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ATTACHMENT A

REVISED PAGES FOR THE

SEFOR TECHNICAL SPECIFICATIONS

PROPOSED CHANGE NO. 2

Sodium Coolant System

Bases

Each of the four sodium coolant loops, including pumps, heat exchangers, and associated controls and coolant equipment, must be operable during reactor operation to assure adequate core cooling capability for normal and emergency conditions. The 300°F minimum temperature in the sodium loops assures that the sodium temperature will be maintained above the plugging temperature to avoid potential oxide plugging problems. The 300°F value provides a reasonable margin above 275°F, which is expected to be the lowest plugging temperature that can be clearly determined. Plugging temperatures below 275°F are difficult to determine, because the characteristic drop in flow with decreasing temperature is not clearly distinguishable at lower temperatures. In addition, the 300°F minimum temperature in the primary loops assures adequate shutdown margin for the core as specified in 3.3.A.

The pump-around loop circulates sodium continuously between the reactor vessel and the primary drain tank. This loop must be operable during reactor operation to maintain the reactor sodium level within prescribed limits and to provide assurance that the loop is available for accident situations.(1)

The pump around flow rate will be set high enough to avoid numerous low flow alarms due to normal flow rate variations. The average flow rate over a reasonable period of time will not exceed 2 gpm during reactor operation.

The argon cover gas system is required to be operable to maintain the conditions described in Specification 3.4.D. The vent vacuum pump is required to reprime the auxiliary primary coolant system in the event sodium is lost from that system during some abnormal (accident) condition.

The cover gas pressure in the secondary system is set equal to the cover gas pressure in the reactor (which is at a lower elevation) so that the secondary sodium pressure in the IHX will be greater than the primary sodium pressure in the IHX. This will assure that leakage of radioactive sodium from the primary coolant system to the secondary coolant system will not occur.(2)

Under normal operating conditions, the secondary sodium pressure will exceed the primary sodium pressure by about 30 psi in the main IHX and about 43 psi in the auxiliary IHX. Small leaks may occur in the IHX, but the differential pressure will prevent the radioactive primary sodium from entering the secondary system. If the primary and secondary loops were drained, the gas leakage through the allowable IHX leak, which corresponds to a hole size of about 22 mils for the main IHX, would be about 95 ft^3 in a 24 hour period at a pressure differential of 10 psi.(3) This is about .15% of the primary

3.7 Radioactive Waste Control System

Applicability

Applies to those components which control the collection, storage, and release of radioactive waste materials.

Objective

To assure the capability for safe control of radioactive waste materials and to define the limiting conditions for release of effluents from the reactor system.

Specification

A. At least one of the three waste gas compressors shall be operable.

B. For reactor startup, at least two waste gas compressors shall be operable.

C. The rate of discharge, Q_x, of radioactive effluent, x, from the plant stack shall be limited in accordance with the following equations:(1)

1. Annual average release rate, except halogens and particulates with half-lives greater than 8 days;

$$\sum_x \frac{Q_x}{MPC_x} \leq 4.0 \times 10^{10} \frac{cc}{sec}$$

2. For periods less than 48 hours in any seven consectuve days, hourly average release rate, except halogens and particulates with half-lives greater than 8 days;

$$\sum_x \frac{Q_x}{MPC_x} \leq 1.7 \times 10^{11} \frac{cc}{sec}$$

3. Annual average release rate of radioactive halogens and particulates with half-lives greater than 8 days;

$$\sum_x \overline{MPC}_x \quad 5.6 \times 10^{7} \frac{cc}{sec}$$

4. For periods less than 48 hours in any seven consecutive days, hourly average release rate of radioactive halogens and particulates with half-lives greater than 8 days;

$$\sum_x \frac{Q_x}{MPC_x} \leq 5.6 \times 10^{8} \frac{cc}{sec}$$

References

(1) 10 CFR 20, Appendix B, Table II.

