section of ITS. This information provides details of the method of implementation which are not directly pertinent to the actual requirement. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The details relocated to the VFTP, RBTSP, DFOTP, and RBLRTP will be controlled by 10 CFR 50.59. The CTS location and ITS location for each of these items is listed below. This change is consistent with NUREG-1430.

CTS Location	New Location
3.13.1.e	VFTP
ANO-342 3.15.1.e	VFTP
4.6.1.4.e	DFOTP
4.10	VFTP
4.11	VFTP
4.17	VFTP

LA4 The CTS Operating License Condition 2.C(6) requirement for monitoring of iodine in vital areas (except the containment atmosphere which is retained in ITS 5.5.3) is moved to a licensee controlled document such as the Safety Analysis Report (SAR). This information provides details of the method of implementation of other requirements (e.g., radiation protection) which are not directly pertinent to the safe shutdown of the unit, and which are no longer directly controlled by Technical Specifications (since they did not meet the inclusion criteria of 10 CFR 50.36.) Since this detail is not necessary to adequately describe the actual regulatory requirement, it can be moved to a licensee controlled document without a significant impact on safety. Placing this detail in controlled documents provides adequate assurance that it will be maintained. The SAR will be controlled by 10 CFR 50.59 and 50.71(e).

CTS DISCUSSION OF CHANGES

LA5 This information has been moved to a licensee controlled document such as the Explosive Gas and Storage Tank Radioactivity Program (EG&STRMP). A description of the Program is incorporated into the Administrative Controls section of ITS which includes appropriate limits, actions, and surveillance requirements. The information moved to the TRM provides details of the method of implementation which are not directly pertinent to the actual requirement. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The details relocated to the EG&STRMP will be controlled by 10 CFR 50.59. The CTS location and ITS location for each of these items is listed below. This change is consistent with NUREG-1430.

CTS Location	New Location
3.24	EG&STRMP
3.25.1	EG&STRMP
3.25.2	EG&STRMP
4.28 & Figure 3.7.4-1	EG&STRMP
4.29.1	EG&STRMP
4.29.2	EG&STRMP

LA6 This information has been moved to a licensee controlled document such as the Bases, Safety Analysis Report (SAR), QAPM, TRM, etc. This information provides details of the method of implementation that are not directly pertinent to the actual requirement. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The Bases will be controlled by the Process identified in Chapter 5 of the proposed ITS. The details relocated to the SAR and TRM will be controlled by 10 CFR 50.59. The details relocated to the QAPM will be controlled by 10 CFR 50.54(a)(3). The CTS location and ITS location for each of these items is listed below. This change is consistent with NUREG-1430.

	CTS Location	New Location
ANO-346	Table 4.1-2 Item 12	TRM
	4.2.4	SAR (3.2.4)
	4.2.5	QAPM
	6.12.2	TRM

5.5.2
5.5.3

Not addressed by TSIP.

(4) Physical Protection

EOI shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10CFR73.55 (51 FR 27817 and 27822) and to the authority of 10CFR50.90 and 10CFR50.54(p). The plan, which contains Safeguards Information protected under 10CFR73.21, is entitled: "Arkansas Nuclear One Industrial Security Plan," with revisions submitted through August 4, 1995. The Industrial Security Plan also includes the requirements for guard training and qualification in Appendix A and the safeguards contingency events in Chapter 7. Changes made in accordance with 10CFR73.55 shall be implemented in accordance with the schedule set forth therein.

5.5.2

(5) Systems Integrity Not used.

EOI shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

LA4
SAR

5.5.3

(6) Iodine Monitoring Not used.

EOI shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration ~~in vital areas~~ under accident conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

5.5.10

5.5.10 (7) Secondary Water Chemistry Monitoring Not used.

This program provides controls for

A secondary water chemistry monitoring ~~program shall be implemented~~ to ~~minimize~~ inhibit steam generator tube degradation. This program shall include: A1

1. Identification of a sampling schedule for the critical ~~parameters~~ variables and control points for these ~~parameters~~ variables;

2. Identification of the procedures used to measure the values of the critical ~~parameters~~;

3. Identification of process sampling points;

4. Procedures for the recording and management of data;

5. Procedures defining corrective actions for off-control point chemistry conditions; and

6. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events required to initiate a corrective action

Not addressed by TSIP.

(8) FIRE PROTECTION

EOI shall implement and maintian in effect all provisions of the approved Fire Protection Program as described in Amendment 9A to the Safety Analysis Report and as approved in the Safety Evaluation dated March 31, 1992, subject to the following provision:

The licensee may make changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

3. This license is effective as of the date of issuance and shall expire at midnight, May 20, 2014.

FOR THE ATOMIC ENERGY COMMISSION

Original Signed by:
A. Giambusso

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Attachment:
Appendices A and B - Technical Specifications

Date of Issuance: May 21 1974

5.5.15

<LATER>
(3.8)

3.7.3 Both 125 VDC electrical power subsystems shall be operable when the unit is above the cold shutdown condition.

A. With one 125 VDC electrical power subsystem inoperable:

LATER

5.5.15

1. verify that there are no inoperable safety related components associated with the operable 125 VDC electrical subsystem which are redundant to the inoperable 125 VDC electrical power subsystem,

L4

2. verify the operability of the diesel generator associated with the operable 125 VDC electrical subsystem immediately, and

3. restore the 125 VDC electrical subsystem to operable status within 8 hours.

<LATER>
(3.8)

B. With one 125 VDC electrical power subsystem inoperable, and unable to satisfy the requirements or allowable outage times of 3.7.3.A.1, 3.7.3.A.2, or 3.7.3.A.3, the unit shall be placed in hot shutdown within 12 hours and in cold shutdown within an additional 24 hours.

3.7.4 Battery cell parameters shall be within limits when the associated DC electrical power subsystems are required to be operable.

A. With one or more batteries with one or more battery cell parameters not within Table 4.6-1 Category A or B limits:

1. Within 1 hour, verify pilot cell electrolyte level and float voltage meet Table 4.6-1 Category C limits,

2. Within 24 hours and once per 7 days thereafter, verify battery cell parameters meet Table 4.6-1 Category C limits, and

3. Within 31 days, restore battery cell parameters to Table 4.6-1 Category A and B limits.

B. With one or more batteries with one or more battery cell parameters not within Table 4.6-1 Category A or B limits and unable to satisfy the requirements or allowable outage times of 3.7.4.A.1, 3.7.4.A.2, or 3.7.4.A.3, declare the associated battery inoperable immediately and perform the required actions of 3.7.3.A.

C. With one or more batteries with electrolyte temperature of the pilot cell not within the limits of Specification 4.6.2.8, electrolyte temperature of representative cells not within the limits of Specification 4.6.2.6 or with one or more batteries with one or more battery cell parameters not within Table 4.6-1 Category C limits, declare the associated battery inoperable immediately and perform the required actions of 3.7.3.A.

Bases

The electrical system is designed to be electrically self-sufficient and provide adequate, reliable power sources for all electrical equipment during startup, normal operation, safe shutdown and handling of all emergency situations. To prevent the concurrent loss of all auxiliary power, the various sources of power are independent of and isolated from each other.

LATER

5.5.8

SURVEILLANCE REQUIREMENTS (Continued)

4.0.5 (Continued)

5.5.8

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice ~~inspection and~~ testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications: A5

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice ~~inspection and~~ testing activities	Required frequencies for performing inservice ~~inspection and~~ testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually <ADD: Every 9 MO>	At least once per 366 days
<ADD: "Biennially">	

A5 A7

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice ~~inspection and~~ test activities. A5

~~d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements~~ A1

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requiremetns of any Technical Specification. <INSERT SR 3.0.3 appl.> A3

4.1 OPERATIONAL SAFETY ITEMS

<LATER> (3.3A) (3.3B) (3.3C) (3.3D)

~~Applicability~~

~~Applies to items directly related to safety limits and limiting conditions for operation.~~

~~Objective~~

~~To specify the minimum frequency and type of surveillance to be applied to unit equipment and conditions.~~

~~Specification~~

~~a. The minimum frequency and type of surveillance required for reactor protective system and engineered safeguards system instrumentation when the reactor is critical shall be as stated in Table 4.1-1.~~

LATER

OPERATIONAL SAFETY ITEMS (continued)

4.1 (Continued)

<LATER> (3.3A) (3.3B) (3.3C) (3.3D)

b. ~~Equipment and sampling test shall be performed as detailed in Table 4.1-2 and 4.1-3.~~

c. ~~Discrepancies noted during surveillance testing will be corrected and recorded.~~

LATER

<LATER> (3.2)

d. ~~A power distribution map shall be made to verify the expected power distribution at periodic intervals at least every 10 effective full power days using the incore instrumentation detector system.~~

LATER

BASES

A2

~~4.0.1 through 4.0.5 Establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10CFR 50.36(c)(3):~~

~~"Surveillance Requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."~~

~~4.0.1 Establishes the requirement that surveillances must be performed during the operational modes or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a mode or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an operational mode for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified.~~

5.5.11

<ADD PROGRAM DESCRIPTION>

<LATER> (3.7)

3.13 PENETRATION ROOM VENTILATION SYSTEM

LATER

Applicability

Applies to the operability of the penetration room ventilation system.

Objective

To ensure that the penetration room ventilation system will perform within acceptable levels of efficiency and reliability.

Specification

3.13.1 Two independent circuits of the penetration room ventilation system shall be operable whenever reactor building integrity is required with the following performance capabilities:

ANO-348

5.5.11.a.1 5.5.11.b.1 a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flow, ~~at ± 10%~~ of 1800 cfm on HEPA filters and charcoal adsorber banks shall show ≥ 99% DOP removal and ≥ 99% halogenated hydrocarbon removal. M11

5.5.11.c.1 b. The results of laboratory carbon sample analysis from the charcoal adsorber banks shall show the methyl iodide penetration less than 5.0% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity of 95%.

5.5.11.a.1 5.5.11.b-1 5.5.11.d c. Fans shall be shown to operate within ± 10% of design flow.

ANO-348

5.5.11.d d. The pressure drop across the combined HEPA filters, other filters in the system, and charcoal adsorber banks shall be less than 6 inches of water at system design flow rate, ~~at ± 10%~~ of 1100 cfm. M5 M11

e. Air distribution shall be uniform within ± 20% across HEPA filters and charcoal adsorbers when tested initially and after any maintenance or testing that could affect the air distribution within the penetration room ventilation system. LA3 VFTP

f. Each circuit of the system shall be capable of automatic initiation.

<LATER> (3.7)

3.13.2 If one circuit of the penetration room ventilation system is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days provided that during such seven days all active components of the other circuit shall be operable.

3.13.3 If the requirements of Specifications 3.13.1 and 3.13.2 cannot be met, the reactor shall be placed in the cold shutdown condition within 36 hours.

LATER

<ADD SR 3.0.2 & SR 3.0.3 applicability statement> A13

ANO-335

5.0

AZ

Bases

The penetration room ventilation system is designed to collect and process potential reactor building penetration leakage to minimize environmental activity levels resulting from post accident reactor building leaks. The system consists of sealed penetration rooms, two redundant filter trains and two redundant fans discharging to the unit vent. The entire system is activated by a reactor building engineered safety features signal and initially requires no operator action. Each filter train is constructed with a prefilter, a HEPA filter and a charcoal adsorber in series. The design flow rate through each of these filters is 2000 scfm, which is significantly higher than the 1.25 scfm maximum leakage rate from the reactor building at a leak rate of 0.1% per day.

High efficiency particulate air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should ensure a radioactive methyl iodide removal efficiency of a least 90 percent for expected accident conditions. Acceptable removal efficiency is shown by a methyl iodide penetration of less than 5.0% when tests are performed in accordance with ASTM D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon," at a temperature of 30°C and a relative humidity of 95%. The penetration acceptance criterion is determined by the following equation:

$$\text{Allowable Penetration} = \frac{[100\% - \text{methyl iodide efficiency for charcoal credited in accident analysis}]}{\text{safety factor of 2}}$$

Applying a safety factor of 2 is acceptable because ASTM D3803-1989 is a more accurate and demanding test than older tests.

If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10CFR100 guidelines for the accidents analyzed. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

If one circuit of the penetration room ventilation system is found to be inoperable, there is not an immediate threat to the containment system performance and reactor operation may continue for a limited period of time while repairs are being made.

ANO-335

5.5.11

(LATER) (3.7)

3.15 FUEL HANDLING AREA VENTILATION SYSTEM

LATER

Applicability

Applies to the operability of the fuel handling area ventilation system.

Objective

To ensure that the fuel handling area ventilation system will perform within acceptable levels of efficiency and reliability.

Specification

ANO-348

3.15.1 The fuel handling area ventilation system shall be in operation whenever irradiated fuel handling operations are in progress in the fuel handling area of the auxiliary building and shall have the following performance capabilities:

5.5.11.a.1 5.5.11.b.1 a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flow rate of 29000 cfm ~~±10%~~ on HEPA filters and charcoal adsorber banks shall show ≥ 99% DOP removal and ≥ 99% halogenated hydrocarbon removal. a system M11

5.5.11.c.2 b. The results of laboratory carbon sample analysis shall show the methyl iodide penetration less than 5.0% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity of 95%.

ANO-348

5.5.11.a.1 5.5.11.b.1 5.5.11.d — c. Fans shall be shown to operate within ±10% design flow.

5.5.11.d d. The pressure drop across ~~the combined~~ HEPA filters, other filters in the system, and charcoal adsorber banks shall be less than 6 inches of water at system design flow rate of 29000 cfm ~~±10%~~. M5 M11

e. ~~Air distribution shall be uniform within ±20% across HEPA filters and charcoal adsorbers when tested initially and after any maintenance or testing that could affect the air distribution within the fuel handling area ventilation system.~~ LA3

(LATER) (3.7)

~~3.15.2 If the requirements of Specification 3.15.1 cannot be met, irradiated fuel movement shall not be started (any irradiated fuel assembly movement in progress may be completed). The provisions of Specification 3.0.3 are not applicable.~~ LATER

~~**Bases**~~

~~The fuel handling area ventilation system is designed to filter the auxiliary building atmosphere during fuel handling operations to limit the release of activity should a fuel handling accident occur. The system consists of one circuit containing two exhaust fans and a filter train. The fans are redundant and only one is required to be operating. The filter train consists of a prefilter, a HEPA filter and a charcoal adsorber in series.~~ A2

ANO-335

5.0

High efficiency particulate air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine absorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should ensure a radioactive methyl iodide removal efficiency of at least 90 percent for expected accident conditions. Acceptable removal efficiency is shown by a methyl iodide penetration of less than 5.0% when tests are performed in accordance with ASTM D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon," at a temperature of 30°C and a relative humidity of 95%. The penetration acceptance criterion is determined by the following equation:

$$\text{Allowable Penetration} = \frac{[100\% - \text{methyl iodide efficiency for charcoal credited in accident analysis}}{\text{safety factor of 2}}$$

Applying a safety factor of 2 is acceptable because ASTM D3803-1989 is a more accurate and demanding test than older tests.

If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10CFR100 guidelines for the accidents analyzed. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

A2

ANO-335

<Add Program description>

5.5.12 3.24 EXPLOSIVE GAS MIXTURE

LA5 EG&STRMP

Applicability

Applies to the Waste Gas System hydrogen/oxygen analyzers.

Objective

To prevent accumulation of explosive mixture in the waste gas system.

Specification

3.24.1 The Concentration of hydrogen/oxygen shall be limited in the waste gas decay tanks to Region "A" of Figure 3.24-1.

3.24.2 When the hydrogen/oxygen concentration in any of the decay tanks enters Region "B" of Figure 3.24-1, corrective action shall be taken to return the concentration values to Region "A" within 24 hours.

3.24.3 The provisions of Specification 3.0.3 are not applicable.

A2

Bases

These hydrogen/oxygen limits provide reasonable assurance that no hydrogen/oxygen explosion could occur to allow rupture of the waste gas decay tanks. The hydrogen and oxygen limits are based on information in NUREG/CR-2726 "Light Water Reactor Hydrogen Manual".

<Add SR 3.0.2 & SR 3.0.3 applicability statement> A3

3.5.8

BASES (continued)

<LATER> (3.0)

includes consideration for plant conditions, adequate planning, availability of personnel, the time required to perform the surveillance, and the safety significance of the delay in completing the required surveillance. This provision also provides a time limit for the completion of Surveillance Requirements that become applicable as a consequence of mode changes imposed by Action requirements and for completing Surveillance Requirements that are applicable when an exception to the requirements of Specification 4.0.4 is allowed. If a surveillance is not completed within the 24-hour allowance, the time limits of the Action requirements are applicable at that time. When a surveillance is performed within the 24-hour allowance and the Surveillance Requirements are not met, the time limits of the Action requirements are applicable at the time that the surveillance is terminated. If the Action requirements are greater than 24 hours, sufficient time exists to complete the surveillance.

LATER

Surveillance Requirements do not have to be performed on inoperable equipment because the Action requirements define the remedial measures that apply. However, the Surveillance Requirements have to be met to demonstrate that inoperable equipment has been restored to OPERABLE status.

4.0.4 Establishes the requirement that all applicable surveillances must be met before entry into an operational mode or other condition of operation specified in the Specification. The purpose of this Specification is to ensure that system and component OPERABILITY requirements or parameter limits are met before entry into a mode or condition for which these systems and components ensure safe operation of the facility. This provision applies to changes in operational modes or other specified conditions associated with plant shutdown as well as startup.

Under the provisions of this specification, the applicable Surveillance Requirements must be performed within the specified surveillance interval to ensure that the Limiting Conditions for Operation are met during initial plant startup or following a plant outage.

When a shutdown is required to comply with Action requirements, the provision of Specification 4.0.4 do not apply because this would delay placing the facility in a lower mode of operation.

A2

4.0.5 Establishes the requirment that inservice inspection of ASME Code Class 1,2, and 3 components and inservice testing of ASME Code Class 1,2, and 3 pumps and valves shall be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. These requirements apply except when relief has been provided in writing by the Commission.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

5.5.8

BASES (continued)

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. The requirements of Specification 4.0.4 to perform surveillance activities before entry into an operational mode or other specified condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps and valves to be tested up to one week after return to normal operation. The Technical Specification definition of OPERABLE does not allow a grace period before a component, that is not capable of performing its specified function, is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

A2

4.1 Bases

Check

Failures such as blown instrument fuses, defective indicators, faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator Action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear plant systems, when the plant is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation.

