VERMONT YANKEE NUCLEAR POWER CORPORATION

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June 21, 2001
BVY 01-52

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

Subject: **Vermont Yankee Nuclear Power Station**
License No. DPR-28 (Docket No. 50-271)
Revised Technical Specification Bases

Pursuant to 10CFR50.59, Vermont Yankee (VY) has revised the Technical Specification (TS) Bases provided herewith. These changes primarily result from VY's review of the Updated Safety Analysis Report (UFSAR) to ensure consistency among licensing basis documents.

Attachment 1 to this letter provides a mark-up of the current Technical Specification Bases pages, and Attachment 2 contains the retyped Technical Specification Bases pages with the changes incorporated. The revised pages are being provided to you to ensure that the USNRC and VY have identical TS Bases pages.

If you have any questions on this transmittal, please contact Mr. Jeffrey T. Meyer at (802) 258-4105.

Sincerely,

VERMONT YANKEE NUCLEAR POWER CORPORATION

Gautam Sen

Gautam Sen
Licensing Manager

Attachments

cc: USNRC Region 1 Administrator
USNRC Resident Inspector - VYNPS
USNRC Project Manager - VYNPS
Vermont Department of Public Service

A001

Docket No. 50-271
BVY 01-52

Attachment 1

Vermont Yankee Nuclear Power Station

Revised Technical Specification Bases

Marked-up Version of the Current Technical Specification Bases

BASES:

1.2 REACTOR COOLANT SYSTEM

The reactor coolant system is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1335 psig as measured by the vessel steam space pressure indicator is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The 1375 psig value is derived from the design pressures of the reactor pressure vessel, and the coolant system piping. The respective design pressures are 1250 psig at 575°F and 1148 psig at 560°F. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code, Section III-A for the pressure vessel, ASME Boiler and Pressure Vessel Code Section III-C for the recirculation pump casing, and USASI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10% over design pressure (110% x 1250 = 1375 psig), and the USASI Code permits pressure transients up to 20% over the design pressure (120% x 1148 = 1378 psig).

The safety valves are sized to prevent exceeding the pressure vessel code limit for the worst-case isolation (pressurization) event (MSIV closure) assuming indirect (~~neutron flux~~) scram.

nuclear system high pressure

2.2 REACTOR COOLANT SYSTEM

The settings on the reactor high pressure scram, reactor coolant system relief and safety valves, have been established to assure never reaching the reactor coolant system pressure safety limit as well as assuring the system pressure does not exceed the range of the fuel cladding integrity safety limit. In addition to preventing power operation above 1055 psig, the pressure scram backs up the APRM neutron flux scram for steam line isolation type transients. (See FSAR Section 14.5 and Supplement 2 to Proposed Change No. 14, November 12, 1973.)

The design pressure is 1250 psig at 575°F for both the reactor pressure vessel and the recirculation system piping.

BASES: 3.1 (Cont'd)

High radiation levels in the main steam line tunnel above that due to the normal nitrogen and oxygen radioactivity is an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds three times normal background. The purpose of this scram is to reduce the source of such radiation to the extent necessary to prevent release of radioactive materials to the turbine. An alarm is initiated whenever the radiation level exceeds 1.5 times normal background to alert the operator to possible serious radioactivity spikes due to abnormal core behavior. The air ejector off-gas monitors serve to back up the main steam line monitors to provide further assurance against release of radioactive materials to site environs by isolating the main condenser off-gas line to the main stack.

The main steam line isolation valve closure scram is set to scram when the isolation valves are 10 percent closed from full open in 3-out-of-4 lines. This scram anticipates the pressure and flux transient, which would occur when the valves close. By scramming at this setting, the resultant transient is insignificant.

A reactor mode switch is provided which actuates or bypasses the various scram functions [illegible] status.

The [illegible]

The Augmented Off-Gas (AOG) monitors provide further assurance against the release of radioactive materials to the site environs by isolating the AOG stack valve.

The IRM sy[illegible] conjunction with the [illegible] against excessive power levels in the startup and intermediate power ranges. A source range monitor (SRM) system is also provided to supply additional neutron level information during startup and can provide scram function with selected shorting links removed during refueling. Thus, the IRM and the reduced APRM are normally required in the startup mode and may be required in the refuel mode. During some refueling activities which require the mode switch in startup; it is allowable to disconnect the LPRMs to protect them from damage during under vessel work. In lieu of the protection provided by the reduced APRM scram, both the IRM scram and the SRM scram in noncoincidence are used to provide neutron monitoring protection against excessive power levels. In the power range, the normal APRM system provides required protection. Thus, the IRM system and 15% APRM scram are not required in the run mode.

If an unsafe failure is detected during surveillance testing, it is desirable to determine as soon as possible if other failures of a similar type have occurred and whether the particular function involved is still operable or capable of meeting the single failure criteria. To meet the requirements of Table 3.1.1, it is necessary that all instrument channels in one trip system be operable to permit testing in the other trip system.

Thus, when failures are detected in the first trip system tested, they would have to be repaired before testing of the other system could begin. In the majority of cases, repairs or replacement can be accomplished quickly. If repair or replacement cannot be completed in a reasonable time, operation could continue with one tripped system until the surveillance testing deadline.

