ENCLOSURE 8

MONTICELLO NUCLEAR GENERATING PLANT
LICENSE AMENDMENT REQUEST TO
SUPPORT 24-MONTH FUEL CYCLES

CURRENT MONTICELLO TS BASES PAGES
MARKED UP WITH PROPOSED CHANGES

(Provided for Information Only)

9 PAGES FOLLOW

Bases 4.0:

A. This specification provides that surveillance activities necessary to ensure the Limiting Conditions for Operations are met and will be performed during the periods when the Limiting Conditions for Operation are applicable.

B. A tolerance for performing surveillance activities beyond the nominal interval is provided to allow operational flexibility because of scheduling and performance considerations. **[It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are specified to be performed at least once each Refueling Interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed once each Refueling Interval. Likewise, it is not the intent that Refueling Interval surveillances be performed during power operation unless it is consistent with safe plant operation.]** Each surveillance test is completed within plus 25% of each scheduled date. Scheduled dates are based on dividing each calendar year into four 13-week "surveillance" quarters consisting of 3 4-week "surveillance" months and one "catch-up" week. This method of scheduling permits certain tests to always be scheduled on certain days of the week.

C. The specification ensures that surveillance activities associated with a Limiting Condition for Operation (LCO) have been performed within the specified time interval prior to entry into a plant condition for which the LCO is applicable. Under the terms of this specification, for example, during-initial plant startup or following extended plant outage, the surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment to Operable status.

D. "Affected equipment" refers to the specific equipment on which a surveillance is being performed. If there is an LCO that corresponds to the specific equipment that has failed the surveillance, then that LCO shall be entered. If there is no corresponding LCO, then the effect of inoperability of the specific equipment that has failed the surveillance shall be evaluated (i.e., by applying the definition of operability) and actions taken as appropriate (e.g., to comply with the technical specifications).

E. The specification establishes the flexibility to defer declaring affected equipment or variables inoperable or outside specified limits when a surveillance has not been completed within schedule. A delay period of up to 24 hours or up to the limit of the specified frequency, whichever is greater, applies from the point in time that it is discovered that the surveillance has not been performed and not from the time that the specified frequency was not met. This delay period permits adequate time for the completion of a surveillance before complying with required LCO specifications or other remedial measures that might preclude completion of the surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the surveillance, the safety significance of the delay in completing the surveillance, and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the requirements.

Bases 4.2:

The instrumentation in this section will be functionally tested and calibrated at regularly scheduled intervals. Although this instrumentation is not generally considered to be as important to plant safety as the Reactor Protection System, the same design reliability goals are applied. As discussed in Section 4.1 Bases, monthly or quarterly testing is generally specified unless the testing ~~must be~~ **[is]** conducted during **[a]** refueling ~~outages~~ **[interval]**. Quarterly calibration is specified unless the calibration ~~must be~~ **[is]** conducted during **[a]** refueling ~~outages~~ **[interval]**. Where applicable, sensor checks are specified on a once/12 hours or once/day basis.

NEXT PAGE IS 76

Bases 3.4/4.4:

A. The design objective of the standby liquid control system is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown assuming that none of the withdrawn control rods can be inserted. To meet this objective, the liquid control system is designed to inject a quantity of boron which produces a concentration of boron in the reactor core in less than 125 minutes sufficient to bring the reactor from full power to a 3% delta k subcritical condition considering the hot to cold reactivity swing, xenon poisoning and an additional 25% boron concentration margin to allow for leakage and imperfect mixing.

The time requirement (125 minutes) for insertion of the boron solution was selected to override the rate of reactivity insertion due to cooldown of the reactor following the xenon poison peak.

The ATWS Rule (10 CFR 50.62) requires the addition of a new design requirement to the generic SLC System design basis. Changes to flow rate, solution concentration or boron enrichment to meet the ATWS Rule do not invalidate the original system design basis. Paragraph (c)(4) of 10 CFR 50.62 states that:

> "Each boiling water reactor must have a Standby Liquid Control System (SLCS) with a minimum flow capacity and boron content equivalent in control capacity to 86 gallons per minute of 13 weight percent sodium pentaborate solution" (natural boron enrichment).