A2 & R TRM

Calibration

Calibration shall be performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels shall be calibrated at least twice weekly (during steady state operating conditions) against a heat balance standard to compensate for instrumentation drift. During nonsteady state operation, the nuclear flux channels shall be calibrated daily to compensate for instrumentation drift and changing rod patterns and core physics parameters.

A2

5.5.12

<ADD Program Description>

3.25 RADIOACTIVE EFFLUENTS

3.25.1 Radioactive Liquid Holdup Tanks

Applicability: At all times.

Objective: To ensure that the limits of 10 CFR 20 are not exceeded.

LA5

EG+STMP

Specifications:

3.25.1 A. The quantity of radioactive material contained in each unprotected* outside temporary radioactive liquid storage tank shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

B. With the quantity of radioactive material exceeding the above limit, immediately suspend all additions of radioactive material to the affected tank and within 48 hours reduce the tank contents to within the limit and describe the events leading to the condition in the next Radioactive Effluent Release Report pursuant to Specification 6.12.2.6.

C. The provisions of Specification 3.0.3 are not applicable.

Bases:

This specification is provided to ensure that in the event of an uncontrolled release of the contents of the tank* the resulting concentrations would be less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in the unrestricted area.

*Tanks included in this specification are those outdoor temporary tanks that 1) are not surrounded by liners, dikes, or walls capable of holding the tank contents, and 2) do not have overflows and surrounding area drains connected to the liquid radwaste treatment system.

A2

<ADD: SR 3.0.2 & SR 3.0.3 applicability statement> A3

5.5.12

<ADD Program Description>

3.25.2 **Radioactive Gas Storage Tanks**

Applicability: At all times

Objective: To restrict the amount of activity in a radioactive gas holdup tank.

LA5

EG; STRM1

Specifications:

3.25.2 A. The quantity of radioactivity contained in each gas storage tank shall be limited to 78,782 curies noble gases (Xe-133 equivalent).

B. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit and describe the events leading to the condition in the next Radioactive Effluent Release Report pursuant to Specification 6.12.2.6.

C. The provisions of Specification 3.0.3 are not applicable.

Bases:

Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that, in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to a member of the public at the nearest exclusion area boundary will not exceed 0.5 rem. This is consistent with Branch Technical Position ETSB 11-5 in NUREG-0800, July 1981.

A2

<ADD. SR 3.0.2 & 3.0.3 applicability statement> A3

AHD-339

5.5.8

SURVEILLANCE REQUIREMENTS

LATER) (3.0)

4.0.1 Surveillance Requirements shall be met during the operational modes or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the surveillance interval. — LATER

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval defined by Specification 4.0.2, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the Action requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The time at which the Action is taken may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the Action requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an operational mode or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation has been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to operational modes as required to comply with Action requirements.

5.5.8

4.0.5 Surveillance Requirements for inservice ~~inspection and~~ testing of ASME Code Class 1,2, and 3 components shall be applicable as follows: (A5)

a. ~~Inservice inspection of ASME Code Class 1,2, and 3 components and inservice testing of ASME Code Class 1,2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i),~~ (A5)

5.0

Table 4.1-2 (Cont.)

Minimum Equipment Test Frequency

	Item	Test	Frequency	
<LATER> (3.4B)	11. Decay heat removal system isolation valve automatic closure and isolation system	Functioning	Each Refueling Shutdown	LATER
ANO-346	12. Flow limiting annulus on main feedwater line at reactor building penetration	Verify, at normal operating conditions, that a gap of at least 0.025 inches exists between the pipe and the annulus.	One year, two years, three years, and every five years thereafter measured from date of initial test.	LA6
<LATER> (3.7)	13. Main steam isolation valves	a. Exercise through approximately 10% travel b. Cycle	a. Quarterly b. Every 18 months	LATER
	14. Main feedwater isolation valves	a. Exercise through approximately 5% travel b. Cycle	a. Quarterly b. Every 18 months	
<LATER> (3.4A)	15. Reactor internals vent valves	Demonstrate operability by: a. Conducting a remote visual inspection of visually accessible surfaces of the valve body and disc sealing faces and evaluating any observed surface irregularities. b. Verifying that the valve is not stuck in an open position, and c. Verifying through manual actuation that the valve is fully open with a force of ≤ 400 lbs (applied vertically upward).	Each refueling shutdown	LATER

5.0

4.2 REACTOR COOLANT SYSTEM SURVEILLANCE

Applicability

Applies to the surveillance of the reactor coolant system pressure boundary.

Objective

To assure the continued integrity of the reactor coolant system pressure boundary.

Specification

A1

4.2.1 Prior to initial unit operation, an ultrasonic test survey shall be made of reactor coolant system pressure boundary welds as required to establish preoperational integrity and baseline data for future inspections.

A6

4.2.2 Post-operational inspections of components shall be made in accordance with the methods and intervals indicated in Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC.

A5

5.0

A5

4.2.3 ~~The structural integrity of the reactor coolant system boundary shall be maintained at the level required by the original acceptance standards throughout the life of the station. Any evidence, as a result of the tests outlined in Table IS-261 of Section XI of the code, that defects have developed or grown, shall be investigated.~~

4.2.4 ~~To assure the structural integrity of the reactor internals throughout the life of the unit, the two sets of main internals bolts (connecting the core barrel to the core support, shield and to the lower grid cylinder) shall remain in place and under tension. This will be verified by visual inspection to determine that the welded bolt locking caps remain in place. All locking caps will be inspected after hot functional testing and whenever the internals are removed from the vessel during a refueling or maintenance shutdown. The core barrel to core support shield caps will be inspected each refueling shutdown.~~ LA6 SAR

4.2.5 ~~Sufficient records of each inspection shall be kept to allow comparison and evaluation of future inspections.~~ LA6 QAPM

5.5.7

4.2.6 Surface and volumetric examination of the reactor coolant pump flywheels will be conducted coincident with refueling or maintenance shutdowns such that during 10 year intervals all four reactor coolant pump flywheels will be examined. Such examinations will be performed to the extent possible through the access ports, i.e., those areas of the flywheel accessible without motor disassembly. The surface and volumetric examination may be accomplished by Acoustic Emission Examination as an initial examination method. Should the results of the Acoustic Emission Examination indicate that additional examination is necessary to ensure the structural integrity of the flywheel, then other appropriate NDE methods will be performed on the area of concern.

<INSERT 77> A3

A2

~~**Bases**~~

~~The surveillance program has been developed to comply with the applicable edition of Section XI and addenda of the ASME Boiler and Pressure Vessel Code, Inservice Inspection of Nuclear Reactor Coolant Systems, as required by 10CFR50.55a, to the extent practicable within limitations of design, geometry and materials of construction.~~

<CTS INSERT 77>

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Reactor Coolant Pump Flywheel Inspection Program inspection frequencies.

5.0

4.3 TESTING FOLLOWING OPENING OF SYSTEM

A5

Applicability

Applies to test requirements for Reactor Coolant System integrity.

Objective

To assure Reactor Coolant System integrity prior to return to criticality following normal opening, modification, or repair.

Specification

4.3.1 When Reactor Coolant System repairs or modifications have been made, these repairs or modifications shall be inspected and tested to meet all applicable code requirements prior to the reactor being made critical.

<LATER> (2.0)

4.3.2 Following any opening of the Reactor Coolant System, it shall be leak tested at not less than 2155 psig, prior to the reactor being made critical, in accordance with the ASME Boiler and Pressure Vessel Codes, Section XI; IWA-5000.

LATER

4.3.3 The limitations of Specification 3.1.2 shall apply.

A5

Bases

A2

Repairs or modifications made to the Reactor Coolant System are inspectable and testable under applicable codes, such as B 31.7, and ASME Boiler and Pressure Vessel Code, Section XI.

For normal opening, the integrity of the Reactor Coolant System in terms of strength, is unchanged. The ASME Boiler and Pressure Vessel Code, Section XI; IWA-5000 requires a system leak test at nominal operating pressure (2155 psig) following system opening. At the end of refueling outages, this test also satisfies the requirements of IWB-2500, Table IWB-2500-1; Category B-P items B15.10, B15.20, B15.30, B15.40, B15.50, B15.60, and B15.70 for all Class I pressure retaining components.

REFERENCES

(1) FSAR, Section 4
(2) ASME Boiler and Pressure Vessel Code, Section XI

5.5.13

4.6 AUXILIARY ELECTRICAL SYSTEM TESTS

Applicability

Applies to the periodic testing and surveillance requirements of the auxiliary electrical system to ensure it will respond promptly and properly when required.

Specification

4.6.1 Diesel Generators

(LATER) (3.8)

1. Each diesel generator shall be manually started each month and demonstrated to be ready for loading within 15 seconds. The signal initiating the start of the diesel shall be varied from one test to another (start with handswitch at control room panel and at diesel local control panel) to verify all starting circuits are operable. The generator shall be synchronized from the control room and loaded to full rated load and allowed to run until diesel generator operating temperatures have stabilized. LATER

2. A test shall be conducted once every 18 months to demonstrate the ability of the diesel generators to perform as designed by:

   a. simulating a loss of off-site power,

   b. simulating of loss of off-site power in conjunction with an ESF signal,

   c. simulating interruption of off-site power and subsequent reconnection of the on-site power source to their respective busses, and

   d. operating the diesel generator for ≥1 hour after operating temperatures have stabilized.

3. Each diesel generator shall be given an inspection once every 18 months following the manufacturer's recommendations for this class of standby service. (A one-time extension of this interval is allowed so that these may be performed during the 1R9 refueling outage, and completed no later than December 1, 1990.)

AND-343

5.5.13 & LATER (3.8)

4. During the monthly diesel generator test specified in paragraph 1 above, the following shall be performed: L6 & LATER

(LATER) (3.8)

   a. The diesel generator starting air compressors shall be checked for operation and their ability to recharge the air receivers.

   b. The diesel oil transfer pumps shall be checked for operability and their ability to transfer oil to the day tank. LATER

   c. The day tank fuel level shall be verified.

   d. The emergency storage tank fuel level shall be verified.

5.5.13 <ADD: Diesel Fuel Oil Testing Program description> LA3

<ADD: SR 3.0.2 & SR 3.0.3 Applicability statement> A3

<ADD: New fuel oil testing> M9

e. Diesel fuel from the emergency storage tank shall be sampled and found to be within acceptable limits ~~specified in Table 1 of ASTM D975-68 when checked for viscosity, water, and sediment.~~ L6

~~5. Once every 31 days the pressure in the required starting air receiver tanks shall be verified to be ≥ 175 psig.~~

~~Once every 18 months, the capacity of each diesel oil transfer pump shall be verified to be at least 10 gpm.~~

<LATER> (3.8)

~~4.6.2 DC Sources and Battery Cell Parameters~~

~~1. Verify battery terminal voltage is ≥ 124.7 V on float charge once each 7 days.~~ LATER

~~2. Verify battery capacity is adequate to supply, and maintain in operable status, the required emergency loads for the design duty cycle when subjected to either a battery service test or a modified performance discharge test once every 18 months.~~

~~3. Verify battery capacity is ≥ 80% of the manufacturers rating when subjected to a performance discharge test or a modified performance discharge test once every 60 months, once every 24 months when battery has reached 85% of the service life with capacity ≥ 100% of the manufacturers rating and showing no degradation, and once every 12 months when battery shows degradation or has reached 85% of the service life and capacity is < 100% of the manufacturer's rating.~~

~~4. Any battery charger which has not been loaded while connected to its 125V d-c distribution system for at least 30 minutes during every quarter shall be tested and loaded while connected to its bus for 30 minutes.~~

~~5. Verify battery pilot cell parameters meet Table 4.6-1 Category A limits once per 7 days.~~

~~6. Verify average electrolyte temperature of representative cells is ≥ 60°F once per 92 days.~~

~~7. Verify battery cell parameters meet Table 4.6-1 Category B limits once per 92 days and once within 24 hours after a battery discharge to < 110 V and once within 24 hours after a battery overcharge to > 145 V.~~

~~8. Verify electrolyte temperature of pilot cell is ≥ 60°F once per 31 days.~~

~~4.6.3 Emergency Lighting~~

~~The correct functioning of the emergency lighting system shall be verified once every 18 months.~~ R TRM

5.5.11

4.10 CONTROL ROOM EMERGENCY VENTILATION AND AIR CONDITIONING SYSTEM SURVEILLANCE

LATER

Applicability

Applies to the surveillance of the control room emergency ventilation and air conditioning systems.

Objective

To verify an acceptable level of efficiency and operability of the control room emergency ventilation and air conditioning systems.

Specification

〈LATER〉 (3.7)

4.10.1 Each train of control room emergency air conditioning shall be demonstrated Operable:

a. At least once per 31 days on a staggered test basis by:

1. Starting each unit and

2. Verifying that each unit operates for at least 1 hour and maintains the control room air temperature ≤84°F D.B.

b. At least once per 18 months by verifying a system flow rate of 9900 cfm ±10%.

4.10.2 Each Control Room Emergency Ventilation System shall be demonstrated Operable:

a. At least once per 31 days on a Staggered Test Basis by initiating, from the Control Room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

5.5.11 〈LATER〉 (3.7)

b. At least once per 18 months or 1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or 2) following significant painting, fire, or chemical release in any ventilation zone communicating with the system by: A8 LATER

ANO-347

5.5.11.a.2
5.5.11.b.2

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 2000 cfm ±10%. M10

ANO-347

5.5.11.c.3

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, ~~meets the~~ laboratory testing criteria of ASTM D3803-1989 when tested at 30°C and 95% relative humidity for a methyl iodide penetration of: LA3 VFTP A8

a. ≤2.5% for 2 inch charcoal adsorber beds, or

b. ≤0.5% for 4 inch charcoal adsorber beds.

5.5.11.a.2
5.5.11.b.2
5.5.11.d

3. Verifying a system flow rate of 2000 cfm ±10% during system operation when tested in accordance with ANSI N510-1975.

5.5.11

ANO-347

5.5.11
5.5.11.c.3

$ <LATER> (3.7)

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989 when tested at 30°C and 95% relative humidity for a methyl iodide penetration of: A8 LA3

1. ≤ 2.5% for 2 inch charcoal adsorber beds, or

2. ≤ 0.5% for 4 inch charcoal adsorber beds.

5.5.11
5.5.11.d

d. At least once per 18 months by: A8

1. Verifying that the pressure ~~drop across the combined HEPA~~ filters and charcoal adsorber banks is < 6 inches of water while operating at a flowrate of 2000 cfm ±10%. other filters in the system N5

<LATER> (3.7)

2. Verifying that on a Control Room high radiation test signal, the system automatically isolates the Control Room within 10 seconds and switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks. LATER

5.5.11
5.5.11.a.2

$ <LATER> (3.7)

e. After each complete or partial replacement of the HEPA filter bank by verifying that the HEPA filter banks remove ≥99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 2000 cfm ±10%. A8

5.5.11
5.5.11.b.2

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove ≥99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 2000 cfm ±10%. A8

Bases

The purpose of the control room emergency ventilation system is to limit the particulate and gaseous fission products to which the control area would be subjected during an accidental radioactive release in or near the Auxiliary Building. The system is designed with 100 percent capacity filter trains which consist of a prefilter, high efficiency particulate filters, charcoal adsorbers and a fan.

Since the emergency ventilation system is not normally operated, a periodic test is required to insure operability when needed. During this test the system will be inspected for such things as water, oil, or other foreign material; gasket deterioration, adhesive deterioration in the HEPA units; and unusual or excessive noise or vibration when the fan motor is running. Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop should be determined at least once per operating cycle to show system performance capability.

A2

<ADD: SR 3.0.2 & SR 3.0.3 applicability statement> A3

5.5.11

A.2

Bases (Continued)

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. The charcoal adsorber efficiency test procedures should allow for obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. Tests of the charcoal adsorbers with DOP aerosol shall be performed in accordance with ANSI N510 (1975) "Standard for Testing of Nuclear Air Cleaning Systems." Any HEPA filters found defective shall be replaced with filters qualified according to Regulatory Position C.3.d of Regulatory Guide 1.52. If laboratory test results are unacceptable, all charcoal adsorbents in the system shall be replaced with charcoal adsorbent qualified according to Regulatory Guide 1.52.

The operability of the control room emergency air conditioning Systems ensure that the ambient air temperature does not exceed the allowable temperature for the equipment and instrumentation cooled by this system and the Control Room will remain habitable for Operations personnel during and following all credible accident conditions.

Operation of the systems for 15 minutes every month will demonstrate operability of the emergency ventilation and emergency air conditioning systems. All dampers and other mechanical and isolation systems will be shown to be operable.

If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significant shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

5.5.11

4.11 PENETRATION ROOM VENTILATION SYSTEM SURVEILLANCE

<LATER> (3.7)

Applicability

Applies to the surveillance of the penetration room ventilation system. — LATER

Objective

To verify an acceptable level of efficiency and operability of the penetration room ventilation system.

Specification

M11 LA3 VFTP & LATER

ANo-348

5.5.11.d &(LATER) (3.7)

4.11.1 ~~At intervals not to exceed 18 months,~~ the pressure drop across the combined HEPA filters, and charcoal adsorber banks shall be demonstrated to be less than 6 inches of water at system design flow rate of 1800 cfm ~~(± 10%)~~. , other filters in the system, — M5

5.5.11 &(LATER) (3.7)

~~4.11.2 Initially and after any maintenance or testing that could affect the air distribution within the penetration room ventilation system, air distribution shall be demonstrated to be uniform within ±20% across HEPA filters and charcoal adsorbers.~~ — LA3 VFTP & LATER

<LATER> (3.7)

~~4.11.3 At intervals not to exceed 18 months, automatic initiation of the penetration room ventilation system shall be demonstrated.~~ — LATER

5.5.11 &(LATER) (3.7)

~~4.11.4a The tests and sample analysis of Specification 3.13.1a, b, & c. shall be performed at intervals not to exceed 18 months or after every 720 hours of system operation and following significant painting, fire or chemical release in any ventilation zone communicating with the system.~~

~~b. Cold DOP testing shall also be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.~~

~~c. Halogenated hydrocarbon testing shall also be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.~~ — LA3 VFTP & LATER

<LATER> (3.7)

~~4.11.5 Each circuit shall be operated at least 1 hour every month. This test shall be considered satisfactory if control board indication verifies that all components have responded properly to the actuation signal.~~ — LATER

5.5.11

4.17 FUEL HANDLING AREA VENTILATION SYSTEM SURVEILLANCE

<LATER> (3.7)

Applicability

Applies to the surveillance of the fuel handling area ventilation system. LATER

Objective

To verify an acceptable level of efficiency and operability of the fuel handling area ventilation system.