BASES:

3.2 PROTECTIVE INSTRUMENTATION

In addition to reactor protection instrumentation which initiates a reactor scram, station protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the reactor operator's ability to control, or terminate a single operator error before it results in serious consequences. This set of Specifications provides the limiting conditions of operation for the primary system isolation function and initiation of the core standby cooling and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of any component of such systems even during periods when portions of such systems are out of service for maintenance, testing, or calibration, and (ii) to prescribe the trip settings required to assure adequate performance. This set of Specifications also provides the limiting conditions of operation for the control rod block system and surveillance instrumentation.

Isolation valves (Note 1) are installed in those lines that penetrate the primary containment and must be isolated during a loss-of-coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by protective instrumentation shown in Table 3.2.2 which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required. The objective is to isolate the primary containment so that the limits of 10 CFR 100 are not exceeded during an accident. The objective of the low turbine condenser vacuum trip is to minimize the radioactive effluent releases to as low as practical in case of a main condenser failure. Subsequent releases would continue until operator action was taken to isolate the main condenser unless the main steam line isolation valves were closed automatically on low condenser vacuum. The manual bypass is required to permit initial startup of the reactor during low power operation.

The instrumentation which initiates primary system isolation is connected in a dual channel arrangement. Thus, the discussion given in the bases for Specification 3.1 is applicable here.

The low reactor water level instrumentation is set to trip when reactor water level is 127" above the top of the enriched fuel. This trip initiates closure of Group 2 and 3 primary containment isolation valves. For a trip setting of 127" above the top of the enriched fuel, the valves will be closed before perforation of the clad occurs even for the maximum break and, therefore, the setting is adequate.

The top of the enriched fuel (351.5" from vessel bottom) is designated as a common reference level for all reactor water level instrumentation. The intent is to minimize the potential for operator confusion which may result from different scale references.

The low-low reactor water level instrumentation is set to trip when reactor water level is 82.5" H_2O indicated on the reactor water level instrumentation above the top of the enriched fuel. This trip initiates closure of the Group 1 primary containment isolation valves and also activates the ECCS and RCIC System and starts the standby diesel generator system. This trip setting level was chosen to be low enough to prevent spurious operation, but high enough to initiate ECCS operation and primary system isolation so that no melting of the fuel cladding will occur, and so that post-accident cooling can be accomplished and the limits of 10CFR100 will not be violated.

Note 1 - Isolation valves are grouped as listed in ~~Table 3.7.1.~~ Table 4.7.2.

BASES: 3.2 (Cont'd)

For the complete circumferential break of 28-inch recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria. The instrumentation also covers the full range of spectrum breaks and meets the above criteria.

The high drywell pressure instrumentation is a backup to the water level instrumentation, and in addition to initiating ECCS, it causes isolation of Group 2, 3, and 4 isolation valves. For the complete circumferential break discussed above, this instrumentation will initiate ECCS operation at about the same time as the low-low water level instrumentation, thus, the results given above are applicable here also. Group 2 isolation valves include the drywell vent, purge, and sump isolation valves. High drywell pressure activates only these valves because high drywell pressure could occur as the result of nonsafety-related causes such as not purging the drywell air during startup. Total system isolation is not desirable for these conditions and only the valves in Group 2 are required to close. The water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents and causes a trip of all primary system isolation valves.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break acci- steam flow, instrumentatio- lation valves. The prima- in the main stea- ac

Certain isolation valves including the TIP blocking valves, CAD inlet and outlet, drywell vent, purge and sump valves are isolated on high drywell pressure. However, since high drywell pressure could occur as the result of non-safety-related causes, such as not venting the drywell during startup, complete system isolation is not desirable for these conditions and only certain valves are required to close. The water level instrumentation initiates protection for the full spectrum of loss of coolant accidents and causes a trip of certain primary system isolation valves.

o- brea- ation discusse- ease of radioactivity, re exceeded.

High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure resulting from a control rod drop accident. This instrumentation causes closure of Group 1 valves, the only valves required to close for this accident. With the established setting of 3 times normal background and main steam line isolation valve closure, fission product release is limited so that 10CFR100 limits are not exceeded for the control rod drop accident. With an alarm setting of 1.5 times normal background, the operator is alerted to possible gross fuel failure or abnormal fission product releases from failed fuel due to transient reactor operation.

Pressure instrumentation is provided which trips when main steam line pressure drops below 800 psig. A trip of this instrumentation results in closure of Group 1 isolation valves. In the refuel, shutdown, and startup modes, this trip function is provided when main steam line flow exceeds 40% of rated capacity. This function is provided primarily to provide protection against a pressure regulator malfunction which would cause the

BASES: 3.6 and 4.6 (Cont'd)

Whenever an isotopic analysis is performed, a reasonable effort will be made to determine a significant percentage of those contributors representing the total radioactivity in the reactor coolant sample. Usually at least 80 percent of the total gamma radioactivity can be identified by the isotopic analysis.

It has been observed that radioiodine concentration can change rapidly in the reactor coolant during transient reactor operations, such as reactor shutdown, reactor power changes, and reactor startup if failed fuel is present. As specified, additional reactor coolant samples shall be taken and analyzed for reactor operations in which steady-state radioiodine concentrations in the reactor coolant indicate various levels of iodine releases from the fuel. Since the radioiodine concentration in the reactor coolant is not continuously measured, reactor coolant sampling would be ineffective as a means to rapidly detect gross fuel element failures. However, some capability to detect gross fuel element failures is inherent in the radiation monitors in the off-gas system on the main steam line.

