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3.4 DECAY HEAT REMOVAL CAPABILITY

Applicability

Applies to the operating status of systems and components that function to remove decay heat when one or more fuel bundles are located in the reactor vessel.

Objective

To define the conditions necessary to assure continuous capability of decay heat removal.**

Specification

3.4.1 Reactor Coolant System temperature greater than 250°F.

3.4.1.1 With the Reactor Coolant System temperature greater than 250°F, three independent EFW pumps and associated flow paths shall be OPERABLE with:

a. Two EFW pumps, each capable of being powered from an OPERABLE emergency bus, and one EFW pump capable of being powered from an OPERABLE steam supply system. Specification 3.0.1 applies.

b. With one pump or flow path* inoperable, restore the inoperable pump or flow path to OPERABLE status within 72 hours or be in COLD SHUTDOWN within the next 12 hours. With more than one EFW pump or flow path* inoperable, restore the inoperable pumps or flow paths* to OPERABLE status or be subcritical within 1 hour, in at least HOT SHUTDOWN within the next 6 hours, and in COLD SHUTDOWN within the following 6 hours.

c. Four of six turbine bypass valves OPERABLE.

d. The condensate storage tanks (CST) OPERABLE with a minimum of 150,000 gallons of condensate available in each CST. With a CST inoperable, restore the CST to operability within 72 hours or be in at least HOT SHUTDOWN within the next 6 hours, and COLD SHUTDOWN within the next 30 hours. With more than one CST inoperable, restore the inoperable CST to OPERABLE status or be subcritical within 1 hour, in at least HOT SHUTDOWN within the next 6 hours, and in COLD SHUTDOWN within the following 6 hours. Specification 3.0.1 applies.

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*For the purpose of this requirement, an OPERABLE flow path shall mean an unobstructed path from the water source to the pump and from the pump to a steam generator.

**These requirements supplement the requirements of Sections 3.1.1.1.c, 3.1.1.2, 3.3.1 and 3.8.3.

3.4.1.2 With the Reactor Coolant System temperature greater than 250°F, all eighteen (18) main steam safety valves shall be OPERABLE or, if any are not OPERABLE, the maximum overpower trip setpoint (see Table 2.3-1) shall be reset as follows:

Maximum Number of Safety Valves Disabled on Any Steam Generator	Maximum Overpower Trip Setpoint (% of Rated Power)
1	92.4
2	79.4
3	66.3

With more than 3 main steam safety valves inoperable, restore at least fifteen (15) main steam safety valves to OPERABLE status within 4 hours or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3.4.2 Reactor Coolant System temperature 250°F or less.

3.4.2.1 With Reactor Coolant temperature 250°F or less, at least two of the following means for maintaining decay heat removal capability shall be OPERABLE and at least one shall be in operation except as allowed by Specifications 3.4.2.2, 3.4.2.3 and 3.4.2.4.

a. Decay Heat Removal String "A".

b. Decay Heat Removal String "B".

c. Reactor Coolant Loop "A", its associated OTSG, and its associated emergency feedwater flowpath.

d. Reactor Coolant Loop "B", its associated OTSG, and its associated emergency feedwater flowpath.

3.4.2.2 Operation of the means for decay heat removal may be suspended provided the core outlet temperature is maintained below saturation temperature.

3.4.2.3 The number of means for decay heat removal required to be operable per 3.4.2.1 may be reduced to one provided that one of the following conditions is satisfied:

a. The Reactor is in a Refueling Shutdown condition with the Fuel Transfer Canal water level greater than 23 feet above the reactor vessel flange.

b. Reactor coolant temperature is less than 140°F with BWST level greater than 44 feet and an associated flow path through the RCS OPERABLE such that core outlet temperature can be maintained subcooled for at least 7 days.

c. Equipment Maintenance on one of the means for decay heat removal specified by 3.4.2.1 is required and the equipment outage does not exceed 7 days.

3.4.2.4 Specification 3.4.2.1 does not apply when either of the following conditions exist:

a. Decay heat generation is less than 188 KW with the RCS full.

b. Decay heat generation is less than 100 KW with the RCS drained down for maintenance.

3.4.2.5 With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible.

Bases

A reactor shutdown following power operation requires removal of core decay heat. Normal decay heat removal is by the steam generators with the steam dump to the condenser when RCS temperature is above 250°F and by the decay heat removal system below 250°F. Core decay heat can be continuously dissipated up to 15 percent of full power via the steam bypass to the condenser as feedwater in the steam generator is converted to steam by heat absorption. Normally, the capability to return feedwater flow to the steam generators is provided by the main feedwater system.

