PROPOSED CHANGE RTS-152
TO THE
DUANE ARNOLD ENERGY CENTER
TECHNICAL SPECIFICATIONS

The holders of License DPR-49 for the Duane Arnold Energy Center propose to amend Appendix A (Technical Specifications) to said license by deleting current pages and replacing them with the attached new pages. A list of the affected pages is given below.

The purpose of this change is to reduce the required minimum flow rate of the Residual Heal Removal Service Water (RHRSW) pumps as given in Section 4.5.C.1(b) of the Technical Specifications. This change was prompted by numerous instances of failing to meet the required flow rate during surveillance testing. As part of the effort to solve the problem, Iowa Electric contracted General Electric (GE) to analyze the RHRSW system to determine the minimum flow rate required to meet the design basis conditions. The RHRSW system's primary function is to provide cooling water to the Residual Heat Removal (RHR) system heat exchangers during various modes of the RHR system. The design specification for the RHR system states that the shutdown cooling mode is considered to be the design basis, but that the steam condensing mode should also be evaluated as it may sometimes govern. Attachment 1 contains the GE analysis of these modes of the RHR system. The results of which justify a reduction of up to 30% in the RHRSW flow rate from the present requirements.

As part of the Mark I containment modifications, suppression pool temperature limits were set which would prevent unstable steam condensation during blowdowns to the containment. NUREG-0783 established guidelines for demonstrating conformance to the limits on local suppression pool temperatures for T-type quenchers on Safety Relief Valve discharge lines. Attachment 2 contains the GE analysis on the suppression pool cooling mode of the RHR system using the NUREG guidelines. Appendix B of this report demonstrates conformance to the above limits with a 15% reduction in the present RHRSW flow rate.

The design bases for modes of the RHR system which use the RHR heat exchangers have been analyzed with reduced RHRSW flow. The results of these analyses show that a 15% reduction in the presently required RHRSW flow rate is justified.

The changes being made are as follows:

1) Reduce by 15% the required minimum flow rate for the RHRSW system as given in Section 4.5.C.1(b).

2) Update the Bases and References for Section 4.5 to support the 15% reduction in RHRSW flow.

3) Update the Bases and References for Section 3.7 to include a discussion of the NUREG-0783 requirements and the results of the GE analysis.

4) Consolidate text on pages 3.7-1, 3.7-1a, 3.7-2 and delete pages 3.7-1a and 3.7-1b.

List of Pages Affected

3.5-5
3.5-18
3.5-26
3.7-1
3.7-1a*
3.7-1b*
3.7-2
3.7-32
3.7-32a**
3.7-49

* page deleted
** new page

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT		
		Item	Frequency
	b)	Flow Rate Test-Each RHR service water pump shall deliver at least 2040 gpm at a TDH of 610 ft. or more.	After major pump maintenance and every 3 months
2. From and after the date that one of the RHR Service Water subsystem pumps is made or found to be inoperable for any reason, reactor operation must be limited to thirty days unless operability of that pump is restored within this period. During such thirty days all other active components of the RHR Service Water subsystem are operable.	2.	When it is determined that one RHR Service Water pump is inoperable, the remaining components of that subsystem and the other subsystems shall be demonstrated to be operable immediately and daily thereafter.	
3. From and after the date that one RHR Service Water Subsystem is made or found to be inoperable for any reason, reactor operation is limited to seven days unless operability of that subsystem is restored within this period. During such seven days all active components of the other RHR Service Water subsystem and its associated diesel-generators required for operation of such components (if no external source of power were available), shall be operable.	3.	When one RHR Service Water subsystem becomes inoperable, the operable subsystem and the diesel-generators required for operation of such components shall be demonstrated to be operable immediately and daily thereafter.	

maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure, design deficiency, etc., caused the out-of-service period, then the demonstration of operability should be thorough enough to assure that a similar problem does not exist on the remaining components. For example, if an out-of-service period were caused by failure of a pump to deliver rated capacity, the other pumps of this type might be subjected to a capacity test. In any event, surveillance procedures, as required by Section 6 of these specifications, detail the required extent of testing.

The pump capacity test is a comparison of measured pump performance parameters to shop performance tests. Tests during normal operation will be performed by measuring the flow indication and/or the pump discharge pressure will be measured and its power requirement will be used to establish flow at that pressure.

Analyses were performed to determine the minimum required flow rate of the RHR Service Water pumps in order to meet the design basis case (Reference 4) and the NUREG-0783 requirements (Reference 5). (See Section 3.7.A.1 Bases for a discussion of the NUREG requirements.) The results of these analyses justify reducing the required flowrate to 2040 gpm per pump, a 15% reduction in the original 2400 gpm per pump requirement.

D. HPCI System

The HPCI system is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the nuclear system and loss-of-coolant, which

3.5 REFERENCES

1. Jacobs, I.M., "Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards", General Electric Company, APED, April 1968 (APED 5736).

2. General Electric Company, General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50, Appendix K, NEDO-20566, 1974, and letter MFN-255-77 from Darrell G. Eisenhut, NRC, to E.D. Fuller, GE, Documentation of the Reanalysis Results for the Loss-of-Coolant Accident (LOCA) of Lead and Non-lead Plants, dated June 30, 1977.