Isotopic analyses required by Specification 4.6.B.1.b may be performed by a gamma scan and gross beta and alpha determination.

and

BASES:

3.7 STATION CONTAINMENT SYSTEMS

A. Primary Containment

The integrity of the primary containment and operation of the core standby cooling systems in combination limit the off-site doses to [illegible] to those suggested in 10 CFR 100 in the event of a [illegible] system piping. Thus, containment integrity is [illegible] potential for violation of the primary reactor [illegible]sts. Concern about such a violation exists [illegible]ctor is critical, above atmospheric pressure and [illegible] above 212°F. An exception is made to this requirement during initial core loading and while a low power test program is being conducted and ready access to the reactor vessel is required. The reactor may be taken critical during the period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth to less than 1.30% delta k.

internal design pressure for the pressure suppression chamber.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1000 psig.

Since all the gases in the drywell are purged into the pressure suppression chamber air space during a loss-of-coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the allowable ~~pressure suppression chamber pressure~~. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber (Reference Section 5.2 FSAR).

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 44 psig, which is below the design of 56 psig.(3) The minimum volume of 68,000 ft^3 results in a submergency of approximately four feet. The majority of the Bodega tests(2) were run with a submerged length of four feet and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate.

The maximum temperature at the end of blowdown tested during the [illegible] Bodega Bay tests was 170°F and this is [illegible] for complete condensation of the [illegible]ld occur for temperature

the calculated peak accident containment pressure is approximately 44 psig, which is below the ASME design pressure of 56 psig.

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(1) Robbins, C. H., "Tests on a Full Scale 1/48 Segment of the Humbolt Bay Pressure Suppression Containment", GEAP-3596, November 17, 1960.

(2) Bodega Bay Preliminary Hazards Summary Report, Appendix 1, Docket 50-205, December 28, 1962.

(3) ~~Code Allowable peak accident~~ pressure is 62 psig.

Internal design

BASES: 3.7 (Cont'd)

In conjunction with the Mark I Containment Long-Term Program, a plant unique analysis was performed (see Vermont Yankee letter, dated April 27, 1984, transmitting Teledyne Engineering Services Company Reports, TR-5319-1, Revision 2, dated November 30, 1983 and TR-5319-2, Revision 0) which demonstrated that all stresses in the suppression chamber structure, including shell, external supports, vent system, internal structures, and attached piping meet the structural acceptance criteria of NUREG-0661. The maintenance of a drywell-suppression chamber differential pressure of 1.7 psid and a suppression chamber water level corresponding to a downcomer submergence of less than 4.54 ft. will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

Using a 50°F rise (Section 5.2.4 FSAR) in the suppression chamber water temperature and a minimum water volume of 68,000 ft^3, the 170°F temperature which is used for complete condensation would be approached only if the suppression pool temperature is 120°F prior to the DBA-LOCA. Maintaining a pool temperature of ~~100~~°F will assure that the 170°F limit is not approached.

90

In the event a relief valve inadvertently opens or sticks open, operating procedures define the action to be taken. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

~~Double~~ isolation valves are provided on lines which penetrate the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident. Details of the isolation valves are discussed in Section 5.2 of the FSAR.

Manual primary containment isolation valves that are required to be closed by the definition of Primary Containment Integrity may be opened intermittently under administrative controls. These controls consist of stationing a dedicated operator, with whom Control Room communication is immediately available, in the immediate vicinity of the valve controls. In this way, the penetration can be rapidly isolated when a need for primary containment isolation is indicated.

Generally, double

BASES: 3.7 (Cont'd)

An alternate electrical power source for the purposes of Specification 3.7.B.1.b shall consist of either an Emergency Diesel Generator (EDG) or the Vernon Hydro tie line. Maintaining availability of the Vernon Hydro tie line as an alternative to one of the EDGs in this condition provides assurance that standby gas treatment can, if required, be operated without placing undue constraints on EDG maintenance availability. Inoperability of both trains of the SGTS or both EDGs during refueling operations requires suspension of activities that represent a potential for releasing radioactive material to the secondary containment, thus placing the plant in a condition that minimizes risk.

Use of the SGTS, without the fan and the 9 kW heater in operation, as a vent path during torus venting does not impact subsequent adsorber [illegible]ity because of the very low flows and because humidity control [illegible]ined by the standby 1 kW heaters, therefore operation in this [illegible]oes not accrue as operating time.

Generally, double

D. Primary Containment Isolation Valves

~~Double~~ isolation valves are provided on lines that penetrate the primary containment and communicate directly with the reactor vessel and on lines that penetrate the primary containment and communicate with the primary containment free space. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident.

E. Reactor Building Automatic Ventilation System Isolation Valves (RBAVSIVs)

The function of the RBAVSIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs). The operability requirements for RBAVSIVs help ensure that an adequate secondary containment boundary is maintained during and after an accident by minimizing potential paths to the environment. The RBAVSIVs must be operable (or the penetration flow path isolated) to ensure secondary containment integrity and to limit the potential release of fission products to the environment. The valves covered by this Limiting Condition for Operation are included in the Inservice Testing Program.

In the event that there are one or more RBAVSIVs inoperable, the affected penetration flow path(s) must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. The required action must be completed within the eight hour or four hour completion time, as applicable. The specified time periods are reasonable considering the time required to isolate the penetration, and the probability of a DBA occurring during this short time.