ANO-348

Specification

M11

5.5.11
5.5.11.d
& LATER (3.7)

4.17.1 ~~At intervals not to exceed 18 months,~~ pressure drop across the combined HEPA filters, and charcoal adsorber banks, other filters in the system, shall be demonstrated to be less than 6 inches of water at system design flow rate of 39000 cfm (±10%). LA3 VFTP & LATER M5

5.5.11 & LATER (3.7)

4.17.2 Initially and after any maintenance or testing that could affect the air distribution within the fuel handling area ventilation system, air distribution shall be demonstrated to be uniform within ±20% across HEPA filters and charcoal adsorbers. LA3 VFTP & LATER

5.5.11 & LATER (3.7)

4.17.3 a. The tests and sample analysis of Specification 3.15.1.a, b, & c shall be performed within 720 system operating hours prior to irradiated fuel handling operations in the auxiliary building, and prior to irradiated fuel handling in the auxiliary building following significant painting, fire or chemical release in any ventilation zone communicating with the system. LA3 VFTP & LATER

b. Cold DOP testing shall also be performed prior to irradiated fuel handling in the auxiliary building after each complete or partial replacement of a HEPA filter bank or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon testing shall also be performed prior to irradiated fuel handling in the auxiliary building after each complete or partial replacement of a charcoal adsorber bank or after any structural maintenance on the system housing.

<LATER> (3.7)

4.17.4 The system shall be operated for at least 10 hours prior to initiation of irradiated fuel handling operations in the auxiliary building if it has not been operated for at least 10 hours within the previous 30 days. LATER

Bases

Since the fuel handling area ventilation system may be in operation when fuel is stored in the pool but not being handled, its operability must be verified before handling of irradiated fuel. Operation of the system for 10 hours before irradiated fuel handling operations and performance of Specification 4.17.3 will demonstrate operability of the active system components and the filter and adsorber systems. A2

5.5.9 4.18 STEAM GENERATOR (SG) TUBE ~~TUBING~~ SURVEILLANCE PROGRAM A1

~~Applicability~~

~~Applies to the surveillance of tubing of each steam generator.~~ A1

~~Objective~~

This program provides controls to ensure the A1

~~To ensure~~ integrity of the steam generator tubing through a defined inservice surveillance program, and to minimize exposure of personnel to radiation during performance of the surveillance program.

<INSERT 110j A> A12

~~Specification~~

~~4.18.1 Baseline Inspection~~

a. The first steam generator tubing inspection performed ~~according to~~ in accordance with 5.5.9.b and 5.5.9.c.1 ~~Specifications 4.18.2 and 4.18.3.a~~ shall be considered as constituting the baseline condition for subsequent inspections.

b. ~~4.18.2~~ Examination Methods:

1 ~~a~~. Inservice inspection of steam generator tubing shall include non-destructive examination by eddy-current testing or other equivalent techniques. The inspection equipment shall provide a sensitivity that will detect defects with a penetration of 20 percent or more of the minimum allowable as-manufactured tube wall thickness except for a sleeved tube at the lower sleeve end. A1

2 ~~b~~. For examination of the sleeved steam generator tubing at the lower sleeve end, the indications will be compared to those obtained during the baseline sleeved tube inspection. Significant deviations between these indications will be considered sufficient evidence to warrant designation as a degraded tube. Direct quantification of the 40 percent through-wall plugging limit is available with eddy-current testing.

c. ~~4.18.3~~ Selection and Testing.

The steam generator sample size is specified in Table ~~4.18-1~~ 5.5.9-1. The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table ~~4.18-2~~ 5.5.9-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in ~~Specification 4.18.4~~ 5.5.9.d and the inspected tubes shall be verified acceptable per the acceptance criteria of ~~Specification 4.18.5~~ 5.5.9.e. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in both steam generators; the tubes selected for these inspections shall be selected on a random basis except: A1

<CTS INSERT 110jA>

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the SG Tube Surveillance Program inspection frequencies.

5.5.9

<All changes> — A1

1 ~~f~~. The first sample inspection during each inservice inspection (subsequent to the baseline inspection) of each steam generator shall include:

i ~~1~~. All nonplugged tubes that previously had detectable wall penetrations (>20%), except tubes in which the wall penetration has been spanned by a sleeve, and

ii ~~2~~. At least 50% of the tubes inspected shall be in those areas where experience has indicated potential problems, except where specific groups are inspected per ~~Specification 4.18.3.a.3.~~ 5.5.9.c.1.iii

A tube inspection (pursuant to ~~Specification 4.18.5.a.9~~ 5.5.9.e.1.ix) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

iii ~~3~~. Tubes in the following groups may be excluded from the first random sample if all tubes in a group in both steam generators are inspected. The inspection may be concentrated on those portions of the tubes where imperfections were previously found. No credit will be taken for these tubes in meeting minimum sample size requirements. Where only a portion of the tube is inspected, the remainder of the tube will be subjected to the random inspection.

(1) Group A-1: Tubes within one, two or three rows of the open inspection lane.

(2) Group A-2: Unplugged tubes with sleeves installed.

(3) Group A-3: Tubes in the wedge-shaped group on either side of the lane region (Group A-1) as defined by Figure ~~4.18.1~~ 5.5.9-1

iv ~~4~~. Tubes with axially-oriented tube end cracks (TEC) which have been left inservice for the previous cycle shall be inspected with a rotating coil eddy current technique in the area of the TEC and characterized in accordance with topical report BAW-2346P, Rev.0, during all subsequent SG inspection intervals pursuant to ~~4.18.4~~ 5.5.9.d. The results of this examination may be excluded from the first random sample. Tubes with axial TECs identified during previous inspections which meet the criteria to remain in service will not be included when calculating the inspection category of the OTSG.

v ~~5~~. Implementation of the upper tubesheet ODIGA alternate repair criteria requires a 100% bobbin coil inspection of the non-plugged and non-sleeved tubes, spanning the defined region of the upper tubesheet, during all subsequent SG inspection intervals pursuant to ~~4.18.4~~ 5.5.9.d. Tubes with ODIGA identified during previous inspections which meet the criteria to remain in service will not be included when calculating the inspection category for the OTSG. The defined region begins one inch above the upper tubesheet secondary face and ends at the nearest tube roll transition. ODIGA indications detected by the bobbin coil probe shall be characterized using rotating coil probes in accordance with topical report BAW-10235P, Revision 1.

5.5.9

<all changes> A1

1 ~~a~~. The first sample inspection during each inservice inspection (subsequent to the baseline inspection) of each steam generator shall include:

i ~~1~~. All nonplugged tubes that previously had detectable wall penetrations (>20%), except tubes in which the wall penetration has been spanned by a sleeve, and

ii ~~2~~. At least 50% of the tubes inspected shall be in those areas where experience has indicated potential problems, except where specific groups are inspected per ~~Specification 4.18.3.a.3~~ 5.5.9.c.1.iii

A tube inspection (pursuant to ~~Specification 4.18.5.a.9~~ 5.5.9.e.1.ix) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

iii ~~3~~. Tubes in the following groups may be excluded from the first random sample if all tubes in a group in both steam generators are inspected. The inspection may be concentrated on those portions of the tubes where imperfections were previously found. No credit will be taken for these tubes in meeting minimum sample size requirements. Where only a portion of the tube is inspected, the remainder of the tube will be subjected to the random inspection.

(1) Group A-1: Tubes within one, two or three rows of the open inspection lane.

(2) Group A-2: Unplugged tubes with sleeves installed.

(3) Group A-3: Tubes in the wedge-shaped group on either side of the lane region (Group A-1) as defined by Figure ~~4.18-1~~ 5.5.9-1

iv ~~4~~. Tubes with axially-oriented tube end cracks (TEC) which have been left inservice for the previous cycle shall be inspected with a rotating coil eddy current technique in the area of the TEC and characterized in accordance with topical report BAW-2346P, Rev.0, during all subsequent SG inspection intervals pursuant to ~~4.18.4~~ 5.5.9.d. The results of this examination may be excluded from the first random sample. Tubes with axial TECs identified during previous inspections which meet the criteria to remain in service will not be included when calculating the inspection category of the OTSG.

v ~~5~~. Implementation of the upper tubesheet ODIGA alternate repair criteria requires a 100% bobbin coil inspection of the non-plugged and non-sleeved tubes, spanning the defined region of the upper tubesheet, during all subsequent SG inspection intervals pursuant to ~~4.18.4~~ 5.5.9.d. Tubes with ODIGA identified during previous inspections which meet the criteria to remain in service will not be included when calculating the inspection category for the OTSG. The defined region begins one inch above the upper tubesheet secondary face and ends at the nearest tube roll transition. ODIGA indications detected by the bobbin coil probe shall be characterized using rotating coil probes in accordance with ANO Engineering Report No. 00-R-1005-01.

ANO-354

2. ~~b~~. All tubes which have been repaired using the reroll process will have the new roll area inspected during the inservice inspection.

3. ~~c~~. The second and third sample inspections ~~during each~~ inservice inspection as required by Table ~~4.18-2~~ 5.5.9-2 may be less than a full tube inspection by concentrating the inspection on those areas of the tube sheet array and on those portions of the tubes where tubes with imperfections were previously found. A1

4. The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected, are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

NOTES:

(1) In all inspections, previously degraded tubes whose degradations have not been spanned by a sleeve must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

(2) Where special inspections are performed pursuant to 5.5.9.c.1.iii ~~4.18.3.a.3~~, defective or degraded tubes found as a result of the inspection shall be included in determining the Inspection Results Category for that special inspection but need not be included in determining the Inspection Results Category for the general steam generator inspection.

(3) Where special inspections are performed pursuant to 5.5.9.c.2 ~~4.18.3.b~~ defective or degraded tube indications found in the new roll area as a result of the inspection and any indications found above the new roll area, are not included in the determination for the inspection results category of a general steam generator inspection.

A1

5.5.9

<ALL CHANGES> A1

~~4.18.4 Inspection Intervals~~

d. The above-required inservice inspections of steam generator tubes shall be performed at the following frequencies:

1. ~~a.~~ The baseline inspection shall be performed during the first refueling shutdown. Subsequent inservice inspections shall be performed at intervals of not less than 10 nor more than 24 calendar months after the previous inspection. If the results of two consecutive inspections for a given group* of tubes following service under all volatile treatment (AVT) conditions fall into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval for that group may be extended to a maximum of 40 months.

2. ~~b.~~ If the results of the inservice inspection of a steam generator performed in accordance with Table ~~4.18-2~~ 5.5.9-2 at 40-month intervals for a given group* of tubes fall in Category C-3, subsequent inservice inspections shall be performed at intervals of not less than 10 nor more than 20 calendar months after the previous inspection. The increase in inspection frequency shall apply until a subsequent inspection meets the conditions specified in ~~4.18.4.a~~ 5.5.9.d.1 and the interval can be extended to 40 months.

3. ~~c.~~ Additional unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table ~~4.18-2~~ 5.5.9-2 during the shutdown subsequent to any of the following conditions:

i. ~~1.~~ Primary-to-secondary leakage in excess of the limits of Specification ~~3.1.6.3.b~~ 3.4.13 (Inservice inspection not required if leaks originate from tube-to-tubesheet welds). If the leaking tube is from either Group A-1 or A-3 as defined in Specification ~~4.18.3.a.3~~ 5.5.9.c.1.iii, all of the tubes in the affected group in this steam generator may be inspected in lieu of the first sample inspection specified in Table ~~4.18-2~~ 5.5.9-2. If the degradation mechanism which caused the leak is limited to a specific portion of the tube length, the inspection per this paragraph may be limited to the affected portion of the tube length. If the results of this inspection fall into the C-3 category, all of the tubes in the same group in the other steam generator will also be similarly inspected.

If the leaking tube has been repaired by the reroll process and is leaking in the new roll area, all of the tubes in the steam generator that have been repaired by the reroll process will have the new roll area inspected. If the results of this inspection fall into the C-3 category, all of the tubes with rerolled areas in the other steam generator will also be similarly inspected. This inspection will be in lieu of the first sample inspection specified in Table ~~4.18-2~~ 5.5.9-2.

ii. ~~2.~~ A seismic occurrence greater than the Operating Basis Earthquake,

iii. ~~3.~~ A loss-of-coolant accident requiring actuation of the engineered safeguards, or

iv. ~~4.~~ A main steam line or feedwater line break.

---

*A group of tubes means: (a) All tubes inspected pursuant to ~~4.18.3.a.3~~ 5.5.9.c.1.iii or (b) All tubes in a steam generator less those inspected pursuant to ~~4.18.3.a.3~~

5.5.9

c ~~4.18.5~~ Acceptance Criteria

1 ~~a~~. As used in this specification:

i ~~1~~. Tubing or Tube means that portion of the tube or sleeve which forms the primary system to secondary system pressure boundary.

ii ~~2~~ Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

iii ~~3~~. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either the inside or outside of a tube.

iv ~~4~~ Degraded Tube means a tube containing imperfections ≥ 20% of the nominal wall thickness caused by degradation, except where all degradation has been spanned by the installation of a sleeve or repaired by a rerolled joint.

The reroll repair process be used to repair tubes with defects in the upper and lower tubesheet areas as described in topical report, BAW-2303P, Revision 4.

v ~~5~~. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

vi ~~6~~. Defect means an imperfection of such severity that it exceeds the plugging limit except where the imperfection has been spanned by the installation of a sleeve. A tube containing a defect in its pressure boundary is defective.

vii ~~7~~. Plugging Limit means the imperfection depth at or beyond 40% of the nominal tube wall thickness for which the tube shall be sleeved, rerolled, or removed from service because it may become unserviceable prior to the next inspection. This does not apply to ODIGA indications within the defined region of the upper tubesheet. These indications shall be assessed for continued plant operation in accordance with ANO Engineering Report No. 00-R-1005-01, Rev. 1.

Axially-oriented TEC indications in the tube that do not extend beyond the adjacent cladding portion of the tube sheet into the carbon steel portion are not included in this definition. These indications shall be assessed for continued plant operation in accordance with topical report BAW-2346P, Rev. 0.

viii ~~8~~ Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in ~~Specification 4.18.4.c~~ 5.5.9.d.3 A1

ix ~~9~~ Tube Inspection means an inspection of the steam generator tube from the point of entry completely to the point of exit. For tubes that have been repaired by the reroll process within the tubesheets, that portion of the tube outboard of the new roll can be excluded from future periodic inspection requirements because it is no longer part of the pressure boundary once the repair roll is installed.

ANO-345
ANO-354

A18
A19

5.5-9

A2

Bases

The surveillance requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

In general, steam generator tubes that are degraded beyond the repair limit can either be plugged, sleeved, or rerolled. The steam generator (SG) tubes that are plugged are removed from service by the installation of plugs at both ends of the associated tube and thus completely removing the tube from service. When the tube end cracking (TEC) alternate repair criteria is applied, axially-oriented indications found not to extend from the tube sheet cladding region into the carbon steel region may be left in service under the guidelines of topical report BAW-2346P, Rev. 0. When the upper tubesheet outer diameter intergranular attack (ODIGA) alternate repair criteria is applied, indications found within the defined region of the upper tubesheet may be left in service under the guidelines of topical report BAW-10235P, Revision 1. The defined region begins one inch above the upper tubesheet secondary face and ends at the nearest tube roll transition. Following a SG inspection, an operational assessment is performed to ensure primary-to-secondary leak rates will be maintained within the assumptions of the accident analysis.

Degraded steam generator tubes can also be repaired by the installation of sleeves which span the area of degradation and serve as a replacement pressure boundary for the degraded portion of the tube, thus permitting the tube to remain in service.

Degraded steam generator tubes can also be repaired by the rerolling of the tube in the upper or lower tubesheet to create a new roll area and pressure boundary for the tube. The portion of the tube that is outboard of the repair roll is the portion of the tube closest to the primary side of the tubesheet and includes tubing from the tube end up to and including the heel expansion transition. The 1-inch repair roll is considered to be within the pressure boundary. If more than one repair roll is present, the outboard portion includes tubing from the tube end to the heel transition and the beginning of the 1-inch repair roll that is farthest from the primary side of the tubesheet. The rerolling repair process will be used to repair defects in the upper and lower tubesheet in accordance with BAW-2303P, Revision 4.

All tubes which have been repaired using the reroll process will have the new roll area inspected during future inservice inspections. Defective or degraded tube indications found in the new roll and any indications found in the original roll need not be included in determining the Inspection Results Category for the generator inspection.

The reroll repair process can be used to repair tubes with defects in the upper and lower tubesheet areas. Installation of multiple repair rolls in a single tube is acceptable. The new roll area must be free of detectable degradation in order for the repair to be considered acceptable. After the new roll area is initially deemed acceptable, future degradation in the new roll area will be analyzed to determine if the tube is defective and needs to be removed from service or repaired. The reroll repair process is described in the topical report, BAW-2303P, Revision 4.

ANO-345

LAR

A18

TABLE ~~4.18-1~~ 5.5.9-1 A1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No
No. of Steam Generators per Unit	Two
First Inservice Inspection	Two
Second & Subsequent Inservice Inspection	One[1]

Table Notation:

[1] The inservice inspection may be limited to one steam generator on alternating schedule encompassing 3N% of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

TABLE ~~4.18-2~~ 5.5.9-2 A1

STEAM GENERATOR TUBE INSPECTION [2,3]

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.[1]	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug, reroll, or sleeve defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug, reroll, or sleeve defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug, reroll, or sleeve defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G. plug, reroll, or sleeve defective tubes and inspect 2S tubes in other S.G. ~~Special Report to NRC pursuant to 6.12.5.d~~	Other S.G. is C-1	None	N/A	N/A
			Other S.G. is C-2	Perform action for C-2 result of second sample	N/A	N/A
			Other S.G. is C-3	Inspect all tubes in each S.G. and plug, reroll, or sleeve defective tubes. ~~Special Report to NRC pursuant to 6.12.5.d~~	N/A	N/A

A4

NOTES: 1 S=3N/n % Where N is the number of steam generators in ~~the unit, and~~ n is the number of steam generators inspected during an inspection.