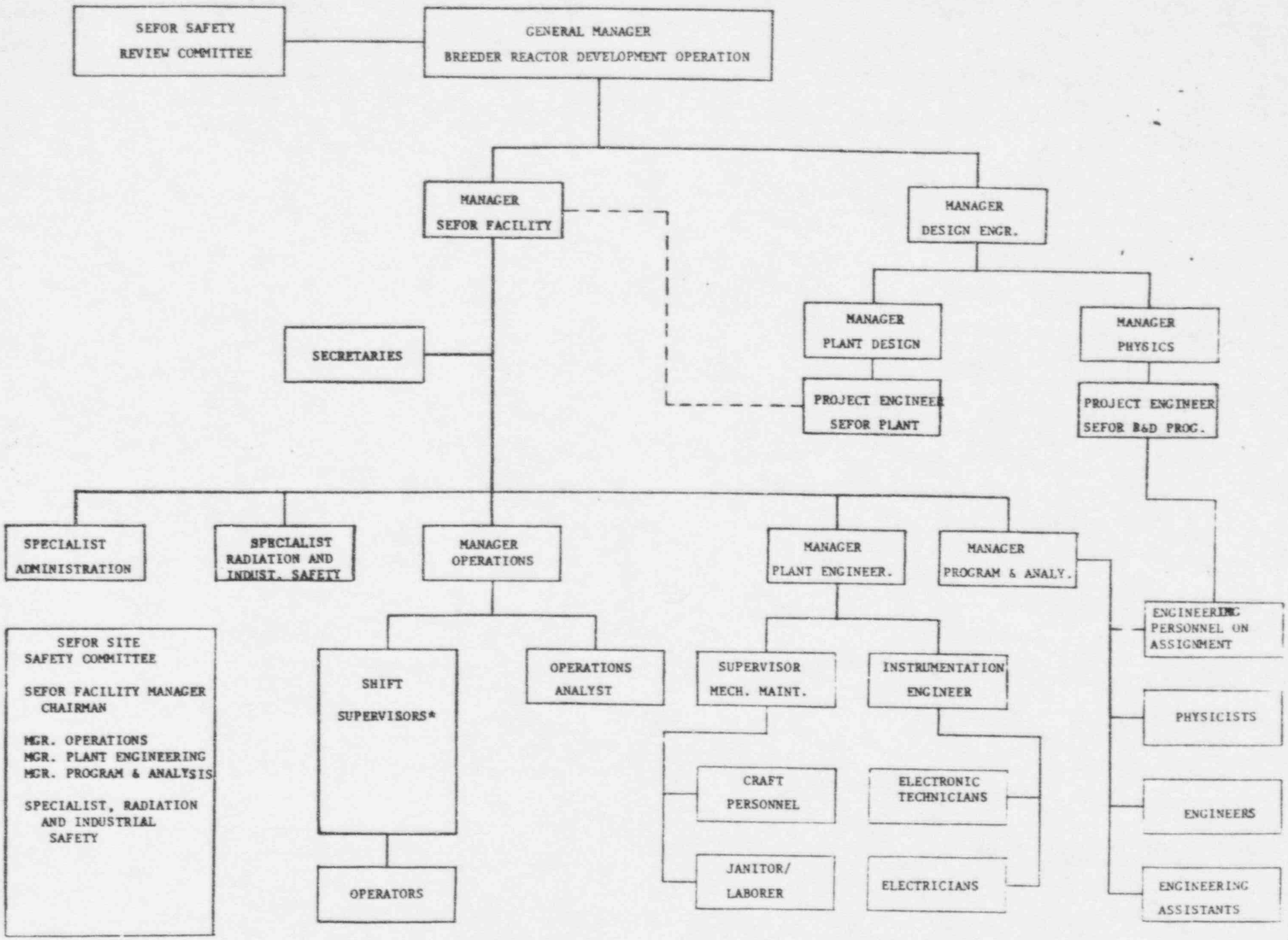
(2) SEFOR FDSAR, Volume I, Para. 9.3.1.2, pp. 9-2 and 9-3.

(3) SEFOR FDSAR, Supplement 19, Appendix A. Summary of SEFOR Meteorological Data Final Report, May 15, 1967 to May 15, 1968.

(4) A Brief Survey of the Meteorological Aspects of Atmospheric Pollution, H.E. Cramer, Bulletin of the American Meteorological Society 40 (4): 165-171.

(5) Watson, E.C. and Gamertsfelder, C.C., "Environmental Radioactive Contamination as a Factor in Nuclear Plant Siting Criteria," HW-SA-2809, February, 1963.

(6) SEFOR FDSAR, Supplement 21, p. 32.



*PART 55 LICENSES REQUIRED FOR THESE POSITIONS.

d. Personnel requiring Part 55 licenses shall be as indicated in Figure 6-1.