3. General Electric, Loss-of-Coolant Accident Analysis Report for Duane Arnold Energy Center (Lead Plant), NEDO-21082-02-1A, Rev. 2, June 1982.

4. General Electric Company, Analysis of Reduced RHR Service Water Flow at the Duane Arnold Energy Center, NEDE-30051-P, January 1983.

5. General Electric Company, Duane Arnold Energy Center Suppression Pool Temperature Response, NEDC-22082-P, March 1982.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.7 PLANT CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. At any time that the nuclear system is pressurized above atmospheric or work is being done which has the potential to drain the vessel, the suppression pool water volume and temperature shall be maintained with the following limits.

a. Maximum water volume - 61,500 cubic feet

b. Minimum water volume - 58,900 cubic feet

c. Maximum water temperature

(1) During normal power operation - 95F.

(2) During testing which adds heat to the suppression pool, the water temperature shall not exceed 10F above the normal power operation limit specified in (1) above. In connection with such testing, the pool temperature must be reduced to below the normal power operation limit specified in (1) above within 24 hours.

4.7 PLANT CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment system integrity.

Objective:

To verify the integrity of the primary and secondary containments.

Specification:

A. Primary Containment

1.a. The pressure suppression pool water level and temperature shall be checked once per day.

b. Whenever there is indication of relief valve operation or testing which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated.

c. Whenever there is indication of relief valve operation with the temperature of the suppression pool reaching 160F or more and the primary coolant pressure greater than 200 psig, an external visual examination of the suppression chamber shall be conducted before resuming power operation.

d. A visual inspection of the suppression chamber interior, including water line regions, shall be made at each major refueling outage.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
(3) The reactor shall be scrammed from any operating condition if the pool temperature reaches 110F. Power operation shall not be resumed until the pool temperature is reduced below the normal power operation limit specified in (1) above.	
(4) During reactor isolation conditions, the reactor shall be depressurized to less than 200 psig at normal cooldown rates if the pool temperature reaches 120°F.	
2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 Mw(t).	2. The primary containment integrity shall be demonstrated as follows: a. Type A Test Primary Reactor Containment Integrated Leakage Rate Test 1) The interior surfaces of the drywell and torus shall be visually inspected each operating cycle for evidence of deterioration. In addition, the external surfaces of the torus below the water level shall be inspected on a routine basis for evidence of torus corrosion or leakage. Except for the initial Type A test, all Type A tests shall be performed without any preliminary leak detection surveys and leak repairs immediately prior to the test. If a Type A test is completed but the acceptance criteria of Specification 4.7.A.2.a.(9) is not satisfied and repairs are necessary, the Type A test need not be repeated provided locally measured leakage reductions, achieved by repairs, reduce the containment's overall measured leakage rate sufficiently to meet the acceptance criteria.

to downcomer submergence, this specification is adequate. The maximum temperature at the end of blowdown tested during the Humbolt Bay 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 temperatures above 170°F.

Using a 50°F rise (Table 6.2-1, UFSAR) in the suppression chamber water temperature and a minimum water volume of 58,900 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 95°F will assure that the 170°F limit is not approached.

As part of the program to reduce the loads on BWR containments, the NRC issued NUREG-0783, which limits local suppression pool temperatures during Safety Relief Valve (SRV) actuations. Stable steam condensation is assured in the vicinity of T-type quenchers on SRV discharge lines if the following limits on local suppression pool temperatures are met:

1. For all plant transients involving SRV operations during which the steam flux through the quencher perforations exceeds 94 lbm/ft^2-sec, the suppression pool local temperature shall not exceed 200°F.

2. For all plant transients involving SRV operations during which the steam flux through the quencher perforations is less than 42 lbm/ft^2-sec, the suppression pool local temperature shall be at least 20°F subcooled.

3. For all plant transients involving SRV operations during which the steam flux through the quencher perforations exceeds 42 lbm/ft^2-sec, but less than 94 lbm/ft^2-sec, the suppression pool local temperature is obtained by linearly interpolating the local temperatures established under aforementioned items 1 and 2.

Maintaining the suppression pool temperature below the normal operating limit of 95°F, and scramming the reactor if the pool temperature reaches 110°F, will ensure that the local temperature limits outlined above are not exceeded during plant transients.[7]

Should it be necessary to drain the suppression chamber, this should only be done when there is no requirement for core standby cooling systems operability as explained in Bas.. 3.5.G or the requirements of Specification 3.5.G.4 are met.

2. Inerting

Safety Guide No. 7 assumptions for metal-water reactions result in hydrogen concentrations in excess of the Safety

3.7.A & 4.7.A REFERENCES

1. Section 14.6 of the FSAR.

2. ASME Boiler and Pressure Vessel Code, Nuclear Vessels, Section III, maximum allowable internal pressure is 62 psig.

3. Staff Safety Evaluation of DAEC, USAEC, Directorate of Licensing, January 23, 1973.

4. 10 CFR 50.54, Appendix J, Reactor Containment Testing Requirements, Federal Register, August 27, 1971.

5. DAEC Short-Term Program Plant Unique Analysis, NUTECH Doc. No. IOW-01-065, August 1976.

6. Supplement to DAEC Short-Term Program Plant Unique Analysis, NUTECH Doc. No. IOW-01-071, October 1976.

7. General Electric Company, Duane Arnold Energy Center Suppression Pool Temperature Response, NEDC-22082-P, March 1982.