The described minimum system parameters (equivalent to 24 gpm, 10.7% concentration and 55 atom percent Boron-10 enrichment) will ensure an equivalent injection capability that meets the ATWS rule requirement.

Boron enrichment concentration, solution temperature, and volume (including check of tank heater and pipe heat tracing system) are checked on a frequency to assure a high reliability of operation of the system should it ever be required. Only one of the two standby liquid control pumping circuits is needed for proper operation of the system. If one pumping circuit is found to be inoperable, there is no immediate threat to shutdown capability, and reactor operation may continue while repairs are being made. A reliability analysis indicates that the plant can be operated safely in this manner for ten days. For additional margin, the allowable out of service time has been reduced to seven days.

The ~~only practical time to test the~~ standby liquid control system is **[tested]** during ~~a~~ **[each]** refueling ~~outage~~ **[interval]** ~~and~~ by initiation from local stations. Components of the system are checked periodically as described above and ~~make~~ **[therefore]** a functional test of the entire system on a frequency of less than once each refueling ~~outage~~ **[interval is]** unnecessary. A test of explosive charges from one manufacturing batch is made to assure that the replacement charges for the tested system are satisfactory. A continual check of the firing circuit continuity is provided by pilot lights in the control room.

The relief valves in the standby liquid control system protect the system piping and positive displacement pumps which are nominally designed for 1500 psi from overpressure. The pressure relief valves discharge back to the standby liquid control solution tank.

Bases 3.6/4.6 (Continued) :

The safety/relief valves have two functions; 1) over-pressure relief (self-actuation by high pressure), and 2) Depressurization/ Pressure Control (using air actuators to open the valves via ADS, Low-Low Set system, or manual operation).

The safety function is performed by the same safety/relief valve with self-actuated integral bellows and pilot valve causing main valve operation. Article 9, Section N-911.4(a)(4) of the ASME Pressure Vessel Code Section III Nuclear Vessels (1965 and 1968 editions) requires that these bellows be monitored for failure since this would defeat the safety function of the safety/relief valve.

Low-Low Set Logic has been provided on three non-Automatic Pressure Relief System valves. This logic is discussed in detail in the Section 3.2 Bases. This logic, through pressure sensing instrumentation, reduces the opening setpoint and increases the blowdown range of the three selected valves following a scram to eliminate the discharge line water leg clearing loads resulting from multiple valve openings.

Testing of the safety/relief valves in accordance with ANSI/ASME OM-1-1981 each refueling ~~outage~~ **[interval]** ensures that any valve deterioration is detected. An as-found tolerance value of 3% for safety/relief valve setpoints is specified in ANSI/ASME OM-1-1981. Analyses have been performed with the valves assumed to open at 3% above their setpoint of 1109 psig. The 1375 psig Code limit is not exceeded in any case. When the setpoint is being bench checked, it is prudent to disassemble one of the safety/relief valves to examine for crud buildup, bending of certain actuator members or other signs of possible deterioration.

Provision also has been made to detect failure of the bellows monitoring system. Testing of this system once per cycle provides assurance of bellows integrity.

F. Deleted

Bases 3.6/4.6 (Continued) :

To provide assurance of snubber functional reliability, a representative sample of 10% of the installed snubbers will be functionally tested during plant shutdowns at intervals of no more than ~~18~~ **24** months +25%. Observed failures of these sample snubbers will require functional testing of additional units.

The service life of a snubber is evaluated via manufacturer input and through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc. . .). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

one-inch opening of any one valve or a 1/8-inch opening for all eight valves, measured at the bottom of the disc with the top of the disc at the seat. The position indication system is designed to detect closure within 1/8 inch at the bottom of the disc.