3. Qualifications with regard to education and operating experience for key supervisory personnel shall be as follows:

a. SEFOR Facility Manager

B.S. in Engineering or Science or equivalent in experience. Seven years experience in the design, construction, installation, operation, development, and maintenance of nuclear facilities.

Demonstrated detailed and comprehensive knowledge in related technical fields, including reactor physics, radiological hazards control, nuclear engineering and instrument engineering.

Five years experience or one year experience at SEFOR plus two years experience elsewhere in the supervision and management of the construction and operation of reactor facilities. Demonstrated ability to plan, organize, and direct reactor plant operations.

b. Manager, Plant Engineering

B.S. in Engineering or Science or equivalent in experience. Five years experience or equivalent in the operation and maintenance of power-generation facilities, including a minimum of one year in responsible supervisory positions in the operation or maintenance of such facilities.

Ability to plan, program, and direct activities of engineering and craft personnel.

Demonstrated ability in the design and application of equipment and devices, with a thorough understanding of process equipment such as pumps, fans, heat exchangers and generators, heaters, etc., as applicable to nuclear facilities.

c. Manager, Operations

B.S. in Engineering or Science or equivalent in experience. Five years experience in the operation and maintenance of reactor or nuclear power facilities, including minimum of one year in supervisory positions in the operation and maintenance of such facilities.

Demonstrated ability to organize and coordinate plant operations. Comprehensive knowledge of problems associated with startup and initial operation of reactor facilities, including knowledge of radiological hazards, technical aspects of reactor operation of control systems, radiation shielding, contamination control, etc. Demonstrated good judgment necessary to make correct decisions under rapidly changing conditions.

d. Manager, Programs and Analysis

B.S. in Engineering or Science or equivalent in experience. Five years experience in the design, operation, analysis and programming of a variety of reactor types or nuclear power facilities, including at least one year in responsible supervisory position in such organizations.

Comprehensive knowledge of reactor physics, reactor design, reactor operation, radiation shielding, fluid flow, thermodynamics, instrumentation, and related technologies.

Demonstrated capability for directing the efforts of physicists and engineers.

Ability to develop techniques and test procedures to carry out a reactor experimental program.

Demonstrated knowledge of the practical aspects of the operation of reactors, including their characteristics, limitations, and safe operating requirements.

e. Specialist, Radiation and Industrial Safety

B.S. in Engineering or Science or equivalent in experience.

Three years experience in analytical chemistry or radiochemistry and health physics.

Demonstrated ability in evaluation of radiation hazards, design and development of radiation monitoring equipment, and in conducting health-physics studies.

Thorough understanding of radiation dosimetry and a working knowledge of design of radiation facilities, shielding calculations and design of ventilation control, radioactive waste processing, calibration of radiation measuring instrumentation, maximum permissible radiation exposure levels, and good radiological safety and health protection practices. Must be cognizant of local and state industrial safety requirements. Demonstrated ability in teaching, lecturing, and implementing safe practices and procedures.

f. Supervisor, Mechanical Maintenance

B.S. in Engineering or Science, or equivalent in experience with a high school education and apprenticeship training.

Three years of maintenance experience with reactor or power generating equipment, with one year of this experience as a supervisor in the maintenance of associated equipment for reactors or power generating equipment.

Knowledge of craft techniques and mechanical maintenance procedures applicable to nuclear facilities.

Cognizance of radiation and safety procedures and regulations, as applicable to nuclear facilities.

g. Instrumentation Engineer

B.S. in Engineering or Science, or equivalent in experience. Three years experience in design, installation, calibration and maintenance of process or nuclear instrumentation. Cognizance of significance of control and instrumentation systems with respect to reactor operation and safety. Demonstrated ability to analyze systems for adequacy to meet systems requirements and to conceive, assemble, and install necessary modifications to meet systems requirements.

h. Shift Supervisor

B.S. in Engineering or Science, or equivalent in experience. Three years experience in the operation of reactor or nuclear facilities.

ATTACHMENT B

ERRATA 7 FOR FDSAR,

SUPPLEMENT 19

PAGES 53, 54, 55

Answer 5 (Continued)

3. Reactor Vessel Sodium Level Change

A third method of leak detection is provided by a change in sodium level in the reactor vessel. The sensitivity of this leak detection system depends on the rate at which the pump-around system supplies sodium to the reactor vessel.