2 For tubes inspected pursuant to ~~4.18.3.c.3~~ 5.5.9.c.1.iii: No action ~~is~~ required for C-1 results. For C-2 results in one or both steam generators plug, reroll, or sleeve defective tubes. For C-3 results in one or both steam generators, plug, reroll, or sleeve defective tubes and provide a ~~Special~~ Report to NRC pursuant to ~~6.12.5.d~~ 5.6.7

3 No more than ten thousand (10,000) sleeves may be installed in both ~~ANO-1~~ steam generators combined.

A1

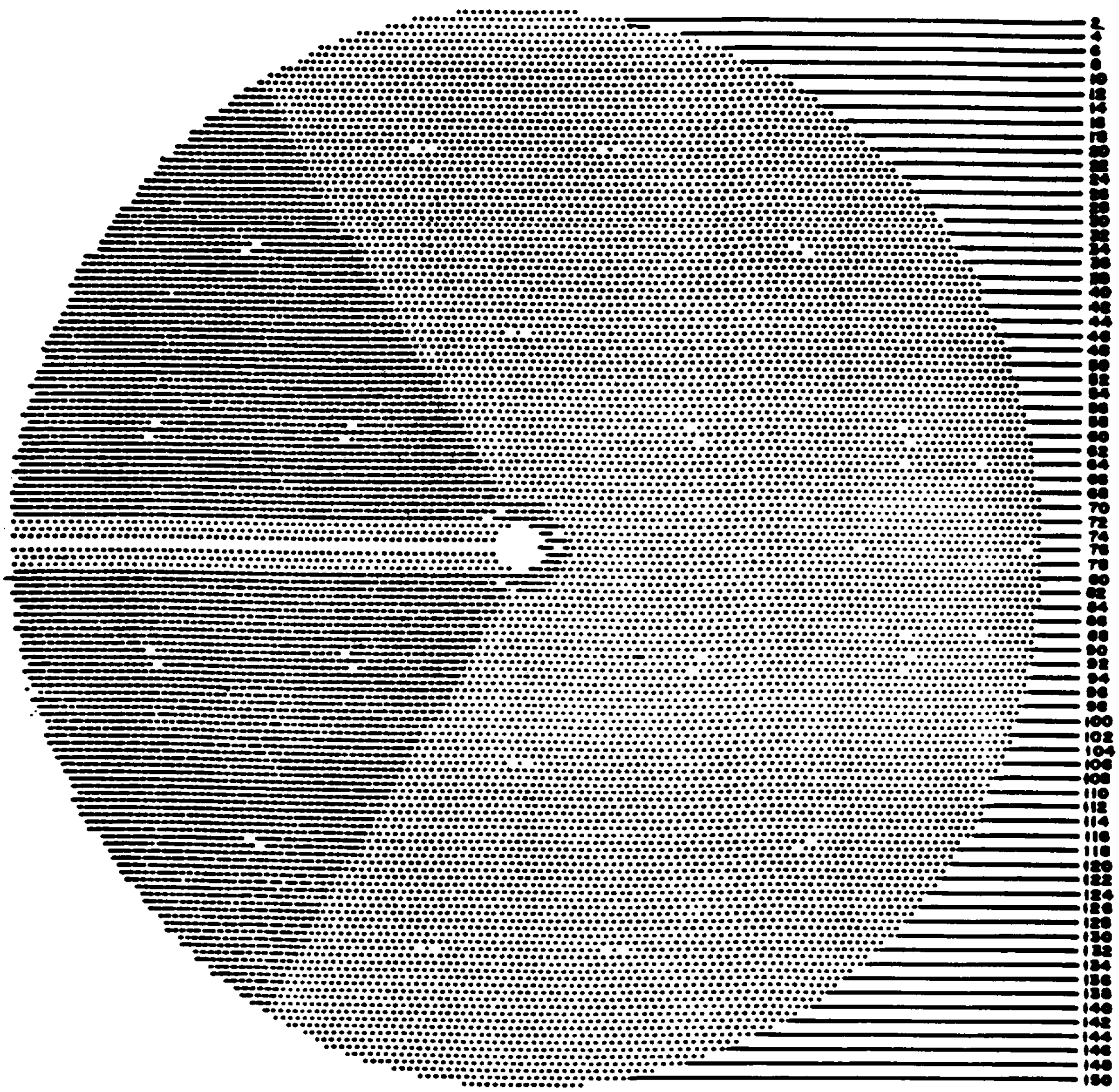
Amendment No. ~~24~~,~~41~~,~~86~~,~~106~~,~~118~~,~~172~~,190

5.5.9

Figure ~~4.18.1~~ 5.5.9-1

Upper Tube Sheet View of Wedge Shaped Group (Group A-3) per Specification ~~4.18.3.a.3~~ 5.5.9.c.1.iii

A1



DESCRIPTION	TUBE COUNT
GROUP A - 1: Lane region tubes as defined in 5.5.9.c.1.iii(1) ~~4.18.3.a.3(1)~~	382
GROUP A - 3: Wedge shaped group depicted by darkened region of figure	4880

A1

5.5.16

4.26 REACTOR BUILDING PURGE VALVES

Applicability

This specification applies to the reactor building purge supply and exhaust isolation valves.

Objective

To assure reactor building integrity.

Specification

A1

<LATER> (3.6) 4.26.1 The reactor building purge supply and exhaust isolation valves shall be determined closed at least once per 31 days when containment integrity is required by TS 3.6.1. LATER

5.5.16 4.26.2 Prior to exceeding conditions which require establishment of reactor building integrity per TS 3.6.1, the leak rate of the purge supply and exhaust isolation valves shall be verified to be within acceptable limits per TS 4.4.1, unless the test has been successfully completed within the last three months. M8

Bases

Determination of reactor building purge valve closure will ensure that reactor building integrity is not unintentionally breached.

As a result of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration," it was concluded that excess leakage past valve resilient seals is typically caused by severe environmental conditions and/or wear due to use. Recommended leak test frequencies of three months are deemed to be adequate to detect seal degradation of resilient seals.

The three month test need not be conducted with the precision of the Type C 10CFR50, Appendix J criteria, however the test must be sufficient to detect degradation.

A2

4.27 DECAY HEAT REMOVAL

APPLICABILITY

⟨LATER⟩ (3.4A) Applies to surveillance of the decay heat removal system and to the reactor coolant loops and associated reactor coolant pumps as needed for decay heat removal. LATER

OBJECTIVE

To assure the operability of the decay heat removal system and the reactor coolant loops as needed for decay heat removal.

SPECIFICATION

4.27.1 The required reactor coolant pumps shall be determined operable once per seven (7) days by verifying correct breaker alignments and indicated power availability.

4.27.2 The required decay heat removal loop(s) shall be determined operable per Specification 4.2.2. A5

⟨LATER⟩ (3.4A) 4.27.3 The required steam generator(s) shall be determined operable by verifying the secondary side water level to be ≥20 inches on the startup range at least once per 12 hours. LATER

4.27.4 The required reactor coolant loop(s) shall be determined operable by verifying the required loop(s) to be in operation and circulating reactor coolant at least once per 12 hours.

⟨LATER⟩ (3.4A & 3.9) 4.27.5 The required decay heat removal loop shall be determined to be in operation at least once per 12 hours. LATER

5.5.12

~~4.28 EXPLOSIVE GAS MIXTURE~~

~~Applicability~~

~~Applies to the Waste Gas System hydrogen/oxygen analyzers.~~

~~Objective~~

~~To prevent accumulation of explosive mixtures in the waste gas system.~~

~~Specification~~

~~4.28.1 The concentration of hydrogen/oxygen in the waste gas system shall be monitored continuously by either the primary or redundant waste gas analyzer during waste gas compressing operations to the waste gas decay tanks.~~

~~4.28.2 During waste gas system operation, with no H_2/O_2 analyzer in service, without delay suspend all additions of waste gas to the decay tanks or take grab samples for analysis every 4 hours during degassing operations, daily during other operations. The analysis of these samples shall be completed within 8 hours of taking the sample.~~

LA5

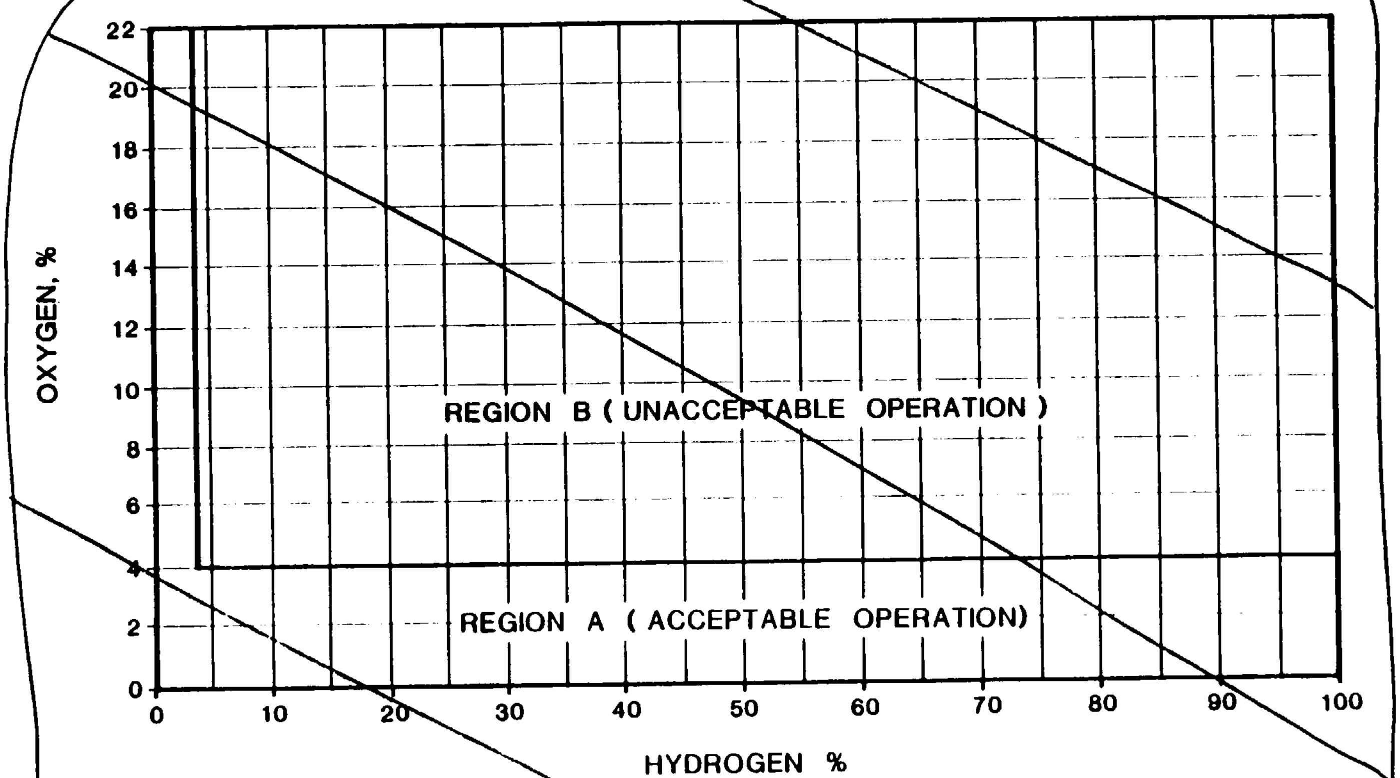
EGISTRMP

~~Bases~~

~~This specification is to assure that the hydrogen/oxygen concentration will be kept within the limits in Figure 3.24-1 and therefore not enter the flammable region concentrations in the waste gas decay tanks.~~

~~Grab samples are to be taken every 4 hours during degassing operations when both hydrogen/oxygen analyzers are out of service. These samples are to be analyzed within 8 hours to assure that the hydrogen/oxygen concentration is within the limits in Figure 3.24-1. During other Waste Gas compressor operations, the hydrogen/oxygen concentration is not as subject to change, therefore grab samples are to be taken every 24 hours.~~

A2



HYDROGEN – OXYGEN LIMITS FOR ANO – 1 WASTE GAS SYSTEM

FIGURE 3.2 4-1

LA5

EG&TRMP

5.5.12

5.5.12

LA5

ESGSTRMD

4.29 RADIOACTIVE EFFLUENTS

4.29.1 <u>Radioactive Liquid Holdup Tanks</u>

<u>Applicability:</u> At all times

<u>Objective:</u> To ensure that the limits of 10 CFR 20 are not exceeded.

<u>Specification:</u>

4.29.1 The quantity of radioactive material contained in an outside temporary radioactive liquid storage tank shall be determined to be within the limit of Specification 3.25.1 by analyzing a representative sample of the contents of the tank at least once per 7 days when radioactive materials are being added to the tank.

<u>Bases:</u>

A2

This specification is provided to ensure that in the event of an uncontrolled release of the contents of the tank the resulting concentrations would be less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in the unrestricted area.

5.5.12

4.29.2 Radioactive Gas Storage Tanks

Applicability: At all times

Objective: To ensure meeting the requirements of Specification 3.25.2.

Specification:

4.29.2 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the limits of Specification 3.25.2 at least once per 24 hours when radioactive materials are being added to the tank and the reactor coolant activity exceeds the limits of Specification 3.1.4.1.b.

LAS

EGr STRMP

Bases:

This specification is provided so that the requirements of Specification 3.25.2 are met.

A2

5.1
5.2.1
5.2.2

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

5.1.1 ~~6.1.1~~ The ANO-1 plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

5.1.2 ~~6.1.2~~ An individual with an active Senior Reactor Operator (SRO) license shall be designated as responsible for the control room command function while the unit is ~~above the Cold Shutdown condition~~ MODES 1, 2, 3, 4. With the unit not ~~above the Cold Shutdown condition~~, an individual with an active SRO license or Reactor Operator license shall be designated as responsible for the control room command function. A1

6.2 ORGANIZATION

5.2.1 ~~6.2.1~~ OFFSITE AND ONSITE ORGANIZATIONS

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements including the unit specific titles of those personnel fulfilling its responsibilities of the positions delineated in these Technical Specifications shall be documented in the Safety Analysis Report (SAR).

b. The ANO-1 plant manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. A specified corporate executive shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety. The specified corporate executive shall be documented in the SAR.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

UNIT STAFF

5.2.2.f ~~6.2.2~~ The operations manager or the assistant operations manager shall hold a senior reactor operator license. ~~Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.~~

A5

5.2.2
5.3.1

5.2.2.e ~~6.2.2.1~~ Administrative controls shall be established to limit the amount of overtime worked by plant staff performing safety-related functions. These administrative controls shall be in accordance with the guidance provided by the NRC Policy Statement on working hours (Generic Letter 82-12).

6.3. FACILITY STAFF QUALIFICATIONS

ANS 3.1-1978

5.3.1 ~~6.3.1~~ Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI ~~N18.1-1971~~ for comparable position, except for the designated radiation protection manager, who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. M1

~~6.4 DELETED~~

~~6.5 DELETED~~

A1

5.2.2

Table 6.2-1

ARKANSAS NUCLEAR ONE

MINIMUM SHIFT CREW COMPOSITION #

UNIT 1

A1

	LICENSE CATEGORY	ABOVE COLD SHUTDOWN	COLD AND REFUELING SHUTDOWNS	
5.2.2.b	SOL	2	1*	A5
	OL	2	1	
5.2.2.a	NON-LICENSED	~~3~~ 2	1	LA2

*Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising refueling operations after the initial fuel loading. SAR A5

5.2.2.c #Shift crew composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements ~~of Table 6.2-1~~. A1

Additional Requirements:

1. At least one licensed Operator shall be in the control room when fuel is in the reactor. A5

2. At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.

5.2.2.d ~~3.~~ An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

4. All refueling operations after the initial fuel loading shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation. A5

5.2.2.g 5. ~~When the unit is above the Cold Shutdown condition~~ In MODES 1, 2, 3, and 4 an individual shall provide advisory technical support for the unit operations shift supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. A1

5.4.1
5.5.1
5.5.3

6.6 DELETED — A1

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

<LATER>
(2.0)

a. The facility shall be placed in at least hot shutdown within one hour. — LATER

b. The Nuclear Regulatory Commission shall be notified pursuant to 10 CFR 50.72 and a report submitted pursuant to the requirements of 10 CFR 50.36 and Specification 6.6.

6.8 PROCEDURES AND PROGRAMS

5.4.1 6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

5.4.1.a a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, ~~November, 1972~~ Rev 2, February 1978. — M3

b. Refueling operations.

c. Surveillance and test activities of safety related equipment. — A1

d. (Deleted)

e. (Deleted)

5.4.1.c f. Fire Protection Program Implementation.

g. New and spent fuel storage. — A1

5.4.1.d (5.5.1) h. Offsite Dose Calculation Manual and Process Control Program implementation at the site. — A1

<ADD: 5.4.1.d for "other programs">
<ADD: 5.4.1.b for EOPs per GL 82-33> — M3

ANO-3HD

5.5.16

~~6.8.2~~ (Deleted) A1

~~6.8.3~~ (Deleted)

5.5.16 ~~6.8.4~~ The Reactor Building Leakage Rate Testing Program shall be established, implemented, and maintained: A1

A program shall be established to implement the leakage rate testing of the reactor building as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated reactor building internal pressure for the design basis loss of coolant accident, P_a, is 54 psig.

The maximum allowable reactor building leakage rate, L_a, shall be 0.20% of containment reactor building air weight per day at P_a. A1

Reactor building leakage rate acceptance criteria is $\leq 1.0\ L_a$. During the first unit startup following each test performed in accordance with this program, the leakage rate acceptance criteria are $\boxed{\leq}$ < 0.60 L_a for the Type B and Type C tests and $\boxed{\leq}$ < 0.75 L_a for Type A tests. A11

The provisions of ~~Specification 4.0.2~~ SR 3.0.2 do not apply to the test frequencies specified in the Reactor Building Leakage Rate Testing Program. A1

The provisions of ~~Specification 4.0.3~~ SR 3.0.3 are applicable to the Reactor Building Leakage Rate Testing Program. A1

5.5.4

5.5.4 6.8.5 **The Radioactive Effluent Controls Program shall be established, implemented, and maintained:**

This program conforms with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas conforming to 10 CFR 20, Appendix B, Table II, Column 2; A17

<INSERT 127a A>

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative ~~and projected~~ dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days; A17

<INSERT 127a B>

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose ~~commitment~~, conforming to 10 CFR 50, Appendix I;

g. Limitations on ~~the dose rate~~ from the site resulting from radioactive material released in gaseous effluents to areas at or beyond the site boundary ~~conforming to the dose associated with 10 CFR 20, Appendix B, Table II, Column 1~~; A1 A17

<INSERT 127a C>

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on ~~the annual dose or dose commitment~~ to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190. A1

<INSERT 127a D> A17

<CTS INSERT 127aA>

... ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402.