At each refueling ~~outage~~ **[interval]** and following any significant maintenance on the vacuum breaker valves, positive seating of the vacuum breakers will be verified by leak test. The leak test is conservatively designed to demonstrate that leakage is less than that equivalent to leakage through a one-inch orifice which is about 3% of the maximum allowable. This test is planned to establish a baseline for valve performance at the start of each operating cycle and to ensure that vacuum breakers are maintained as nearly as possible to their design condition. This test is not planned to serve as a limiting condition for operation.

During reactor operation, an exercise test of the vacuum breakers will be conducted monthly. This test will verify that disc travel is unobstructed and will provide verification that the valves are closing fully through the position indication system. If one or more of the vacuum breakers do not seat fully as determined from the indicating system, a leak test will be conducted to verify that leakage is within the maximum allowable. Since the extreme lower limit of switch detection capability is approximately 1/16″, the planned test is designed to strike a balance between the detection switch capability to verify closure and the maximum allowable leak rate. A special test was performed to establish the basis for this limiting condition. During the first refueling outage all ten vacuum breakers were shimmed 1/16″ open at the bottom of the disc. The bypass area associated with the shimming corresponded to 63% of the maximum allowable[1]. The results of this test are shown in Figure 3.7.1. Two of the original ten vacuum breakers have since been removed.

When a drywell-suppression chamber vacuum breaker valve is exercised through an opening-closing cycle, the position indicating lights at the remote test panels are designed to function as follows:

Fully Closed	2 Green	- On
	2 Red	- Off
Intermediate Position	2 Green	- Off
	2 Red	- Off
Fully Open	2 Green	- Off
	2 Red	- On

The remote test panels consist of indication and controls in the control room and indication in the reactor building. The control room indication and controls for the drywell to suppression chamber vacuum breakers consist of one red light and one green light for each of the eight valves, a common

Bases 4.7 :

A. Primary Containment

The water in the suppression chamber is used only for cooling in the event of an accident. Daily checks are specified of pool temperature and level to ensure that these parameters are within their allowable ranges.

The interiors of the drywell and suppression chamber are painted to prevent corrosion. The inspection of the paint during each refueling ~~outage~~ **[interval]**, approximately once ~~per year~~ **[every two years]**, assures the paint is intact and is not deteriorating.

Experience with this type of paint indicates that the inspection interval specified is adequate.
Because of the large volume and thermal capacity of the suppression pool, the level and temperature normally change very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and frequently logged during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be points of highest stress. Visual inspection of the suppression chamber including water line regions each refueling ~~outage~~ **[interval]** is adequate to detect any changes in the suppression chamber structures.

The design basis loss of coolant accident was analyzed at the primary containment maximum allowable accident leak rate of 1.2% and has been evaluated by the NRC Staff[(1)]. Computed offsite doses are well below the guidelines of 10 CFR Part 100.

---

(1) Safety Evaluation by the Division of Reactor Licensing, US Atomic Energy Commission, in the Matter of Northern States Power Company Monticello Nuclear Generating Plant, Unit 1, Docket No. 50-263, March 18, 1970, Section 4.1.

B. Standby Gas Treatment System, and C. Secondary Containment

Initiating reactor building isolation and operation of the standby gas treatment system to maintain the design negative pressure within the secondary containment provides an adequate test of the reactor building isolation valves and the standby gas treatment system. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system operational capability. Secondary Containment Capability Test data obtained under non-calm conditions is to be extrapolated to calm wind conditions using information provided in "Summary Technical Report to the United States Atomic Energy Commission, Directorate of Licensing, on Secondary Containment Leak Rate Test", submitted by letter dated July 23, 1973, and as described in NSP letter to the NRC dated August 18, 1995, with subject, "Revision 2 to License Amendment Request Dated June 8, 1994, Standby Gas Treatment and Secondary Containment Technical Specifications."