To improve the sensitivity of this system, the pump-around loop (which provides a continuous supply of sodium to the reactor vessel) will be operated at 2 gpm rather than the 25 gpm reported in Section 4, Vol. I of the FDSAR. With this change, sodium leaks in the primary system greater than 2 gpm will cause a drop in level in the reactor vessel which will cause a reactor scram on low sodium level.

The safety implications of changing the pump-around loop flow rate from 25 gpm to 2 gpm have been evaluated. A review of the safety evaluation given in Section 16.3.3.2.3 of Vol. II of the FDSAR and on Page 1 of Supplement 18 to the FDSAR reveals that the flow rate of the pump-around loop is only important for large guillotine-type failures. For the guillotine failure, the pump-around loop is used to supply sodium to the reactor vessel to refill the auxiliary loop. Since refilling the auxiliary primary system is already an operator controlled action, operation of the pump-around loop at 2 gpm rather than 25 gpm will require one additional operator action for this accident; namely to increase the pump-around loop flow from 2 gpm to 25 gpm. The operator would take this action upon observation of low sodium level in the reactor vessel combined with the alarm which signals low flow in the auxiliary primary system. (The latter would occur when flow stops after the auxiliary system is dumped.) The analysis of the guillotine pipe break given in Supplement 18 was re-evaluated to determine the length of time the operator would have to manually increase the pump-around loop. This analysis indicated that the operator would have at least five minutes to increase the flow in the pump-around loop without exceeding the peak temperature of 1480°F that occurs during the initial phase of the accident (40 seconds after the failure).

Answer 5 (Continued)

Significance of Primary Coolant System Leak to Reflector Control System Operation

A review of the primary coolant system and its relation to the reflector control system indicates that only leaks occurring in the immediate vicinity of the reactor vessel could possibly affect the operation of the reflector control system. The electrical contact-type leak detectors discussed above would detect any significant leak that could occur in the primary system piping so the only sodium leak of concern would be a leak in one of the reactor vessel nozzles, (for example, Point A shown on Figure 5-1), or in the reactor vessel wall itself above the elevation of the safety vessel. A leak detector is located on both sides of the first weld of the piping to the reactor vessel nozzle. The possibility of leaks developing closer to the vessel than the first weld is extremely remote because of the low stress levels in the reactor vessel wall and the nozzles for normal operation. However, if a sodium leak should occur in either of these areas, it would not necessarily be detected by the electrical contact leak detectors located under the piping so the leak indication would be provided by (a) the inner containment atmosphere monitor (Na-24 monitor) or (b) by a sodium level change in the reactor vessel. The size of the leak determines which of these two methods of leak detection will give the first indication of this type of leak. For leaks less than the 2 gpm flow rate of the pump-around loop, the first indication of a leak would be given by the Na-24 detectors. As indicated above, the high sensitivity of this device would provide an indication of high Na-24 activity in the inner containment atmosphere within several minutes after the leak developed. The amount of sodium that would leak out of this less than 2 gpm leak prior to a leak indication would therefore be only a few gallons. The sodium leaking out would flow down past Point B on Figure 5-1, but would be prevented from reaching the reflectors by the high velocity cooling gas (150 ft/sec) at Point C. Thus the reactor could be safely shut down and core cooling would be provided by the auxiliary coolant system.

For leaks greater than the 2 gpm flow rate of the pump-around loop, the sodium level would start to fall in the reactor vessel and reactor scram would occur due to low sodium level. The low level trip, which is 4 inches below the normal operating level, would occur after a net sodium loss of 15 gallons. Thus, for example, a 3 gpm leak would cause a reactor scram in 15 minutes. The total amount of sodium leaking out during this time would be 45 gallons.

Answer 5 (Continued)

Again, the high velocity nitrogen coolant flow (150 ft/sec) at Point C would prevent the 45 gallons of sodium from reaching the reflector region so that a normal reactor scram would occur on low reactor vessel sodium level. As before, core cooling would be provided by the auxiliary coolant system. In actuality, the reactor would be shut down in less than 15 minutes because of a high Na-24 activity indication from the inner containment atmosphere monitor.

For larger leaks (greater than 3 gpm), the reactor would be scrammed in a shorter time due to low sodium level. In the limit, the reactor would be fully shut down about 1 second after a guillotine pipe break. For leaks less than 3 gpm, the leak indication would be obtained from the Na-24 monitor and the reactor shut down manually before a low sodium level scram would occur and before enough sodium could leak out to affect the reflector control system.