<CTS INSERT 127aB>

. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM ...

<CTS INSERT 127aC>

... shall be in accordance with the following:

1. For noble gases: a dose rate ≤ 500 mrem/yr to the whole body and a dose rate ≤ 3000 mrem/yr to the skin, and
2. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrem/yr to any organ;

<CTS INSERT 127aD>

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

5.0

~~6.9 DELETED~~ A1

6.10 RADIATION PROTECTION PROGRAM (A5)

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

5.7 6.11 HIGH RADIATION AREA

5.7.1 6.11.1 In lieu of the "control device" or "alarm signal" required by
5.7.1.a paragraph 20.203(c)(2) of 10CFR20, each high radiation area (as defined in (A1)
5.7.1.b 20.202(b)(3) of 10CFR20) in which the intensity of radiation is 1000
<INSERT 129 A> mrem/hr or less shall be barricaded and conspicuously posted as a high
radiation area and shall be controlled by requiring the issuance of a (L2)
<INSERT 129 B> radiation work permit. Any individual or group of individuals permitted
5.7.1.d to enter such areas shall be provided with or accompanied by one or more of the following:

5.7.1.d.1 a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

5.7.1.d.2 b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a pre-set
<INSERT 129 C> integrated dose is received. Entry into such areas with this
5.7.1.e monitoring device may be made only after the dose rate level in the (L2)
area has been established and personnel have been made (A1)
<INSERT 129D> knowledgeable of them.

5.7.1.d.4.i c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the radiation work permit.

Note: this text is repeated in insert 129C

<CTS INSERT 129A>

Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

<CTS INSERT 129B>

... or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

<CTS INSERT 129C>

..., with an appropriate alarm setpoint, or

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

   (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

   (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, ...

<CTS INSERT 129D>

... These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7.2 ~~6.11.2~~ ~~The requirements of 6.11.1 above, shall also apply to each~~ high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. ~~In addition, locked doors shall be provided to prevent unauthorized entry into such areas and access to these areas shall be maintained under the administrative control of the shift supervisor on duty and/or the designated radiation protection manager.~~ L2

<INSERT 129a A>

<CTS INSERT 129aA>

... at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

   1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designee.

   2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

   1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

   2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

   3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

      (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

      (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

   4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

<CTS INSERT 129aA>
(continued)

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. . These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

5.6
5.6.1

6.12 REPORTING REQUIREMENTS

6.12.1 ~~In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Administrator of the appropriate NRC Regional Office unless otherwise noted.~~ A5

6.12.2 Routine Reports

~~6.12.2.1 Startup Report~~

~~A summary report of plant startup and power escalation testing shall be submitted following 1) receipt of an operating license, 2) amendment to the license involving a planned increase in power level, 3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and 4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.~~ LA6 TRM

~~Startup reports shall be submitted within 1) 90 days following completion of the startup test program, 2) 90 days following resumption or commencement of commercial power operation, or 3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.~~

5.6.1 6.12.2.2 Occupational Exposure Data Report 1/ L3a

An Occupational Exposure Data Report for the previous calendar year shall be submitted by April 30 ~~prior to March 1~~ of each year. The report shall contain a tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving ~~exposures~~ greater than 100 mrem ~~/yr~~ and their associated ~~man~~ rem ~~exposure~~ according to work and job functions, 2/ e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling.

, for whom monitoring was performed,

an annual deep dose equivalent

collective deep dose equivalent (reported in person-

1/ A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

2/ This tabulation supplements the requirements of ~~20.407 of~~ 10 CFR ~~Part~~ 20.2206.

A13

5.6.1 Note....

5.6.1
5.6.2
5.6.3

ionization chamber
electronic dosimeter

5.6.1 The dose assignments to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions. A13

deep dose equivalent

5.6.4 6.12.2.3 Monthly Operating Report

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis by the 15th of each month following the calendar month covered by the report.

~~6.12.2.4 Annual Report~~

~~All challenges to the pressurizer electromatic relief valve (ERV) and pressurizer safety valves shall be reported annually.~~ L3b

5.6.2 6.12.2.5 Annual Radiological Environmental Operating Report *

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 6.12.2.6 Radioactive Effluent Release Report **

in the previous year

prior to May 1 of each year

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

A14

5.6.2 NOTE * A single submittal may be made for ANO. The submittal should combine those sections that are common to both units.

shall

5.6.3 NOTE ** A single submittal may be made for ANO. The submittal ~~should~~ combine those sections that are common to both units. The submittal shall specify the releases of radioactive material from each unit.

5.6.5

5.6.5

6.12.3 CORE OPERATING LIMITS REPORT

6.12.3.1 The core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle or prior to any remaining part of a reload cycle for the following Specifications:

2.1	Safety Limits, Reactor Core - Axial Power Imbalance protective limits and Variable Low RCS Pressure-Temperature Protective Limits
2.3.1	Reactor Protection System trip setting limits - Protection System Maximum Allowable Setpoints for Axial Power Imbalance and Variable low RC system pressure
3.1.8.3	Minimum Shutdown Margin for Low Power Physics Testing
3.5.2.1	Allowable Shutdown Margin limit during Power Operation
3.5.2.2	Allowable Shutdown Margin limit during Power Operation with inoperable control rods
3.5.2.4	Quadrant power Tilt limit
3.5.2.5	Control Rod and APSR position limits
3.5.2.6	Reactor Power Imbalance limits

6.12.3.2 The analytical methods used to determine the core operating limits addressed by the individual Technical Specification shall be those previously reviewed and approved by the NRC in Babcock & Wilcox Topical Report BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses" (the approved revision at the time the reload analyses are performed). The approved revision number shall be identified in the CORE OPERATING LIMITS REPORT.

6.12.3.3 The core operating limits shall be determined so that all applicable limits (e.g. fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.12.3.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance for each reload cycle to the NRC ~~Document Control Desk with copies to the Regional Administrator and Resident Inspector.~~ A5

Note: Next non-blank page is 146

5.6.6

5.6.6 ~~6.12.4~~ Reactor Building Inspection Report

~~6.12.4.1~~ Any degradation exceeding the acceptance criteria of the containment structure detected during the tests required by the Containment Inspection Program shall undergo an engineering evaluation within 60 days of the completion of the inspection surveillance. The results of the engineering evaluation shall be reported to the NRC within an additional 30 days of the time the evaluation is completed. The reportshall include the cause of the condition that does not meet the acceptance criteria, the applicability of the conditions to the other unit, the acceptability of the concrete containment without repair of the item, whether or not repair or replacement is required and, if required, the extent, method, and completion date of necessary repairs, and the extent, nature, and frequency of additional examinations.

5.6.7

5.6 6.12.5 Special Reports

Special reports shall be submitted to the Administrator of the appropriate Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification. A5

a. Deleted A1

<LATER> (3.3D) b. Inoperable Containment Radiation Monitors, Specification 3.5.1, Table 3.5.1-1. LATER

c. Deleted A1

5.6.7 d. Steam Generator Tubing Surveillance - Category C-3 Results, Specification ~~4.20~~ 5.5.9

<LATER> (3.7) e. Miscellaneous Radioactive Materials Source Leakage Tests, Specification 3.12.2. LATER

f. Deleted

g. Deleted

h. Deleted

i. Deleted A1

<LATER> (3.8) j. Degraded Auxiliary Electrical Systems, Specification 3.7.2.H. LATER

k. Inoperable Reactor Vessel Level Monitoring Systems, Table 3.5.1-1

<LATER> (3.3D) l. Inoperable Hot Leg Level Measurement Systems, Table 3.5.1-1 LATER

m. Inoperable Main Steam Line Radiation Monitors, Specification 3.5.1, Table 3.5.1-1.

5.5.14
5.5.15

~~6.13~~ (DELETED) A1

Add ITS 5.5.14, "Tech Spec Bases Control Program" M7

Add ITS 5.5.15, "Safety Function Determination Program M7

5.5.1

5.5.1

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program.

The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports ~~required by Specifications 6.12.2.5 and 6.12.2.6.~~ A1

Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

   2. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after approval of the ~~General Manager, Plant Operations~~ AND; and LA1 SAR

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed and shall also indicate the date (i.e., month and year) the change was implemented.

"R" - Relocation of requirements:

Relocating requirements which do not meet the Technical Specification selection criteria to documents with an established control program allows the Technical Specifications to be reserved only for those conditions or limitations upon reactor operation which are necessary to adequately limit the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety, thereby focusing the scope of Technical Specifications.

Therefore, requirements which do not meet the Technical Specification selection criteria in 10 CFR 50.36 have been relocated to other controlled license basis documents. This regulation addresses the scope and purpose of Technical Specifications. In doing so, it establishes a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in Technical Specifications. These criteria are as follows:

Criterion 1: Installed instrumentation that is used to detect and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2: A process variable that is an initial condition of a design basis accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3: A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission barrier.

Criterion 4: A structure, system or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

The application of these criteria is provided in the "Application of Selection Criteria to the ANO-1 Technical Specifications." Requirements which met the criteria have been included in the proposed improved Technical Specifications. Entergy Operations proposes to remove the requirements which do not meet the criteria from the Technical Specifications and relocate the requirements to a suitable owner controlled document. The requirements in the relocated Specifications will not be affected by this Technical Specification change. Entergy Operations will initially continue to perform the required operation and maintenance to assure that the requirements are satisfied. Relocating specific requirements for systems or variables will have no impact on the system's operability or the variable's maintenance, as applicable.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

License basis document control mechanisms, such as 10 CFR 50.59, 10 CFR 50.54(a)(3), and ITS Section 5, "Administrative Controls," will be utilized for the relocated Specifications as they will be placed in other controlled license basis documents. This would allow Entergy Operations to make changes to these requirements, without NRC approval, as allowed by the applicable regulatory requirements. These controls are considered adequate for assuring structures, systems and components in the relocated Specifications are maintained operable and variables in the relocated Specifications are maintained within limits.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as indicated below:

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change relocates requirements and surveillances for structures, systems, components or variables which did not meet the criteria for inclusion in Technical Specifications as identified in the Application of Selection Criteria to the ANO-1 Technical Specifications. The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected structures, systems, components or variables will be relocated from the Technical Specifications to an appropriate administratively controlled license basis document and maintained pursuant to the applicable regulatory requirements. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or change in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the affected requirement will be relocated to an owner controlled license basis document for which future changes will be evaluated pursuant to the requirements of the applicable regulatory requirements. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

"A" - Administrative changes to requirements:

Reformatting and rewording the remaining requirements in accordance with the style of the improved Babcock & Wilcox Standard Technical Specifications in NUREG-1430 will make the Technical Specifications more readily understandable to plant operators and other users. Application of the format and style will also assure consistency is achieved between specifications. As a result, the reformatting and rewording of the Technical Specifications has been performed to make them more readily understandable by plant operators and other users. During this reformatting and rewording process, no technical changes (either actual or interpretational) to the Technical Specifications were made unless they were identified and justified.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as indicated below:

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change involves reformatting and rewording of the existing Technical Specifications. The reformatting and rewording process involves no technical changes to existing requirements. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   The proposed change will not significantly reduce the margin of safety because it has no impact on any safety analysis assumptions. This change is administrative in nature. As such, there is no technical change to the requirements and therefore, there is no significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

"LA" - Less restrictive, Administrative deletion of requirements:

Portions of some Specifications provide information that is descriptive in nature regarding the equipment, system(s), actions or surveillances. This information is proposed to be deleted from the specification and relocated to other license basis documents which are under licensee control. These documents include the TS Bases, Safety Analysis Report (SAR), Technical Requirements Manual, and Programs and Manuals identified in ITS Section 5, "Administrative Controls." The removal of descriptive information is permissible, because the documents containing the relocated information will be controlled through the applicable process provided by the regulatory requirements, e.g., 10 CFR 50.59, 10 CFR 50.54(a)(3), and ITS Section 5, "Administrative Controls." This will not impact the actual requirements but may provide some flexibility in how the requirement is conducted. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as indicated below:

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change relocates requirements from the Technical Specifications to other license basis documents which are under licensee control. The documents containing the relocated requirements will be maintained using the provisions of applicable regulatory requirements. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be transposed from the Technical Specifications to other license basis documents, which are under licensee control, are the same as the existing Technical Specifications. The documents containing the relocated requirements will be maintained using the provisions of applicable regulatory requirements. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

"M" - More restrictive changes to requirements:

The ANO-1 Technical Specifications are proposed to be modified in some areas to impose more stringent requirements than previously identified. These more restrictive modifications are being imposed to be consistent with the improved Babcock & Wilcox Standard Technical Specifications. Such changes have been made after ensuring the previously evaluated safety analysis was not affected. Also, other more restrictive technical changes have been made to achieve consistency, correct discrepancies, and remove ambiguities from the specification.

The modification of the ANO-1 Technical Specifications and the changes made to achieve consistency within the specifications have been performed in a manner such that the most stringent requirements are imposed, except in cases which are individually evaluated.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change provides more stringent requirements for the ANO-1 Technical Specifications. These more stringent requirements are not assumed to be initiators of analyzed events and will not alter assumptions relative to mitigation of accident or transient events. The change has been confirmed to ensure no previously evaluated accident has been adversely affected. The more stringent requirements are imposed to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes do not impact the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated for ANO-1.

3. Does this change involve a significant reduction in a margin of safety?

The imposition of more stringent requirements prevents a reduction in the margin of plant safety by:

a) Increasing the analytical or safety limit,
b) Increasing the scope of the specification to include additional plant equipment,
c) Increasing the applicability of the specification,
d) Providing additional actions,
e) Decreasing restoration times,
f) Imposing new surveillances, or
g) Decreasing surveillance intervals.

The change is consistent with the safety analysis and licensing basis. Therefore, this change does not involve a reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

ITS Section 5.0: Administrative Controls

Entergy Operations has evaluated these proposed Technical Specification changes and has determined that they involve no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92(c) as indicated below:

5.0 L1 Not used

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

<u>5.0 L2</u>

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The controls for access to a high radiation area are not considered as initiators, nor as a mitigation factor, in any previously evaluated accident. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

No changes are proposed in the manipulation of the plant structures, systems, or components, or in the design of the plant structures, systems, or components. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The requirements for control of high radiation areas provide for the use of alternates to the "control device" or "alarm signal" requirements of 10 CFR 20.1601. This change provides such alternative methods for controlling access. These methods and additional administrative requirements have been determined to provide adequate controls to prevent unauthorized and inadvertent access to such areas. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

5.0 L3

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This change does not result in any changes in hardware or methods of operation. The change in date for submittal of "after the fact" information is not considered in the safety analysis, and cannot initiate or affect the mitigation of an accident in any way. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will impact only the administrative requirements for submittal of information and do not directly impact the operation of the plant. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The margin of safety is not dependent on the submittal of information. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

5.0 L4

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

The DC Sources are used to support mitigation of the consequences of an accident. Equipment powered by the DC Sources continues to be evaluated for loss of function, and previously determined appropriate ACTIONS for such inoperabilities continue to be required. Experience with the reliability of the DC sources indicates that the proposed increase in the Completion Time will not significantly increase the probability of a loss of electric power accident or of any other accident previously evaluated. The proposed ACTION continues to provide adequate assurance of OPERABLE required equipment and therefore, does not involve a significant increase in the consequences of any accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure corrective actions are taken to restore plant systems to OPERABLE status, as assumed in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

This change does not involve a significant reduction in a margin of safety since the OPERABILITY of the equipment and loss of function continue to be evaluated in the same manner. The increase in time allowed for such an evaluation and restoration is minimal and provides additional potential for the preferred action of restoration of the equipment to OPERABLE status, rather than requiring a shutdown transient.

5.0 L5

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

CTS Table 4.1-2, item 12 requires verification of a flow limiting gap that exists between the main feedwater line pipe and an annulus attached to the reactor building penetration on a periodic Frequency. The circular plate which provides the flow limitation is welded to the penetration and not subject to fluctuation except due to radial expansion during heatup which is considered in the design. Therefore, a change in the Frequency to require this verification following any modifications which may affect the required gap continues to provide adequate assurance of this design feature. Additionally, this verification is removed from the Technical Specifications since it is a specific design feature of a structure which is only subject to change via the design change process. As such, the "post-modification" verifications are also required by the design change process, and as with other post-modification type requirements, can be removed from the Technical Specifications without a significant impact on safety. This change does not result in any changes in hardware or methods of operation. Neither the flow limiting gap, nor the hardware which provides the gap is assumed to initiate an accident. Therefore, this change does not involve a significant increase in the probability of an accident previously evaluated. The design does provide for mitigation of a design basis pipe break to limit the consequences. However since the gap is provided by a design feature which is only subject to change by the design change process, periodic verification is unnecessary. Further, removal of this requirement from the Technical Specifications does not change the hardware, nor remove the design controls in place. Therefore, this change does not involve a significant increase in the consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will impact only the administrative requirements for periodic verification of a design feature and do not directly impact the operation of the plant. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The margin of safety is not dependent on the periodic verification of this structural design feature since it is only subject to change by the design change process. Therefore, this change does not involve a significant reduction in a margin of safety.

5.0 L6

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The testing of diesel generator fuel oil is not considered an initiator, or a mitigating factor, in any previously evaluated accident. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

No changes are proposed in the manipulation of the plant structures, systems, or components, or in the design of the plant structures, systems, or components. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The testing of stored diesel generator fuel oil is revised to require the periodic testing of the stored fuel oil only for particulates (replacing the periodic testing per ASTM-D975) once every 31 days. The change reflects industry-standard acceptable DG fuel oil testing programs. Over the storage life of ANO-1 DG fuel oil, the properties tested by ASTM-D975 are not expected to change and performing these tests once on the new fuel oil (see DOC M9) provides adequate assurance of the proper initial quality of fuel oil. The periodic testing for particulates monitors a parameter that reflects degradation of fuel oil and can be trended to provide increased confidence that the stored DG fuel oil will support DG operability. Therefore, this change does not involve a significant reduction in a margin of safety.

ITS DISCUSSION OF DIFFERENCES
ITS Section 5.0: Administrative Controls

1 NUREG 5.1.1, 5.2.1, 5.2.2, & 5.5.1 - Incorporates TSTF-065, Rev 1.

Unit specific changes consistent with current license basis include:

1) The ANO-1 unit specific designator is added to clearly establish the separate requirements that exist for ANO-1 and ANO-2. This prevents possible misinterpretation that the same individual may occupy this position for both ANO units.
2) Current Technical Specifications (CTS) Table 6.2-1 "additional requirement" number 3 is retained in Improved Technical Specifications (ITS) 5.2.2.c as "an individual qualified in radiation protection procedures." This maintains the greater flexibility provided by the CTS for fulfilling this position requirement.