If any required action or completion time cannot be met as a result of one or more inoperable RBAVSIVs, the plant must be placed in a mode or condition where the Limiting Condition for Operation does not apply. To achieve this status during reactor power operation, the reactor must be brought to at least hot shutdown within 12 hours and to cold shutdown within 36 hours. If applicable, core alterations and the movement of irradiated fuel assemblies and the fuel cask in the secondary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be immediately initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

BASES: 4.7 (Cont'd)

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response in the event of a loss-of-coolant accident. ~~The peak drywell pressure would be about 44 psig which would rapidly reduce to 27 psig within 10 seconds following the pipe break. Following the pipe break, the suppression chamber pressure rises to 27 psig within 10 seconds, equalizes with drywell pressure and therefore rapidly decays with the drywell pressure decay.~~[1]

ASME

The design pressure of the drywell and absorption chamber is 56 psig.[2] The design leak rate is 0.5%/day at ~~a pressure of 62 psig~~. As pointed out above, the pressure response of the drywell and suppression chamber following an accident would be the same after about 10 seconds. Based on the primary containment pressure response and the fact that the drywell and suppression chamber functi[illegible] unit, the primary containment will be tested as a unit r[illegible] the individual components separately.

this pressure.

The design basis loss-of-coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 1.5%/day at 44 psig. The analysis showed that with this leak rate and a standby gas treatment system filter efficiency of 90% for halogens, 95% for particulates, and assuming the fission product release fractions stated in TID-14844, the maximum total whole body passing cloud dose is about 1.65 rem and the maximum total thyroid dose is about 280 rem at the site boundary over an exposure duration of two hours. The resultant dose that would occur for the duration of the accident at the low population distance of 5 miles is lower than those stated due to the variability of meteorological conditions that would be expected to occur over a 30-day period. Thus, these doses are the maximum that would be expected in the unlikely event of a design basis loss-of-coolant accident. These doses are also based on the assumption of no holdup in the secondary containment, resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide margin between expected off-site doses and 10 CFR 100 guidelines. An additional factor of two for conservatism is added to the above doses by limiting the test leak rate (L a) to a value of ~~0.80%/day~~.

0.80 wt.%/day.

The calculated peak accident containment pressure would be about 44 psig, which would reduce to 27 psig within about 20 seconds following the pipe break. The suppression chamber pressure rises to about 25 psig within 30 seconds, equalizes with drywell pressure, and then decays with drywell pressure.

14.6

ASME

(1) Section ~~5.2~~ of the FSAR.

(2) 62 psig is the maximum ~~allowable peak accident~~ pressure for this design (56 psig) pressure.

internal design

BASES:

3.8 RADIOACTIVE EFFLUENTS

A. Deleted

B. Deleted

C. Deleted

D. Liquid Holdup Tanks

The tanks listed in this Specification include all outdoor tanks that contain radioactivity that are not surrounded by liners, dikes, or walls capable of holding the tank contents, or that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10CFR Part ~~20.1001 -20.2401,~~ Appendix B, Table 2, Column 2, at the nearest potable water supply and in the nearest surface water supply in an Unrestricted Area.

20.1001-20.2402

E. Deleted

F. Deleted

G. Deleted

H. Deleted

I. Deleted

J. Explosive Gas Mixture

The hydrogen monitors are used to detect possible hydrogen buildups which could result in a possible hydrogen explosion. Automatic isolation of the off-gas flow would prevent the hydrogen explosion and possible damage to the augmented off-gas system. Maintaining the concentration of hydrogen below its flammability limit provides assurance that the releases of radioactive materials will be controlled.

K. Steam Jet Air Ejector (SJAE)

Restricting the gross radioactivity release rate of gases from the main condenser SJAE provides reasonable assurance that the total body exposure to an individual at the exclusion area boundary will not exceed a small fraction of the limits of 10CFR Part 100 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10CFR Part 50.

BASES:

3.10 AUXILIARY ELECTRIC POWER SYSTEMS

A. The objective of this Specification is to assure that adequate power will be available to operate the emergency safeguards equipment. Adequate power can be provided by any one of the following sources: an immediate access source through both startup transformers, backfeed through the main transformer, or either of the two diesel generators. The backfeed through the main transformer is a delayed access off-site power source. The delayed access source is made available by opening the generator no load disconnect switch and establishing a feed from the 345 kV switchyard through the main generator step up transformer and unit auxiliary transformer to the 4.16 kV buses. The delayed access source is available within an hour of loss of main generator capability to assure that fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded.

Electric power can be supplied from the off-site transmission network to the on-site Emergency Safeguards Electric Power Distribution System by two independent sources, one immediate access and one delayed access, designed and located so as to minimize to the extent practicable the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. An additional off-site source, a 4160 V tie line to Vernon Hydroelectric Station, can supply either 4160 V emergency bus. It is used to meet station blackout and Appendix R licensing requirements.

Off-site power is supplied to the 345 kV switchyard from the transmission network by three transmission lines. A 400 MVA autotransformer is connected between the 345 kV north bus and the 115 kV bus. The autotransformer is the normal source for the 115 kV bus and the station startup transformers. The autotransformer also feeds the 115 kV transmission line to Keene.

The immediate access source is supplied from the 345 kV Transmission System through the 345 kV/115 kV autotransformer. It feeds the on-site Electric Power Distribution System through the two 115 kV to 4.16 kV startup transformers and is available within seconds following a design basis accident to assure that core cooling, containment integrity and other vital functions are maintained. An alternate immediate access source through the Keene line may be made available. Its availability is dependent on its preloading which must be limited by system dispatchers prior to it being declared an immediate access source.