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Standby gas treatment system inplace testing procedures will be established utilizing applicable sections of ANSI N510-1989 standard as a procedural guideline only. If painting, fire, or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals, or foreign materials, the same tests and sample analysis should be performed as required for operational use. Replacement adsorbent should be qualified according to the guidelines of Regulatory Guide 1.52 Revision 2 (March 1978) except testing should be IAW D3803-1989. The charcoal adsorber efficiency test procedures will allow for the removal of a representative sample. **[Sample removal shall be in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52 Revision 2 (March 1978) with the exception that sample removal for laboratory testing in accordance with the Regulatory Guide shall be at least once per 24 months.]** The 30°C, 95% relative humidity test per ASTM D 3803-89 is the test method to establish the methyl iodine removal efficiency of adsorbent. The sample will be at least two inches in diameter and a length equal to the thickness of the bed. If the iodine removal efficiency test results are unacceptable, all adsorbent in the system will be replaced. High efficiency particulate filters are installed before and after the charcoal filters to prevent clogging of the carbon adsorbers and to minimize potential release of particulates to the environment. An efficiency of 99% is adequate to retain particulates that may be released to the reactor building following an accident. This will be demonstrated by inplace testing with DOP as the testing medium. Any HEPA filters found defective will be replaced with filters qualified pursuant to regulatory guide position C.3.d of Regulatory Guide 1.52 Revision 2 (March 1978). Once per operating cycle demonstration of HEPA filter pressure drop, operability of inline heaters at rated power, automatic initiation of each standby gas treatment system circuit, and leakage tests after maintenance or testing which could affect leakage, is necessary to assure system performance capability.

Bases 4.9:

The monthly test of the diesel generator is conducted to check for equipment failures and deterioration. Testing is conducted up to equilibrium operating conditions to demonstrate proper operation at these conditions. The diesel will be manually started, synchronized to the bus and load picked up. It is expected that the diesel generator will be run for one to two hours. Diesel generator experience at other generating stations indicates that the testing frequency is adequate to assure a high reliability of operation should the system be required. In addition, during the test when the generator is synchronized to the bus it is also synchronized to the offsite power source and thus not completely independent of this source. To maintain the maximum amount of independence, a thirty day testing interval is also desirable.

The Surveillance Requirement for diesel generator starting air receivers ensures that, without the aid of the refill compressors, sufficient air start capacity for each diesel generator is available. The system design requirements provide power to start each diesel generator engine from two independent air starting systems. Each system consists of a pair of compressed air starting motors, an air dryer, strainer, air line lubricator, and related storage tanks that provide 100 percent redundancy for each diesel generator's starting air system. Starting at a nominal pressure of 200 psig, each air starting system has adequate capacity to start its associated diesel generator five times without recharging. The limit of 165 psig provides minimum air pressure to support three diesel generator engine starts, from each of the two starting air receivers associated with each diesel generator, without recharging, this provides for a total of six (6) starts for the associated diesel generator. The monthly surveillance requirement frequency for verifying the pressure in each starting air receiver takes into account the capacity, capability, redundancy, and diversity of the AC sources and other indications available in the control room, including alarms, to alert the operator to below normal air start pressure. During the monthly load test of the diesel generators, the diesel fuel oil transfer pump and diesel oil service pump will be operated. A sample of diesel fuel will be taken monthly to assure that the quality remains high.

The test of the emergency diesel generator ~~during the~~ **[each]** refueling ~~outage~~ **[interval]** will be more comprehensive in that it will functionally test the system, i.e., it will check diesel starting and closure of diesel breaker and sequencing of loads on the diesel. The diesel will be started by simulation of a loss of coolant accident. In addition, an undervoltage condition will be imposed to simulate a loss of offsite power. The timing sequence will be checked to assure proper loading in the time required. The only load on the diesel is that due to friction and windage and a small amount of bypass flow on each pump. Periodic tests ~~between~~ **[during]** refueling ~~outage~~ **[intervals]** check the diesel to run at full load and the pumps to deliver full flow. Periodic testing of the various components plus a functional test at a refueling interval are sufficient to maintain adequate reliability.

Although station batteries will deteriorate with time, utility experience indicates there is almost no possibility of precipitous failure. The type of surveillance described in this specification is that which has been demonstrated over the years to provide an indication of a cell becoming irregular or unserviceable long before it becomes a failure.

In addition, the checks described also provide adequate indication that the batteries have the specified ampere-hour capability.