2 NUREG 5.2.2 - In the discussion of Unit Staff (ITS 5.2.2), plant specific clarifications are provided to reflect the station two unit design, and that the two units share a common control room envelope, but the control rooms are separated. Unit specific terminology is incorporated to clarify applicability of requirements on a unit specific basis since the unit operations staff is assigned in this manner (i.e., to either ANO-1 or ANO-2), and a specific identification is provided for the applicable column of the table in 10 CFR 50.54(m)(2)(i). The shift manning requirements for "one unit, one control room" are considered to be applicable to each unit at ANO on an individual basis due to the dissimilarity of design of the units. ANO does not attempt to license individuals on both units simultaneously. These changes are consistent with current license basis.

3 NUREG 5.4.1 - An additional clarification is provided in proposed ITS 5.4.1.b to identify the appropriate discussions of emergency operating procedure requirements in Generic Letter 82-33. This change involves no revision of the actual requirements since Section 7 is the only portion of the identified Generic Letter which requires upgrades to the emergency operating procedures. Rather the change provides an editorial clarification to prevent possible misinterpretation of requirements to provide emergency operating procedures for all items identified in the Generic Letter. This change is consistent with current license basis.

4 Not used.

5 NUREG 5.4.1 - The NUREG 5.4 requirements to establish, implement and maintain written procedures covering the activity of "quality assurance for effluent and environmental monitoring" are not adopted. Procedures for effluent and environmental monitoring are required by 10 CFR 50 and Appendix I of Part 50. The QAPM is considered applicable to the implementation procedures for effluent and environmental monitoring for the station. Further, this activity is appropriately addressed in the station Environmental Report (ER) with the following statements: "Radiological analytical methods used in the radiological monitoring program are described in approved procedures as required by the Quality Assurance program for operations.

Additionally, procedure implementation and records are subject to periodic audit by the Quality Assurance Organization." (Ref. ANO-2 ER Section 6.1; Note that the ER is applicable to the site and thus appropriate for both units 1 and 2.) This periodic audit function continues to be implemented through the current QAPM Section 18.3.2.f which provides for periodic audits of the Radiological Environmental Monitoring Program. These controls are considered sufficient since they are not directly pertinent to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. Since these details are also not necessary to adequately describe the pertinent regulatory requirement, they are not mandated by 10 CFR 50.36, and they do not meet the criteria in 10 CFR 50.36, they can be appropriately retained in licensee controlled documents without a significant impact on safety. Retaining these requirements in controlled documents also provides adequate assurance that they will be maintained. Changes to the QAPM are controlled by 10 CFR 50.54. Since the controls are consistent with the QA controls for other activities, the specific listing for effluent and environmental monitoring is unnecessary.

6 NUREG 5.3.1 – NUREG 5.3.1 was revised to reflect CTS 6.3.1 requirements for staff qualification. These changes are consistent with current license basis and QAPM.

7 NUREG 5.5.1 - The RSTS cross reference to other Specifications is not adopted in ITS 5.5.1.b. This is a simple editorial change in presentation which has no impact on the actual requirement. Typically, cross references are not provided in the ITS, and this change is made to provide consistency both within the proposed Specifications and with the previously approved ITS for other EOI stations, i.e., Grand Gulf and River Bend.

8 NUREG 5.5.5 - The program identified in NUREG 5.5.5, "Component Cyclic or Transient Limit," is not adopted for the ANO-1 ITS. This program is currently administratively controlled (Procedure 1010.002) and the limits are addressed in the SAR (and therefore changes are controlled pursuant to 10 CFR 50.59). This is considered adequate for these design limits, and they are therefore, proposed to continue to be so controlled. This change is consistent with current license basis.

9 NUREG 5.5.7 – This change incorporates the CTS 4.2.6 requirements for the Reactor Coolant Pump Flywheel Inspection Program as ITS 5.5.7. ANO-1 is not committed to the requirements of Regulatory Guide 1.14, Revision 1, as stated in the NUREG. Therefore, the current ANO-1 surveillance requirements have been retained. In addition, an SR 3.0.2/SR 3.0.3 applicability statement has been added. The current ANO-1 requirements allow the application of the CTS 4.0.2 and CTS 4.0.3 provisions (which correspond to SR 3.0.2 and SR 3.0.3, respectively) to the CTS 4.2.6 requirements. These changes maintain the current ANO-1 licensing basis in ITS 5.5.7.

10 NUREG 5.5.16 – This change incorporates the CTS 6.8.4 requirements for the Reactor Building Leakage Rate Testing Program as ITS 5.5.16. The ITS Program is virtually identical to CTS requirements with the exception of the following:

1) A minor change was made to correct the reference to ITS SR 3.0.2 and SR 3.0.3 in lieu of CTS 4.0.2 and 4.0.3.
2) The CTS 4.26.2 requirement for leak rate testing of the reactor building purge valves was inserted into the ITS 5.5.16 program. This action consolidates CTS requirements for leak rate testing.

These changes are either editorial or are consistent with current license basis.

11 NUREG 5.5.8 – Incorporates TSTF-279.

12 NUREG 5.5.10 - The Secondary Water Chemistry Program is proposed to be revised to be consistent with the content of the current Operating License Condition 2.C(7) which does not include evaluation of the chemistry results for potential low pressure turbine disc stress corrosion cracking. An evaluation of the secondary water chemistry to maximize the turbine availability is currently accomplished under administrative controls (Procedure 1000.042) and is proposed to continue to be so controlled. This change is consistent with current license basis.

13 NUREG 5.5.11 - The Ventilation Filter Testing Program is proposed to be revised to be consistent with the content of the CTS for testing of HEPA and charcoal filters in safety related ventilation systems. Additionally, item e of the NUREG is not adopted since no heaters are provided in the design of these systems. These changes are consistent with current license basis.

14 NUREG SR 3.8.3.2 Bases – The discussion of the new fuel oil testing referencing "clear and bright" is revised. ANO fuel oil is supplied with added dye, which precludes appropriate "clear and bright" testing. In its place is supplied a reference to the currently utilized "water and sediment" testing of ASTM-D975.

15 Not used.

16 NUREG 5.5.9 & 5.5.13 – Incorporates TSTF-118.

17 NUREG Section 5.6 leads in with a statement about making submittals in accordance with 10 CFR 50.4. Many of the reports addressed are submitted in accordance with Part 20 and are not governed by 50.4. Since this statement is not part of CTS, it is removed from ITS.

18 NUREG 5.6.6 - The NUREG report for the reactor coolant system pressure and temperature limits is not adopted for the ITS. These limits will continue to be provided in the appropriate Limiting Conditions for Operation (LCO) (refer to ITS 3.4.3, "RCS Pressure and Temperature (P/T) Limits"). This change is consistent with current license basis.

19 NUREG 5.6.7 - Incorporates TSTF-037, Rev. 2.

20 NUREG 5.6.8 - The NUREG 5.6.8 reporting requirements related to post accident monitor inoperability are not proposed to be specifically identified in Section 5.0 of the ITS. A Special Report will continue to be required by the ACTIONS for the Post Accident Monitoring Instrumentation LCO (ITS 3.3.15 Required Action B.1), but details for content of the report will be provided only in the associated Bases for the Required Action. These controls are considered sufficient since they are not directly pertinent to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. Since the details of the report are also not necessary to fulfill the pertinent regulatory requirement, they are not mandated by 10 CFR 50.36, and they do not meet the criteria in 10 CFR 50.36, they can be appropriately retained in licensee controlled documents without a significant impact on safety. Retaining these requirements in controlled documents also provides adequate assurance that they will be maintained. Changes to the Bases are controlled by the proposed program in the Administrative Controls Section of the ITS. Additionally, this change is consistent with previously approved ITS for other EOI stations, i.e., Grand Gulf and River Bend.

21 Not used.

The example provided in the NUREG 5.7.1 of individuals qualified in radiation protection procedures "(e.g., Health Physics Technicians)" is not incorporated. This example is unnecessary and is considered likely to be interpreted as more limiting than intended since other individuals may be qualified in radiation protection procedures. This change is consistent with current license basis.

NUREG 5.7.1.c is revised to retain the CTS 6.11.1.c requirements by deleting reference to the Radiation Protection Manager. This change is consistent with current license basis.

22 NUREG 5.5.2 - The listing of systems which are considered Primary Coolant Sources Outside Containment (NUREG 5.5.2) is not incorporated. The systems to which the program is applied have been previously identified in response to NUREG-0578 item 2.1.6.a. The application is adequately controlled through the design modification process and application of 10 CFR 50.59. Therefore, the list of systems to which the program is applied is not included in the CTS and is proposed to continue to be administratively controlled. This change is consistent with current license basis.

23 NUREG 5.1.1 - The requirement for approval of each proposed test, experiment, or modification to systems or equipment that affect nuclear safety by the [Plant Superintendent] is not adopted. CTS Sections 6.5 and 6.8 were previously revised (Amendment No. 179 dated April 25, 1995) to eliminate this detail. Approval requirements for such procedures and modifications are delineated in the QAPM as discussed in the request for and approval of this recent amendment. This change is consistent with current license basis.

24 NUREG 5.2.2 - The requirements of 10 CFR 50.54(m)(2)(iii) and 50.54(k) adequately provide for this shift manning. These regulations, 50.54(m)(2)(iii), require "when a nuclear power unit is in an operational mode other than cold shutdown or refueling, as defined by the unit's Technical Specifications, each licensee shall have a person holding a senior operator license for the nuclear power unit in the control room at all times. In addition to this senior operator, for each fueled nuclear power unit, a licensed operator or senior operator shall be present at the controls at all times." Further, 10 CFR 50.54(k) requires "an operator or senior operator licensed pursuant to part 55 of this chapter shall be present at the controls at all times during the operation of the facility." The NUREG 5.2.2.b requirements will be met through compliance with these regulations and is not required to be re-iterated in the ITS. This change is consistent with TSTF-258, Rev 4, with one exception. 10 CFR 55.4 provides a definition for the phrase "actively performing the functions of an operator or senior operator," for the purposes of operator proficiency, as "an individual has a position on the shift crew that requires the individual to be licensed as defined in the facility's technical specifications,..." Since this 10 CFR 55.4 definition appears to require a facility to define those positions on the shift crew that are credited for gaining or maintaining the skills associated with performing licensed activities, a statement requiring adherence to the minimum shift composition of 10 CFR 50.54(m)(2)(i) has been added.

25 NUREG 5.5.10 - The specific identification of the ITS 5.5.10.c requirement to include "monitoring the discharge of the condensate pumps for evidence of condenser inleakage" is not adopted. The program can adequately control these details as demonstrated by the implementation of the current Operating License Condition 2.C(7). This change is consistent with current license basis.

26 NUREG 5.6.1 & 5.6.3 – Incorporates TSTF-152.

27 NUREG 5.2.2 - The ITS 5.2.2.g requirements for a Shift Technical Advisor (STA) on the unit staff are clarified to indicate that the STA is only required in MODES 1, 2, 3, and 4. This is consistent with CTS Table 6.2-1. This change is consistent with current license basis.

28 NUREG 5.2.2 - The introductory phrase "The unit staff organization shall include the following:" is omitted. This phrase provides no requirements or clarification, and implies that "the following" is intended to be a listing of required organizational elements. However, also included are general requirements for the staff, e.g., absence and overtime limitations, etc. Therefore, the introductory phrase is not appropriate.

29 NUREG 5.1.2 - The identification of the "Shift Supervisor" as responsible for the control room command function is not consistent with the current practice as ANO and is not adopted. The "command and control" functions are currently assigned to a Control Room Supervisor who is not limited to the area of the control room envelope. A Shift Superintendent is also provided who implements many of the functions of the NUREG "Shift Supervisor" and who typically remains in the control room. Further, the command structure is adequately controlled by procedures and "turnover" requirements in the ITS are unnecessary. These changes are consistent with the current license basis.

30 Not used.

31 NUREG 5.5.3 is modified to reflect CTS requirements for sampling of radioactive "iodine".

32 Not used.

33 NUREG 5.2.1 & 5.2.2 - A change similar to TSTF-065, Rev 1, and portions of TSTF-258, Rev 4, is included for the "specified corporate executive position" in ITS 5.2.1.c. Also, the ITS 5.2.2.g discussion is revised so that it does not imply that the STA and the "shift supervisor" must be different individuals. Option 1 of the Commission Policy Statement on Engineering Expertise on Shift is satisfied by assigning an individual with specified educational qualifications to each operating crew as one of the SROs (preferably the shift supervisor) required by 10 CFR 50.54(m)(2)(i) to provide the technical expertise on shift. However, the NUREG wording of "the STA shall provide ... support to the Shift Supervisor..." is considered to be easily misinterpreted to require separate individuals. Therefore, the wording is revised so that the STA function may be provided by either a separate individual or the individual who also fulfills another role in the shift command structure. This is consistent with CTS Table 6.2-1. This change is consistent with current license basis.

34 NUREG 5.5.12 – NUREG 5.5.12 is revised to match the CTS 3.25.1 and 4.29.1 which address only temporary outdoor liquid radwaste tanks. Additionally, an editorial clarification is made in the description of the limits for these tanks to match the Bases for the CTS, i.e., the radioactivity must be "less than the amount that would result in concentrations equal to the limits...," rather than an amount that would be "less than the amount that would result in concentrations less than the limits...." These revisions result in no functional differences in the requirements. Note that this entire Program is bracketed in NUREG-1430.

ITS DISCUSSION OF DIFFERENCES

35 NUREG 5.5.9 – The CTS 4.18 requirements for Steam Generator (SG) Tube Inspection are incorporated into the ITS as specified in the NUREG 5.5.9 Reviewer's Note. The following discuss the required changes:

1) Minor reformatting was necessary to establish consistency with the NUREG.
2) A note that reporting requirements were relocated from CTS 4.18.6 to ITS 5.6.7 was added for clarification.
3) TSTF-118 was incorporated which added a statement that ITS SR 3.0.2 is applicable to the SG Tube Surveillance Program inspection Frequencies.

These changes were in accordance with NUREG guidance, TSTF-118 or were editorial with no change in license basis requirements.

36 NUREG 5.6.10 – The CTS 4.18.6 and CTS 6.12.5.d requirements for the Steam Generator Tube Surveillance Report were incorporated into ITS 5.6.7 as directed the NUREG 5.6.10 Reviewer's Note. Minor reformatting was necessary and cross-reference numbers were changed to accurately reflect the ITS location of the requirements. No relaxation of requirements exists as a result of this change. This change is consistent with NUREG-1430 direction and current license basis.

37 NUREG 5.5.6, "Pre-Stressed Concrete Containment Tendon Surveillance Program," is not incorporated in the ANO-1 ITS. The license amendment #199 revised the reactor building structural integrity requirements to relocate this program from the ANO-1 CTS. NUREG 5.6.9 is revised to reflect the Reactor Building Inspection Report consistent with CTS 6.12.4.

38 NUREG 5.5.13 – Incorporates TSTF-106, Rev 1.

39 NUREG 5.5.15 – Incorporates TSTF-273, Rev 2.

40 NUREG 5.5.2 – Incorporates TSTF-299.

ANO-349

41 NUREG 5.5.4 – Incorporates TSTF-308, Rev 1.

42 NUREG 5.5.4, 5.6.4 & 5.7 – Incorporates TSTF-258, Rev 4. Two editorial changes are reflected in the markup of the Section 5.7 Insert (which is from TSTF-258, Rev 4). The addition of a comma in 5.7.1.b clarifies that the added detail applies to the "equivalent" means. Paragraph 5.7.2.d.3(ii) ends with phrasing that is editorially different than the same requirement found in paragraph 5.7.1.d.4(ii). The ending phrasing used in 5.7.1 is utilized in 5.7.2.

ANO-353

43. NUREG 5.5.14 - Incorporates TSTF-364.

CTS

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

ANO-1 plant manager

5.1.1 The ~~[Plant Superintendent]~~ shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence. (1) 6.1.1

~~The [Plant Superintendent] or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.~~ (23)

5.1.2 ~~The [Shift Supervisor (SS)] shall be responsible for the control room command function. During any absence of the [SS] from the control room~~ while the unit is in MODE 1, 2, 3, or 4, ~~an~~ An individual with an active Senior Reactor Operator (SRO) license shall be designated ~~to assume~~ as responsible for the control room command function. ~~During any absence of the [SS] from the control room while the unit is in MODE 5 or 6, an~~ With the unit not in MODE 1, 2, 3 or 4, an individual with an active SRO license or Reactor Operator license shall be designated ~~to assume~~ as responsible for the control room command function. 6.1.2 (29)

CTS

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations 6.2.1

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power ~~plant~~. unit edit

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the ~~[FSAR]~~; 6.2.1.a

INSERT 5.0-2A

Safety Analysis Report (SAR) 1

b. The ~~[Plant Superintendent]~~ shall be responsible for overall safe operation of the ~~plant~~ and shall have control over those onsite activities necessary for safe operation and maintenance of the ~~plant~~; unit 6.2.1.b edit

ANO-1 plant manager

c. ~~The~~ A ~~[a]~~ specified corporate executive ~~position~~ shall have corporate responsibility for overall ~~plant~~ nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the ~~plant~~ to ensure nuclear safety~~; and~~. unit 6.2.1.c edit 33

INSERT 5.0-2B

d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures. 6.2.1.d

5.2.2 Unit Staff

~~The~~ unit staff ~~organization~~ ~~shall~~ include ~~the~~ ~~following~~: 28

a. A non-licensed operator shall be ~~assigned to each reactor containing~~ fuel, and an additional non-licensed operator 6.2.2 Table 6.2-1 2

on site when

is in the reactor,

(continued)

<INSERT 5.0-2A>

, including the unit specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications,

<INSERT 5.0-2B>

The specified corporate executive shall be identified in the SAR; and

CTS

5.2 Organization

5.2.2 Unit Staff (continued)

on site when the

shall be ~~assigned for each control room from which a~~ reactor is operating in MODES 1, 2, 3, or 4.