A qualified source consists of all breakers, transformers, switches, interrupting devices, cabling and controls required to transmit adequate power from the off-site transmission network to the on-site Emergency Safeguards Buses 3 and 4.

Two 480 V Uninterruptible Power Systems supply power to the LPCIS valves via designated Motor Control Centers. The 480 V Uninterruptible Power Systems are redundant and independent of any on-site ac power sources.

This Specification assures that at least two off-site and two on-site power sources, and both 480 V Uninterruptible Power Systems will be available before the reactor is ~~taken beyond "just critical" testing~~. In addition to assuring power source availability, all of the associated switchgear must be operable as specified to assure that the emergency cooling equipment can be operated, if required, from the power sources.

made critical

Docket No. 50-271
BVY 01-52

Attachment 2

Vermont Yankee Nuclear Power Station

Revised Technical Specification Bases

Retyped Technical Specification Bases Pages

Listing of Affected Technical Specification Bases Pages

Replace the Vermont Yankee Nuclear Power Station Technical Specifications Bases pages listed below with the revised pages included herein. The revised pages contain vertical lines in the margin indicating the areas of change.

Remove	Insert
19	19
31	31
75	75
76	76
141	141
163	163
164	164
166a	166a
167	167
175	175
220	220

BASES:

1.2 REACTOR COOLANT SYSTEM

The reactor coolant system is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1335 psig as measured by the vessel steam space pressure indicator is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The 1375 psig value is derived from the design pressures of the reactor pressure vessel, and the coolant system piping. The design pressure is 1250 psig at 575°F for both the reactor pressure vessel and the recirculation system piping. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code, Section III-A for the pressure vessel, ASME Boiler and Pressure Vessel Code Section III-C for the recirculation pump casing, and USASI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10% over design pressure (110% x 1250 = 1375 psig), and the USASI Code permits pressure transients up to 20% over the design pressure (120% x 1148 = 1378 psig).

The safety valves are sized to prevent exceeding the pressure vessel code limit for the worst-case isolation (pressurization) event (MSIV closure) assuming indirect (nuclear system high pressure) scram.

2.2 REACTOR COOLANT SYSTEM

The settings on the reactor high pressure scram, reactor coolant system relief and safety valves, have been established to assure never reaching the reactor coolant system pressure safety limit as well as assuring the system pressure does not exceed the range of the fuel cladding integrity safety limit. In addition to preventing power operation above 1055 psig, the pressure scram backs up the APRM neutron flux scram for steam line isolation type transients. (See FSAR Section 14.5 and Supplement 2 to Proposed Change No. 14, November 12, 1973.)

BASES: 3.1 (Cont'd)

High radiation levels in the main steam line tunnel above that due to the normal nitrogen and oxygen radioactivity is an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds three times normal background. The purpose of this scram is to reduce the source of such radiation to the extent necessary to prevent release of radioactive materials to the turbine. An alarm is initiated whenever the radiation level exceeds 1.5 times normal background to alert the operator to possible serious radioactivity spikes due to abnormal core behavior. The Augmented Off-Gas (AOG) monitors provide further assurance against the release of radioactive materials to the site environs by isolating the AOG stack valve.

The main steam line isolation valve closure scram is set to scram when the isolation valves are 10 percent closed from full open in 3-out-of-4 lines. This scram anticipates the pressure and flux transient, which would occur when the valves close. By scramming at this setting, the resultant transient is insignificant.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status.

The manual scram function is active in all modes, thus providing for manual means of rapidly inserting control rods during all modes of reactor operation.

The IRM system provides protection against short reactor periods and, in conjunction with the reduced APRM system provides protection against excessive power levels in the startup and intermediate power ranges. A source range monitor (SRM) system is also provided to supply additional neutron level information during startup and can provide scram function with selected shorting links removed during refueling. Thus, the IRM and the reduced APRM are normally required in the startup mode and may be required in the refuel mode. During some refueling activities which require the mode switch in startup; it is allowable to disconnect the LPRMs to protect them from damage during under vessel work. In lieu of the protection provided by the reduced APRM scram, both the IRM scram and the SRM scram in noncoincidence are used to provide neutron monitoring protection against excessive power levels. In the power range, the normal APRM system provides required protection. Thus, the IRM system and 15% APRM scram are not required in the run mode.

If an unsafe failure is detected during surveillance testing, it is desirable to determine as soon as possible if other failures of a similar type have occurred and whether the particular function involved is still operable or capable of meeting the single failure criteria. To meet the requirements of Table 3.1.1, it is necessary that all instrument channels in one trip system be operable to permit testing in the other trip system.

Thus, when failures are detected in the first trip system tested, they would have to be repaired before testing of the other system could begin. In the majority of cases, repairs or replacement can be accomplished quickly. If repair or replacement cannot be completed in a reasonable time, operation could continue with one tripped system until the surveillance testing deadline.

BASES:

3.2 PROTECTIVE INSTRUMENTATION

In addition to reactor protection instrumentation which initiates a reactor scram, station protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the reactor operator's ability to control, or terminate a single operator error before it results in serious consequences. This set of Specifications provides the limiting conditions of operation for the primary system isolation function and initiation of the core standby cooling and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of any component of such systems even during periods when portions of such systems are out of service for maintenance, testing, or calibration, and (ii) to prescribe the trip settings required to assure adequate performance. This set of Specifications also provides the limiting conditions of operation for the control rod block system and surveillance instrumentation.