~~Two unit sites with both units shutdown or defueled require a total of three non-licensed operators for the two units.~~

2

<INSERT 5.0-3A>

b. ~~At least one licensed Reactor Operator (RO) shall be present in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Reactor Operator (SRO) shall be present in the control room.~~

24

2

6.2.2 Table 6.2-1

c. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i), and 5.2.2.a and 5.2.2.g for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

for one unit, one control room,

1

6.2.2 Table 6.2-1

d. ~~A [Health Physics Technician]~~ shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

An individual qualified in radiation protection procedures

~~e. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety related functions (e.g., licensed SROs, licensed ROs, health physicists, auxiliary operators, and key maintenance personnel).~~

~~Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work an [8 or 12] hour day, nominal 40 hour week while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modification, on a temporary basis the following guidelines shall be followed:~~

(continued)

<INSERT 5.0-3A>

The minimum shift crew composition for licensed operators shall meet the minimum staffing requirements of 10 CFR 50.54(m)(2)(i) for one unit, one control room.

CTS

5.2 Organization

5.2.2 Unit Staff (continued)

1. ~~An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time;~~

2. ~~An individual should not be permitted to work more than 16 hours in any 24 hour period, nor more than 24 hours in any 48 hour period, nor more than 72 hours in any 7 day period, all excluding shift turnover time;~~

3. ~~A break of at least 8 hours should be allowed between work periods, including shift turnover time;~~

4. ~~Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.~~

~~Any deviation from the above guidelines shall be authorized in advance by the [Plant Superintendent] or his designee, in accordance with approved administrative procedures, or by higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation.~~

~~Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the [Plant Superintendent] or his designee to ensure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.~~

~~OR~~

e. The amount of overtime worked by unit staff members performing safety related functions shall be limited and controlled in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12). 6.2.2.1

f. The ~~[O~~perations ~~M~~anager or ~~A~~ssistant ~~O~~perations ~~M~~anager~~]~~ shall hold an SRO license. 6.2.2 1

In MODES 1, 2, 3 or 4, an individual

for the unit operations shift crew

This individual

g. ~~The Shift Technical Advisor (STA)~~ shall provide advisory technical support ~~to the Shift Supervisor (SS)~~ in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. ~~In addition, the STA~~ shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. 27 33 6.3.1 Table 6.2-1

CTS

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

~~[Reviewer's Note: Minimum qualifications for members of the unit staff shall be specified by use of an overall qualification statement referencing an ANSI Standard acceptable to the NRC staff or by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of unique organizational structures.]~~

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ~~[Regulatory Guide 1.8, Revision 2, 1987, or more recent revisions, or ANSI Standard acceptable to the NRC staff]. The staff not covered by [Regulatory Guide 1.8]~~ shall meet or exceed the minimum qualifications of ~~[Regulations,]~~ Regulatory Guide~~s, or ANSI Standards acceptable to NRC staff]~~. 6.3.1

1.8, September 1975.

6

ANSI ANS3.1-1978 for comparable positions except for the designated radiation protection manager, who

CTS

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978; 6.8.1.a,b,c,g N/A

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in ~~(~~Section 7.1 of Generic Letter 82-33~~)~~; N/A 3

~~c. Quality assurance for effluent and environmental monitoring;~~ 5

c ~~d~~. Fire Protection Program implementation; and 6.8.1.f

d ~~e~~. All programs specified in Specification 5.5. 6.8.1.h N/A

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

CTS

The following programs shall be established, implemented, and maintained. 6.14

5.5.1 Offsite Dose Calculation Manual (ODCM)

~~a.~~ The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and edit

~~b.~~ The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports ~~required by Specification [5.6.2] and Specification [5.6.3]~~. edit 7

Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

2. a determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after the approval of the ~~[Plant Superintendent]~~ ANO General manager; and 1

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the

(continued)

CTS

5.5 Programs and Manuals

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Primary Coolant Sources Outside Containment LC 2.C(5)

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. ~~The systems include [Low Pressure Injection, Reactor Building Spray, Makeup and Purification, and Hydrogen Recombiner].~~ 22 The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at ~~refueling cycle intervals or less~~ least once per 18 months. The provisions of SR 3.0.2 are applicable. 40

5.5.3 Post Accident Sampling LC 2.C(6) 6.8.1.i

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive ~~gases~~ iodine and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following: 31

a. Training of personnel;

b. Procedures for sampling and analysis; and

c. Provisions for maintenance of sampling and analysis equipment.

5.5.4 Radioactive Effluent Controls Program 6.8.5

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to

(continued)

5.5 Programs and Manuals CTS

5.5.4 Radioactive Effluent Controls Program (continued) 6.8.5

be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM; 6.8.5.a

ten times the consentration values in

to 10 CFR 20.1001 – 20.2402

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ~~10 CFR 20~~, Appendix B, Table 2, Column ~~2~~; 6.8.5.b 42

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM; 6.8.5.c

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I; 6.8.5.d

ANO-349

<INSERT 5.0-9A>

e. ~~Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;~~ 6.8.5.e 41

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I; 6.8.5.f

from the site at or

<INSERT 5.0-9B>

g. Limitations on the dose rate resulting ~~from radioactive~~ material released in gaseous effluents to areas beyond the site boundary ~~conforming to the dose associated with 10 CFR 20, Appendix B, Table 2, Column 1~~; 6.8.5.g 42

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; 6.8.5.h

(continued)

<INSERT 5.0-9A>

ANO-349

Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days.

<INSERT 5.0-9B>

... shall be in accordance with the following:

1. For noble gases: a dose rate ≤ 500 mrem/yr to the whole body and a dose rate ≤ 3000 mrem/yr to the skin, and

2. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrem/yr to any organ;

CTS

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and 6.8.5 i

j. Limitations on the ~~annual dose or dose commitment~~ to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190. 6.8.5 j (42)

<INSERT 5.0-10A>

5.5.5 ~~Component Cyclic or Transient Limit~~ (Not used.) (8)

~~This program provides controls to track the FSAR, Section [ ], cyclic and transient occurrences to ensure that components are maintained within the design limits.~~

5.5.6 ~~Pre-Stressed Concrete Containment Tendon Surveillance Program~~ (Not used.) (37)

~~This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with [Regulatory Guide 1.35, Revision 3, 1989].~~

~~The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.~~

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel ~~per the recommendation of Regulatory position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975.~~ (9)

<INSERT 5.0-10B>

(continued)

<INSERT 5.0-10A>

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

<INSERT 5.0-10B>

. Surface and volumetric examination of the reactor coolant pump flywheels will be conducted coincident with refueling or maintenance shutdowns such that during 10 year intervals all four reactor coolant pump flywheels will be examined. Such examinations will be performed to the extent possible through the access ports, i.e., those areas of the flywheel accessible without motor disassembly. The surface and volumetric examination may be accomplished by Acoustic Emission Examination as an initial examination method. Should the results of the Acoustic Emission Examination indicate that additional examination is necessary to ensure the structural integrity of the flywheel, then other appropriate NDE methods will be performed on the area of concern.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Reactor Coolant Pump Flywheel Inspection Program inspection frequencies.

CTS

5.5 Programs and Manuals (continued)

5.5.8 Inservice Testing Program 4.0.5

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components, ~~including applicable supports.~~ The program shall include the following: 11

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

edit

ASME ~~Boiler and Pressure Vessel~~ Code ~~and applicable Addenda~~ terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

5.5.9 Steam Generator (SG) Tube Surveillance Program 4.18

~~Reviewer's Note: The Licensees current licensing basis steam generator tube surveillance requirements shall be relocated from the LCO and included here. An appropriate administrative controls program format should be used.~~

<INSERT 5.0-11A> 35 16

(continued)

<Insert 5.0-11A (SG Tube Inspection Program)>

This program provides controls to ensure integrity of the steam generator tubing through a defined inservice surveillance program, and to minimize exposure of personnel to radiation during performance of the surveillance program.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the SG Tube Surveillance Program inspection frequencies. 16

a. The first steam generator tubing inspection performed in accordance with 5.5.9.b and 5.5.9.c.1 shall be considered as constituting the baseline condition for subsequent inspections.

b. Examination Methods:

   1. Inservice inspection of steam generator tubing shall include non-destructive examination by eddy-current testing or other equivalent techniques. The inspection equipment shall provide a sensitivity that will detect defects with a penetration of 20 percent or more of the minimum allowable as-manufactured tube wall thickness except for a sleeved tube at the lower sleeve end.

   2. For examination of the sleeved steam generator tubing at the lower sleeve end, the indications will be compared to those obtained during the baseline sleeved tube inspection. Significant deviations between these indications will be considered sufficient evidence to warrant designation as a degraded tube. Direct quantification of the 40 percent through-wall plugging limit is available with eddy-current testing.

c. Selection and Testing

The steam generator sample size is specified in Table 5.5.9-1. The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.5.9-2. The inservice inspection of steam generator tubes shall be performed at the frequencies as specified in 5.5.9.d and the inspected tubes shall be verified acceptable per the acceptance criteria of 5.5.9.e. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in both steam generators; the tubes selected for these inspections shall be selected on a random basis except:

   1. The first sample inspection during each inservice inspection (subsequent to the baseline inspection) of each steam generator shall include:

      i. All nonplugged tubes that previously had detectable wall penetrations (>20%), except tubes in which the wall penetration has been spanned by a sleeve, and

      ii. At least 50% of the tubes inspected shall be in those areas where experience has indicated potential problems, except where specific groups are inspected per 5.5.9.c.1.iii.

      A tube inspection (pursuant to 5.5.9.e.1.ix) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

iii. Tubes in the following groups may be excluded from the first random sample if all tubes in a group in both steam generators are inspected. The inspection may be concentrated on those portions of the tubes where imperfections were previously found. No credit will be taken for these tubes in meeting minimum sample size requirements. Where only a portion of the tube is inspected, the remainder of the tube will be subjected to the random inspection.

(1) Group A-1: Tubes within one, two or three rows of the open inspection lane.

(2) Group A-2: Unplugged tubes with sleeves installed.

(3) Group A-3: Tubes in the wedge-shaped group on either side of the lane region (Group A-1) as defined by Figure 5.5.9-1.

iv. Tubes with axially-oriented tube end cracks (TEC) which have been left inservice for the previous cycle shall be inspected with a rotating coil eddy current technique in the area of the TEC and characterized in accordance with topical report BAW-2346P, Rev.0, during all subsequent SG inspection intervals pursuant to 4.18.4. The results of this examination may be excluded from the first random sample. Tubes with axial TECs identified during previous inspections, which meet the criteria to remain in service, will not be included when calculating the inspection category of the OTSG.

v. Implementation of the upper tubesheet ODIGA alternate repair criteria requires a 100% bobbin coil inspection of the non-plugged and non-sleeved tubes, spanning the defined region of the upper tubesheet, during all subsequent SG inspection intervals pursuant to 4.18.4. Tubes with ODIGA identified during previous inspections, which meet the criteria to remain in service, will not be included when calculating the inspection category for the OTSG. The defined region begins one inch above the upper tubesheet secondary face and ends at the nearest tube roll transition. ODIGA indications detected by the bobbin coil probe shall be characterized using rotating coil probes in accordance with ANO Engineering Report No. 00-R-1005-01.

ANO-354

2. All tubes which have been repaired using the reroll process will have the new roll area inspected during the inservice inspection.

3. The second and third sample inspections during each inservice inspection as required by Table 5.5.9-2 may be less than a full tube inspection by concentrating the inspection on those areas of the tube sheet array and on those portions of the tubes where tubes with imperfections were previously found.

4. The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected, are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

NOTES:

(1) In all inspections, previously degraded tubes whose degradations have not been spanned by a sleeve must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

(2) Where special inspections are performed pursuant to 5.5.9.c.1.iii, defective or degraded tubes found as a result of the inspection shall be included in determining the Inspection Results Category for that special inspection but need not be included in determining the Inspection Results Category for the general steam generator inspection.

(3) Where special inspections are performed pursuant to 5.5.9.c.2, defective or degraded tube indications found in the new roll area as a result of the inspection and any indications found above the new roll area, are not included in the determination for the inspection results category of a general steam generator inspection.

d. The above-required inservice inspections of steam generator tubes shall be performed at the following frequencies:

1. The baseline inspection shall be performed during the first refueling shutdown. Subsequent inservice inspections shall be performed at intervals of not less than 10 nor more than 24 calendar months after the previous inspection. If the results of two consecutive inspections for a given group* of tubes following service under all volatile treatment (AVT) conditions fall into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval for that group may be extended to a maximum of 40 months.

2. If the results of the inservice inspection of a steam generator performed in accordance with Table 5.5.9-2 at 40-month intervals for a given group* of tubes fall in Category C-3, subsequent inservice inspections shall be performed at intervals of not less than 10 nor more than 20 calendar months after the previous inspection. The increase in inspection frequency shall apply until a subsequent inspection meets the conditions specified in 5.5.9.d.1 and the interval can be extended to 40 months.

3. Additional unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 5.5.9-2 during the shutdown subsequent to any of the following conditions:

   i. Primary-to-secondary leakage in excess of the limits of Specification 3.4.13 (inservice inspection not required if leaks originate from tube-to-tubesheet welds). If the leaking tube is from either Group A-1 or A-3 as defined in Specification 5.5.9.c.1.iii, all of the tubes in the affected group in this steam generator may be inspected in lieu of the first sample inspection specified in Table 5.5.9-2. If the degradation mechanism which caused the leak is limited to a specific portion of the tube length, the inspection per this paragraph may be limited to the affected portion of the tube length. If the results of this inspection fall into the C-3 category, all of the tubes in the same group in the other steam generator will also be similarly inspected.
   If the leaking tube has been repaired by the reroll process and is leaking in the new roll area, all of the tubes in the steam generator that have been repaired by the reroll process will have the new roll area inspected. If the results of this inspection fall into the C-3 category, all of the tubes with rerolled areas in the other steam generator will also be similarly inspected. This inspection will be in lieu of the first sample inspection specified in Table 5.5.9-2.

---

*A group of tubes means:
(a) All tubes inspected pursuant to 5.5.9.c.1.iii, or
(b) All tubes in a steam generator less those inspected pursuant to 5.5.9.c.1.iii.

ii. A seismic occurrence greater than the Operating Basis Earthquake,

iii. A loss-of-coolant accident requiring actuation of the engineered safeguards, or

iv. A main steam line or feedwater line break.

e. Acceptance Criteria:

1. Terms as used in this program:

i. Tubing or Tube means that portion of the tube or sleeve which forms the primary system to secondary system pressure boundary.

ii. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

iii. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either the inside or outside of a tube.

ANO-345

iv. Degraded Tube means a tube containing imperfections ≥ 20% of the nominal wall thickness caused by degradation, except where all degradation has been spanned by the installation of a sleeve or repaired by a rerolled joint.

ANO-345

The reroll repair process will be used to repair tubes with defects in the upper and lower tubesheet areas as described in topical report, BAW-2303P, Revision 4.

v. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

vi. Defect means an imperfection of such severity that it exceeds the plugging limit except where the imperfection has been spanned by the installation of a sleeve. A tube containing a defect in its pressure boundary is defective.

ANO-354

vii. Plugging Limit means the imperfection depth at or beyond 40% of the nominal tube wall thickness for which the tube shall be sleeved, rerolled, or removed from service because it may become unserviceable prior to the next inspection. This does not apply to ODIGA indications within the defined region of the upper tubesheet. These indications shall be assessed for continued plant operation in accordance with ANO Engineering Report No. 00-R-1005-01, Rev. 1.

Axially-oriented TEC indications in the tube that do not extend beyond the adjacent cladding portion of the tube sheet into the carbon steel portion are not included in this definition. These indications shall be assessed for continued plant operation in accordance with topical report BAW-2346P, Rev. 0.

viii. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 5.5.9.d.3.

ANO-345

ix. Tube Inspection means an inspection of the steam generator tube from the point of entry completely to the point of exit. For tubes that have been repaired by the reroll process within the tubesheets, that portion of the tube outboard of the new roll can be excluded from future periodic inspection requirements because it is no longer part of the pressure boundary once the repair roll is installed.

2. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug, reroll, or sleeve all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 5.5.9-2.