Isolation valves (Note 1) are installed in those lines that penetrate the primary containment and must be isolated during a loss-of-coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by protective instrumentation shown in Table 3.2.2 which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required. The objective is to isolate the primary containment so that the limits of 10 CFR 100 are not exceeded during an accident. The objective of the low turbine condenser vacuum trip is to minimize the radioactive effluent releases to as low as practical in case of a main condenser failure. Subsequent releases would continue until operator action was taken to isolate the main condenser unless the main steam line isolation valves were closed automatically on low condenser vacuum. The manual bypass is required to permit initial startup of the reactor during low power operation.

The instrumentation which initiates primary system isolation is connected in a dual channel arrangement. Thus, the discussion given in the bases for Specification 3.1 is applicable here.

The low reactor water level instrumentation is set to trip when reactor water level is 127" above the top of the enriched fuel. This trip initiates closure of Group 2 and 3 primary containment isolation valves. For a trip setting of 127" above the top of the enriched fuel, the valves will be closed before perforation of the clad occurs even for the maximum break and, therefore, the setting is adequate.

The top of the enriched fuel (351.5" from vessel bottom) is designated as a common reference level for all reactor water level instrumentation. The intent is to minimize the potential for operator confusion which may result from different scale references.

The low-low reactor water level instrumentation is set to trip when reactor water level is 82.5" H_2O indicated on the reactor water level instrumentation above the top of the enriched fuel. This trip initiates closure of the Group 1 primary containment isolation valves and also activates the ECCS and RCIC System and starts the standby diesel generator system. This trip setting level was chosen to be low enough to prevent spurious operation, but high enough to initiate ECCS operation and primary system isolation so that no melting of the fuel cladding will occur, and so that post-accident cooling can be accomplished and the limits of 10CFR100 will not be violated.

Note 1 - Isolation valves are grouped as listed in Table 4.7.2.

BASES: 3.2 (Cont'd)

For the complete circumferential break of 28-inch recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria. The instrumentation also covers the full range of spectrum breaks and meets the above criteria.

The high drywell pressure instrumentation is a backup to the water level instrumentation, and in addition to initiating ECCS, it causes isolation of Group 2, 3, and 4 isolation valves. For the complete circumferential break discussed above, this instrumentation will initiate ECCS operation at about the same time as the low-low water level instrumentation, thus, the results given above are applicable here also. Certain isolation valves including the TIP blocking valves, CAD inlet and outlet, drywell vent, purge and sump valves are isolated on high drywell pressure. However, since high drywell pressure could occur as the result of non-safety-related causes, such as not venting the drywell during startup, complete system isolation is not desirable for these conditions and only certain valves are required to close. The water level instrumentation initiates protection for the full spectrum of loss of coolant accidents and causes a trip of certain primary system isolation valves.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. In addition to monitoring steam flow, instrumentation is provided which causes a trip of Group 1 isolation valves. The primary function of the instrumentation is to detect a break in the main steam line, thus only Group 1 valves are closed. For the worst case accident, main steam line break outside the drywell, this trip setting of 140 percent of rated steam flow in conjunction with the flow limiters and main steam line valve closure limit the mass inventory loss such that fuel is not uncovered, cladding temperatures remain less than 1295°F and release of radioactivity to the environs is well below 10CFR100.

Temperature monitoring instrumentation is provided in the main steam line tunnel to detect leaks in this area. Trips are provided on this instrumentation and when exceeded cause closure of Group 1 isolation valves. Its setting of ambient plus 95°F is low enough to detect leaks of the order of 5 to 10 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, it is a backup to high steam flow instrumentation discussed above, and for small breaks, with the resultant small release of radioactivity, gives isolation before the limits of 10CFR100 are exceeded.

High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure resulting from a control rod drop accident. This instrumentation causes closure of Group 1 valves, the only valves required to close for this accident. With the established setting of 3 times normal background and main steam line isolation valve closure, fission product release is limited so that 10CFR100 limits are not exceeded for the control rod drop accident. With an alarm setting of 1.5 times normal background, the operator is alerted to possible gross fuel failure or abnormal fission product releases from failed fuel due to transient reactor operation.

Pressure instrumentation is provided which trips when main steam line pressure drops below 800 psig. A trip of this instrumentation results in closure of Group 1 isolation valves. In the refuel, shutdown, and startup modes, this trip function is provided when main steam line flow exceeds 40% of rated capacity. This function is provided primarily to provide protection against a pressure regulator malfunction which would cause the

BASES: 3.6 and 4.6 (Cont'd)

Whenever an isotopic analysis is performed, a reasonable effort will be made to determine a significant percentage of those contributors representing the total radioactivity in the reactor coolant sample. Usually at least 80 percent of the total gamma radioactivity can be identified by the isotopic analysis.

It has been observed that radioiodine concentration can change rapidly in the reactor coolant during transient reactor operations, such as reactor shutdown, reactor power changes, and reactor startup if failed fuel is present. As specified, additional reactor coolant samples shall be taken and analyzed for reactor operations in which steady-state radioiodine concentrations in the reactor coolant indicate various levels of iodine releases from the fuel. Since the radioiodine concentration in the reactor coolant is not continuously measured, reactor coolant sampling would be ineffective as a means to rapidly detect gross fuel element failures. However, some capability to detect gross fuel element failures is inherent in the radiation monitors in the off-gas system and on the main steam line.

Isotopic analyses required by Specification 4.6.B.1.b may be performed by a gamma scan and gross beta and alpha determination.

BASES:

3.7 STATION CONTAINMENT SYSTEMS

A. Primary Containment

The integrity of the primary containment and operation of the core standby cooling systems in combination limit the off-site doses to values less than to those suggested in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical, above atmospheric pressure and temperature above 212°F. An exception is made to this requirement during initial core loading and while a low power test program is being conducted and ready access to the reactor vessel is required. The reactor may be taken critical during the period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth to less than 1.30% delta k.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1000 psig.

Since all the gases in the drywell are purged into the pressure suppression chamber air space during a loss-of-coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the allowable internal design pressure for the pressure suppression chamber. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber (Reference Section 5.2 FSAR).

Using the minimum or maximum water volumes given in the specification, the calculated peak accident containment pressure is approximately 44 psig, which is below the ASME design pressure of 56 psig.[3] The minimum volume of 68,000 ft^3 results in a submergency of approximately four feet. The majority of the Bodega tests[2] were run with a submerged length of four feet and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate.

The maximum temperature at the end of blowdown tested during the Humbolt Bay[1] and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperature above 170°F.

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(1) Robbins, C. H., "Tests on a Full Scale 1/48 Segment of the Humbolt Bay Pressure Suppression Containment", GEAP-3596, November 17, 1960.

(2) Bodega Bay Preliminary Hazards Summary Report, Appendix 1, Docket 50-205, December 28, 1962.

(3) Internal design pressure is 62 psig.

BASES: 3.7 (Cont'd)

In conjunction with the Mark I Containment Long-Term Program, a plant unique analysis was performed (see Vermont Yankee letter, dated April 27, 1984, transmitting Teledyne Engineering Services Company Reports, TR-5319-1, Revision 2, dated November 30, 1983 and TR-5319-2, Revision 0) which demonstrated that all stresses in the suppression chamber structure, including shell, external supports, vent system, internal structures, and attached piping meet the structural acceptance criteria of NUREG-0661. The maintenance of a drywell-suppression chamber differential pressure of 1.7 psid and a suppression chamber water level corresponding to a downcomer submergence of less than 4.54 ft. will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

Using a 50°F rise (Section 5.2.4 FSAR) in the suppression chamber water temperature and a minimum water volume of 68,000 ft^3, the 170°F temperature which is used for complete condensation would be approached only if the suppression pool temperature is 120°F prior to the DBA-LOCA. Maintaining a pool temperature of 90°F will assure that the 170°F limit is not approached.

In the event a relief valve inadvertently opens or sticks open, operating procedures define the action to be taken. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

Generally, double isolation valves are provided on lines which penetrate the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident. Details of the isolation valves are discussed in Section 5.2 of the FSAR.

Manual primary containment isolation valves that are required to be closed by the definition of Primary Containment Integrity may be opened intermittently under administrative controls. These controls consist of stationing a dedicated operator, with whom Control Room communication is immediately available, in the immediate vicinity of the valve controls. In this way, the penetration can be rapidly isolated when a need for primary containment isolation is indicated.

BASES: 3.7 (Cont'd)

An alternate electrical power source for the purposes of Specification 3.7.B.1.b shall consist of either an Emergency Diesel Generator (EDG) or the Vernon Hydro tie line. Maintaining availability of the Vernon Hydro tie line as an alternative to one of the EDGs in this condition provides assurance that standby gas treatment can, if required, be operated without placing undue constraints on EDG maintenance availability. Inoperability of both trains of the SGTS or both EDGs during refueling operations requires suspension of activities that represent a potential for releasing radioactive material to the secondary containment, thus placing the plant in a condition that minimizes risk.

Use of the SGTS, without the fan and the 9 kW heater in operation, as a vent path during torus venting does not impact subsequent adsorber capability because of the very low flows and because humidity control is maintained by the standby 1 kW heaters, therefore operation in this manner does not accrue as operating time.

D. Primary Containment Isolation Valves

Generally, double isolation valves are provided on lines that penetrate the primary containment and communicate directly with the reactor vessel and on lines that penetrate the primary containment and communicate with the primary containment free space. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident.

E. Reactor Building Automatic Ventilation System Isolation Valves (RBAVSIVs)

The function of the RBAVSIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs). The operability requirements for RBAVSIVs help ensure that an adequate secondary containment boundary is maintained during and after an accident by minimizing potential paths to the environment. The RBAVSIVs must be operable (or the penetration flow path isolated) to ensure secondary containment integrity and to limit the potential release of fission products to the environment. The valves covered by this Limiting Condition for Operation are included in the Inservice Testing Program.

In the event that there are one or more RBAVSIVs inoperable, the affected penetration flow path(s) must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. The required action must be completed within the eight hour or four hour completion time, as applicable. The specified time periods are reasonable considering the time required to isolate the penetration, and the probability of a DBA occurring during this short time.