TABLE 5.5.9-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No
No. of Steam Generators per Unit	Two
First Inservice Inspection	Two
Second & Subsequent Inservice Inspection	One[1]

Table Notation:

[1] The inservice inspection may be limited to one steam generator on alternating schedule encompassing 3N% of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

<u>TABLE 5.5.9-2</u>

STEAM GENERATOR TUBE INSPECTION [2,3]

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.[1]	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug, reroll, or sleeve defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug, reroll, or sleeve defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug, reroll, or sleeve defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G. plug, reroll, or sleeve defective tubes and inspect 2S tubes in other S.G.	Other S.G. is C-1	None	N/A	N/A
			Other S.G. is C-2	Perform action for C-2 result of second sample	N/A	N/A
			Other S.G. is C-3	Inspect all tubes in each S.G. and plug, reroll, or sleeve defective tubes.	N/A	N/A

NOTES:

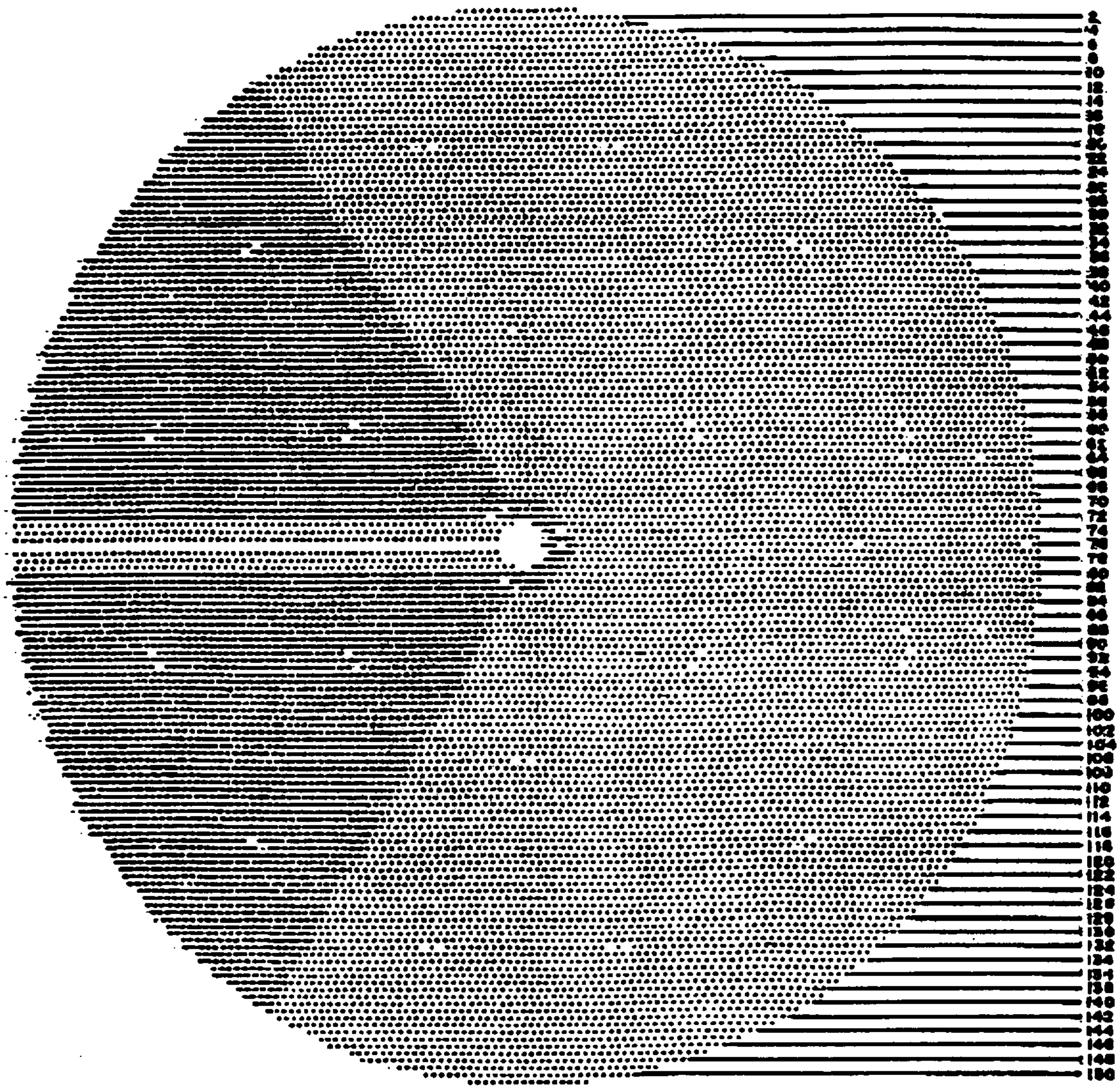
[1] $S = \frac{3N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

[2] For tubes inspected pursuant to 5.5.9.c.1.iii: No action is required for C-1 results. For C-2 results in one or both steam generators plug, reroll, or sleeve defective tubes. For C-3 results in one or both steam generators, plug, reroll, or sleeve defective tubes and provide a report to NRC pursuant to 5.6.7.

[3] No more than ten thousand (10,000) sleeves may be installed in both ANO-1 steam generators combined.

FIGURE 5.5.9-1

Upper Tube Sheet View of Wedge Shaped Group (Group A-3) per 5.5.9.c.1.iii



DESCRIPTION	TUBE COUNT
Group A-1: Lane region tubes as defined in 5.5.9.c.1.iii(1)	382
Group A-3: Wedge shaped group depicted by darkened region of figure	4880

CTS

5.5 Programs and Manuals (continued)

5.5.10 Secondary Water Chemistry LC 2.C(7)

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation, ~~and low pressure turbine disc stress corrosion cracking~~. The program shall include: 12

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points~~, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage~~; 25

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events ~~which is~~ required to initiate corrective action. edit

5.5.11 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered ~~Safety Feature (ESF) Filter~~ Safeguards (ES) ventilation systems filters at the frequencies specified in ~~[Regulatory Guide], and in accordance with~~ Regulatory Guide 1.52, Revision 2. ~~ASME N510-1989, and AG-1].~~ 4.10.2 4.11 4.17

< INSERT 5.0-12A >

~~a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < [0.05%] when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified below [± 10%].~~ 13

ESF Ventilation System	Flowrate
~~[ ]~~	~~[ ]~~

(continued)

CTS

<INSERT 5.0-12A>

The VFTP is applicable to the Penetration Room Ventilation System (PRVS), the Fuel Handling Area Ventilation System (FHAVS), and the Control Room Emergency Ventilation System (CREVS).

a. Demonstrate that an inplace cold DOP test of the high efficiency particulate air (HEPA) filters shows:

3.13.1.a&c
3.15.1.a&c

ANO-348 1. ≥ 99% DOP removal for the PRVS when tested at the system design flowrate of 1800 cfm ± 10% and the FHAVS when tested at the system design flowrate of 39000 cfm ± 10%; and

4.10.2.b.1
b.3
e

ANO-348 2. ≥ 99.95% DOP removal for the CREVS when tested in accordance with Regulatory Guide 1.52, Revision 2, at the system design flowrate of 2000 cfm ± 10%.

b. Demonstrate that an inplace halogenated hydrocarbon test of the charcoal adsorbers shows:

3.13.1.a&c
3.15.1.a&c

ANO-348 1. ≥ 99% halogenated hydrocarbon removal for the PRVS when tested at the system design flowrate of 1800 cfm ± 10% and FHAVS when tested at the system design flowrate of 39000 cfm ± 10%; and

4.10.2.b.1
b.3
f

2. ≥ 99.95% halogenated hydrocarbon removal for the CREVS when tested in accordance with Regulatory Guide 1.52, Revision 2, at the system design flowrate of 2000 cfm ± 10%.

c. Demonstrate that a laboratory test of a sample of the charcoal adsorber meets the laboratory testing criteria of ASTM D3803-1989 when tested at 30°C and 95% relative humidity for a methyl iodide penetration of: 3.13.1.b

ANO-357 1. < 5% for the PRVS; 3.15.1.b

2. < 5% for the FHAVS; and. 4.10.2.b.2
c

3. when obtained as described in Regulatory Guide 1.52, Revision 2, for CREVS:

i. ≤ 2.5% for 2 inch charcoal adsorber beds; and

ii. ≤ 0.5% for 4 inch charcoal adsorber beds

3.13.1.c&d
3.15.1.c&d
4.10.2.b.3
d.1
4.11.1
4.17.1

d. Demonstrate, for the PRVS, FHAVS, and CREVS, that the pressure drop across the combined HEPA filters, other filters in the system, and the charcoal adsorbers is < 6 inches of water when tested at the following system design flowrates ± 10%:

ANO-348

PRVS	1800 cfm
FHAVS	39000 cfm
CREVS	2000 cfm

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < [0.5]% when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified below [± 10%].

ESF Ventilation System	Flowrate
[ ]	[ ]

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in [Regulatory Guide 1.52, Revision 2], shows the methyl iodide penetration less than the value specified below when tested in accordance with [ASTM D3803-1989] at a temperature of ≤ [30°C] and greater than or equal to the relative humidity specified below.

ESF Ventilation System	Penetration	RH
[ ]	[ ]	[ ]

[Reviewer's Note: Allowable penetration = [100% - methyl iodide efficiency for charcoal credited in staff safety evaluation]/ (safety factor).

Safety factor = [5] for systems with heaters.
= [7] for systems without heaters.]

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with [Regulatory Guide 1.52,

13

(continued)

CTS

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

~~Revision 2, and ASME N510-1989] at the system flowrate specified below [± 10%].~~ 13

ESF Ventilation System	Delta P	Flowrate
[ ]	[ ]	[ ]

~~e. [Demonstrate that the heaters for each of the ESF systems dissipate the value specified below [± 10%] when tested in accordance with [ASME N510-1989].]~~

ESF Ventilation System	Wattage
[ ]	[ ]

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

[5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program] NA

This program provides controls for potentially explosive gas mixtures contained in the [Waste Gas ~~Holdup~~ System], [the quantity of radioactivity contained in gas storage tanks ~~or fed into the offgas treatment system~~, and the quantity of radioactivity contained in unprotected temporary outdoor liquid storage tanks]. The gaseous radioactivity quantities shall be determined following the methodology in [Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure"]. The liquid radwaste quantities shall be determined in accordance with ~~[Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures"]~~ the ODCM. edit edit

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the [Waste Gas ~~Holdup~~ System] and a surveillance program to ensure the limits are maintained. Such limits shall be 3.24 4.28

(continued)

5.5 Programs and Manuals

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank and fed into the offgas treatment system is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; 3.25.2 4.29.2

c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the [Liquid Radwaste Treatment System] is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents. 3.25.1 34 4.29.1

temporary

1) 2) equal to

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies. NA

5.5.13 Diesel Fuel Oil Testing Program 4.6.1.4.e

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

1. an API gravity or an absolute specific gravity within limits,

(continued)

d. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program surveillance Frequencies 16

5.5 Programs and Manuals

4.6.1.4.e

5.5.13 Diesel Fuel Oil Testing Program (continued)

2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

3. ~~a clear and bright appearance with proper color~~; water and sediment within limits 14

b. ~~Other properties for ASTM 2D fuel oil are within limits~~ within ~~30~~ 31 days following ~~sampling and~~ addition of the new fuel oil to storage tanks; ~~and~~ 38

INSERT 5.0-16 A

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days ~~in accordance with~~ based on ASTM D-2276, Method A-2 or A-3; and 38

5.5.14 Technical Specifications (TS) Bases Control Program NA

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

1. A change in the TS incorporated in the license; or

2. ~~A change to the updated FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.~~ 43

A change to the updated SAR or Bases that requires NRC approval pursuant to 10 CFR 50.59

ANO-353

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the ~~F~~SAR. edit

~~d~~. Proposed changes that do not ~~meet the~~ this criteria ~~of 5.5.14b above~~ shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e). edit

(continued)

<INSERT 5.0-16A>

..., verify that the properties of the new fuel oil, other than those addressed in a. above, are within limits for ASTM 2D fuel oil.

CTS

5.5 Programs and Manuals (continued)

N/A

5.5.15 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

INSERT 5.0 - 17A

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

39

10

INSERT 5.0-17B

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

6.8.4
4.26.2
3.7.3A.1

<INSERT 5.0-17A>

... and assuming no concurrent loss of offsite power or loss of onsite diesel generator(s), ...

<INSERT 5.0-17B>

5.5.16 Reactor Building Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the reactor building as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

In addition, the reactor building purge supply and exhaust isolation valves shall be leakage rate tested once prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days.

The peak calculated reactor building internal pressure for the design basis loss of coolant accident, P_a, is 54 psig.

The maximum allowable reactor building leakage rate, L_a, shall be 0.20% of reactor building air weight per day at P_a.

Reactor building leakage rate acceptance criteria is $\leq 1.0\ L_a$. During the first unit startup following each test performed in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and Type C tests and $< 0.75\ L_a$ for Type A tests.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Reactor Building Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Reactor Building Leakage Rate Testing Program.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

CTS

~~The following reports shall be submitted in accordance with 10 CFR 50.4.~~ 17

6.12.2.2

5.6.1 Occupational Radiation Exposure Report

AND

-----------------------------NOTE-----------------------------

A single submittal may be made for ~~a multiple unit station~~. The submittal should combine sections common to ~~all~~ both units ~~at the station.~~

edit

INSERT 5.0-18A

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures > 100 mrem/yr and their associated man rem exposure according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance [describe maintenance], waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions. The report shall be submitted by April 30 of each year. [The initial report shall be submitted by April 30 of the year following the initial criticality.] 26

6.12.2.5

5.6.2 Annual Radiological Environmental Operating Report

AND

-----------------------------NOTE-----------------------------

A single submittal may be made for ~~a multiple unit station~~. The submittal should combine sections common to ~~all~~ both units ~~at the station.~~

edit

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual

(continued)

<INSERT 5.0-18A>

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent > 100 mrems and the associated collective deep dose equivalent (reported in person – rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket ionization chamber, thermoluminescence dosimeter (TLD), electronic dosimeter, or film badge measurements. Small exposures totaling < 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year.

CTS

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued) 6.12.2.5

(ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements~~. [in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979]. [The report shall identify the TLD results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result.]~~ In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible. 6.12.2.5

5.6.3 Radioactive Effluent Release Report 6.12.2.6

AND

both

-----NOTE-----

A single submittal may be made for ~~a multiple unit station~~ both. The submittal ~~should~~ shall combine sections common to ~~all~~ both units ~~at the station; however, for units with separate radwaste systems,~~ AND the submittal shall specify the releases of radioactive material from each unit.

edit

26

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1. Part

(continued)

5.6 Reporting Requirements (continued)

CTS

5.6.4 Monthly Operating Reports

6.12.2.3

Routine reports of operating statistics and shutdown experience ~~[, including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves,]~~ shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report. 42

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

6.12.3

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

<INSERT 5.0-20A> ~~The individual specifications that address core operating limits must be referenced here.~~

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following document~~s~~: edit

<INSERT 5.0-20B> ~~Identify the Topical Report(s) by number, title, date, and NRC staff approval document, or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date.~~

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

~~5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)~~ 18

~~a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic~~

(continued)

<INSERT 5.0-20A>

2.1.1	Variable Low RCS Pressure – Temperature Protective Limits
3.1.1	SHUTDOWN MARGIN
3.1.8	PHYSICS TESTS Exceptions – MODE 1
3.1.9	PHYSICS TEST Exceptions - MODE 2
3.2.1	Regulating Rod Insertion Limits
3.2.2	AXIAL POWER SHAPING RODS (APSR) Insertion Limits
3.2.3	AXIAL POWER IMBALANCE Operating Limits
3.2.4	QUADRANT POWER TILT (QPT)
3.2.5	Power Peaking
3.3.1	Reactor Protection System (RPS) Instrumentation
3.4.1	RCS Pressure, Temperature, and Flow DNB Limits
3.4.4	RCS Loops – MODES 1 and 2
3.9.1	Boron Concentration

<INSERT 5.0-20B>

Babcock & Wilcox Topical Report BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses" (the approved revision at the time the reload analyses are performed). The approved revision number shall be identified in the COLR.

CTS

5.6 Reporting Requirements

18

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following: [The individual specifications that address RCS pressure and temperature limits must be referenced here.]

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents: [Identify the NRC staff approval document by date.]

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

[Reviewer's Notes: The methodology for the calculation of the P-T limits for NRC approval should include the following provisions:

1. The methodology shall describe how the neutron fluence is calculated (reference new Regulatory Guide when issued).

2. The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.

3. Low Temperature Overpressure Protection (LTOP) System lift setting limits for the Power Operated Relief Valves (PORVs), developed using NRC-approved methodologies may be included in the PTLR.

4. The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for radiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2.

5. The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG-0800 Standard Review Plan 5.3.2, Pressure-Temperature Limits.]

(continued)

CTS

5.6 Reporting Requirements

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued) 18

[6. The minimum temperature requirements of Appendix G to 10 CFR Part 50 shall be incorporated into the pressure and temperature limit curves.

7. Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature (RT_{NDT}) to the predicted increase in RT_{NDT}; where the predicted increase in RT_{NDT} is based on the mean shift in RT_{NDT} plus the two standard deviation value ($2\sigma_\Delta$) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value (increase in RT_{NDT} + $2\sigma_\Delta$), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology.]

5.6.7 EDG Failures Report 19

[If an individual emergency diesel generator (EDG) experiences four or more valid failures in the last 25 demands, these failures and any nonvalid failures experienced by that EDG in that time period shall be reported within 30 days. Reports on EDG failures shall include the information recommended in Regulatory Guide 1.9, Revision 3, Regulatory Position C.5, or existing Regulatory Guide 1.108 reporting requirement.]

5.6.8 PAM Report 20

When a report is required by Condition B or G of LCO 3.3.[17], "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

(continued)

5.6 Reporting Requirements (continued)

CTS

5.6.9 ~~[Tendon Surveillance Report~~ 6.12.4

~~Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.]~~ 37

<INSERT 5.0-23A>

5.6.7 ~~5.6.10~~ [Steam Generator Tube ~~Inspection~~ Surveillance Report] (S) 4.18.6 6.12.5.d

~~Reviewer's Note: Reports required by the Licensee's current licensing basis regarding steam generator tube surveillance requirements shall be included here. An appropriate administrative controls format should be used.~~ 36

<INSERT 5.0-23B>

~~[Reviewer's Note: These reports may be required covering inspection, test, and maintenance activities. These reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.]~~

<INSERT 5.0-23A>

5.6.6 Reactor Building Inspection Report

Any degradation exceeding the acceptance criteria of the containment structure detected during the tests required by the Containment Inspection Program shall undergo an engineering evaluation within 60 days of the completion of the inspection surveillance. The results of the engineering evaluation shall be reported to the NRC within an additional 30 days of the time the evaluation is completed. The report shall include the cause of the condition that does not meet the acceptance criteria, the applicability of the conditions to the other unit, the acceptability of the concrete containment without repair of the item, whether or not repair or replacement is required and, if required, the extent, method, and completion date of necessary repairs, and the extent, nature, and frequency of additional examinations.

<INSERT 5.0-23B>

a. Following each inservice inspection of steam generator tubes, the complete results of the inspection shall be reported to the NRC. This report, to be submitted within 90 days of inspection completion, shall include:

   1. Number and extent of tubes inspected;

   2. Location and percent of wall-thickness penetration for each indication of an imperfection;

   3. Identification of tubes plugged and tubes sleeved;

   4. Number of tubes repaired by rerolling and number of indications detected in the new roll area of the repaired tubes;

   5. Summary of the condition monitoring and operational assessment results when applying TEC alternate repair criteria; and

   6. Summary of the condition monitoring and the operational assessment results (including growth) when applying the upper tubesheet ODIGA alternate repair criteria.

b. In addition, the Commission shall be notified of the results of steam generator tube inspections which fall into Category C-3 as denoted in Table 5.5.9-2 prior to resumption of plant operation. The written report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

INSERT 5.0-24A

42

5.0 ADMINISTRATIVE CONTROLS

[5.7 High Radiation Area]

5.7.1 Pursuant to 10 CFR 20, paragraph 20.1601(c), in lieu of the requirements of 10 CFR 20.1601, each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but < 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., [Health Physics Technicians]) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates ≤ 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the [Radiation Protection Manager] in the RWP.

5.7.2 In addition to the requirements of Specification 5.7.1, areas with radiation levels ≥ 1000 mrem/hr shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the Shift Foreman on duty or health physics supervision. Doors shall remain locked except during periods of access by personnel

(continued)

<INSERT 5.0-24A>

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP), or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area; or

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designee.

2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

42

[5.7 High Radiation Area]

5.7.2 (continued)

under an approved RWP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

5.7.3 For individual high radiation areas with radiation levels of > 1000 mrem/hr, accessible to personnel, that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, or that cannot be continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.