If any required action or completion time cannot be met as a result of one or more inoperable RBAVSIVs, the plant must be placed in a mode or condition where the Limiting Condition for Operation does not apply. To achieve this status during reactor power operation, the reactor must be brought to at least hot shutdown within 12 hours and to cold shutdown within 36 hours. If applicable, core alterations and the movement of irradiated fuel assemblies and the fuel cask in the secondary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be immediately initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

BASES: 4.7 (Cont'd)

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response in the event of a loss-of-coolant accident. The calculated peak accident containment pressure would be about 44 psig, which would reduce to 27 psig within about 20 seconds following the pipe break. The suppression chamber pressure rises to about 25 psig within 30 seconds, equalizes with drywell pressure, and then decays with drywell pressure.[1]

The ASME design pressure of the drywell and absorption chamber is 56 psig.[2] The design leak rate is 0.5%/day at this pressure. As pointed out above, the pressure response of the drywell and suppression chamber following an accident would be the same after about 10 seconds. Based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

The design basis loss-of-coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 1.5%/day at 44 psig. The analysis showed that with this leak rate and a standby gas treatment system filter efficiency of 90% for halogens, 95% for particulates, and assuming the fission product release fractions stated in TID-14844, the maximum total whole body passing cloud dose is about 1.65 rem and the maximum total thyroid dose is about 280 rem at the site boundary over an exposure duration of two hours. The resultant dose that would occur for the duration of the accident at the low population distance of 5 miles is lower than those stated due to the variability of meteorological conditions that would be expected to occur over a 30-day period. Thus, these doses are the maximum that would be expected in the unlikely event of a design basis loss-of-coolant accident. These doses are also based on the assumption of no holdup in the secondary containment, resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide margin between expected off-site doses and 10 CFR 100 guidelines. An additional factor of two for conservatism is added to the above doses by limiting the test leak rate (L a) to a value of 0.80 wt.%/day.

(1) Section 14.6 of the FSAR.

(2) 62 psig is the maximum internal design pressure for this ASME design (56 psig) pressure.

BASES:

3.8 RADIOACTIVE EFFLUENTS

A. Deleted

B. Deleted

C. Deleted

D. Liquid Holdup Tanks

The tanks listed in this Specification include all outdoor tanks that contain radioactivity that are not surrounded by liners, dikes, or walls capable of holding the tank contents, or that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10CFR Part 20.1001-20.2402, Appendix B, Table 2, Column 2, at the nearest potable water supply and in the nearest surface water supply in an Unrestricted Area.

E. Deleted

F. Deleted

G. Deleted

H. Deleted

I. Deleted

J. Explosive Gas Mixture

The hydrogen monitors are used to detect possible hydrogen buildups which could result in a possible hydrogen explosion. Automatic isolation of the off-gas flow would prevent the hydrogen explosion and possible damage to the augmented off-gas system. Maintaining the concentration of hydrogen below its flammability limit provides assurance that the releases of radioactive materials will be controlled.

K. Steam Jet Air Ejector (SJAE)

Restricting the gross radioactivity release rate of gases from the main condenser SJAE provides reasonable assurance that the total body exposure to an individual at the exclusion area boundary will not exceed a small fraction of the limits of 10CFR Part 100 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10CFR Part 50.

BASES:

3.10 AUXILIARY ELECTRIC POWER SYSTEMS

A. The objective of this Specification is to assure that adequate power will be available to operate the emergency safeguards equipment. Adequate power can be provided by any one of the following sources: an immediate access source through both startup transformers, backfeed through the main transformer, or either of the two diesel generators. The backfeed through the main transformer is a delayed access off-site power source. The delayed access source is made available by opening the generator no load disconnect switch and establishing a feed from the 345 kV switchyard through the main generator step up transformer and unit auxiliary transformer to the 4.16 kV buses. The delayed access source is available within an hour of loss of main generator capability to assure that fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded.

Electric power can be supplied from the off-site transmission network to the on-site Emergency Safeguards Electric Power Distribution System by two independent sources, one immediate access and one delayed access, designed and located so as to minimize to the extent practicable the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. An additional off-site source, a 4160 V tie line to Vernon Hydroelectric Station, can supply either 4160 V emergency bus. It is used to meet station blackout and Appendix R licensing requirements.

Off-site power is supplied to the 345 kV switchyard from the transmission network by three transmission lines. A 400 MVA autotransformer is connected between the 345 kV north bus and the 115 kV bus. The autotransformer is the normal source for the 115 kV bus and the station startup transformers. The autotransformer also feeds the 115 kV transmission line to Keene.

The immediate access source is supplied from the 345 kV Transmission System through the 345 kV/115 kV autotransformer. It feeds the on-site Electric Power Distribution System through the two 115 kV to 4.16 kV startup transformers and is available within seconds following a design basis accident to assure that core cooling, containment integrity and other vital functions are maintained. An alternate immediate access source through the Keene line may be made available. Its availability is dependent on its preloading which must be limited by system dispatchers prior to it being declared an immediate access source.

A qualified source consists of all breakers, transformers, switches, interrupting devices, cabling and controls required to transmit adequate power from the off-site transmission network to the on-site Emergency Safeguards Buses 3 and 4.

Two 480 V Uninterruptible Power Systems supply power to the LPCIS valves via designated Motor Control Centers. The 480 V Uninterruptible Power Systems are redundant and independent of any on-site ac power sources.

This Specification assures that at least two off-site and two on-site power sources, and both 480 V Uninterruptible Power Systems will be available before the reactor is made critical. In addition to assuring power source availability, all of the associated switchgear must be operable as specified to assure that the emergency cooling equipment can be operated, if required, from the power sources.