The main steam safety valves will be able to relieve to atmosphere the total steam flow if necessary. If Main Steam Safety Valves are inoperable, the power level must be reduced, as stated in Technical Specification 3.4.1.2 such that the remaining safety valves can prevent overpressure on a turbine trip.

In the unlikely event of complete loss of off-site electrical power to the station, decay heat removal is by either the steam-driven emergency feedwater pump, or two half-sized motor-driven pumps. Steam discharge is to the atmosphere via the main steam safety valves and controlled atmospheric relief valves, and in the case of the turbine driven pump, from the turbine exhaust. (1)

Both motor-driven pumps are required initially to remove decay heat with one eventually sufficing. The minimum amount of water in the condensate storage tanks, contained in Technical Specification 3.4.1.1, will allow cooldown to 250°F with steam being discharged to the atmosphere. After cooling to 250°F, the decay heat removal system is used to achieve further cooling.

When the RCS is below 250°F, a single DHR string, or single OTSG and its associated emergency feedwater flowpath is sufficient to provide removal of decay heat at all times following the cooldown to 250°F. The requirement to maintain two OPERABLE means of decay heat removal ensures that a single failure does not result in a complete loss of decay heat removal capability. The requirement to keep a system in operation as necessary to maintain the system subcooled at the core outlet provides the guidance to ensure that steam conditions which could inhibit core cooling do not occur.

Limited reduction in redundancy is allowed for preventive or corrective maintenance on the primary means for decay heat removal to ensure that maintenance necessary to assure the continued reliability of the systems may be accomplished.

As decay heat loads are reduced through decay time or fuel off loading, alternate flow paths will provide adequate cooling for a time sufficient to take compensatory action if the normal means of heat removal is lost.

With the reactor vessel head removed and 23 feet of water above the reactor vessel flange, a large heat sink is available for core cooling. The BWST with level at 44 feet provides an equivalent reservoir available as a heat sink. Operability of the BWST is to be determined using calculations based on actual plant data or through plant testing at the time the system is to be declared operable. At such times that either of these means is determined to be operable, removal of the redundant or diverse cooling system is permitted.

Following extensive outages or major core off loading, the decay heat generation being removed from the Reactor Vessel is so low that ambient losses are sufficient to maintain core cooling and no other means of heat removal is required. The system is passive and requires no redundant or diverse backup system. Decay heat generation is calculated in accordance with ANSI 5.1-1979 to determine when this situation exists.

An unlimited emergency feedwater supply is available from the river via either of the two motor-driven reactor building emergency cooling water pumps for an indefinite period of time.

The requirements of Technical Specification 3.4.1.1 assure that before the reactor is heated to above 250°F, adequate auxiliary feedwater capability is available. One turbine driven pump full capacity (920 gpm) and the two half-capacity motor-driven pumps (460 gpm each) are specified. However, only one half-capacity motor-driven pump is necessary to supply auxiliary feedwater flow to the steam generators in the onset of a small break loss-of-coolant accident.

REFERENCES

(1) FSAR Section 10.2.1.3.

4.9 DECAY HEAT REMOVAL CAPABILITY - PERIODIC TESTING

Applicability

Applies to the periodic testing of systems or components which function to remove decay heat.

Objective

To verify that systems/components required for decay heat removal are capable of performing their design function.

Specification

4.9.1 Emergency Feedwater System - Periodic Testing (Reactor Coolant System Temperature greater than 250°F.)

4.9.1.1 Whenever the Reactor Coolant System temperature is greater than 250°F, the EFW pumps shall be tested in the recirculation mode in accordance with the requirements and acceptance criteria of ASME Section XI Article IWP-3210. The test frequency shall be at least every 31 days of plant operation at Reactor Coolant Temperature above 250°F.

4.9.1.2 During testing of the EFW System when the reactor is in STARTUP, HOT STANDBY or POWER OPERATION, if one steam generator flow path* is made inoperable, a dedicated qualified individual who is in communication with the control room shall be continuously stationed at the EFW local manual valves (See Table 4.9-1). On instruction from the Control Room Operator, the individual shall realign the valves from the test mode to their operational alignment.

4.9.1.3 At least once per 31 days each valve listed in Table 4.9-1 shall be verified to be in the status specified in Table 4.9-1, when required to be OPERABLE.

4.9.1.4 On a quarterly basis, verify that the manual control (HIC-849/850) valve station functions properly.

4.9.1.5 On a quarterly basis, EFV-30A and B shall be checked for proper operation by cycling each valve over its full stroke.

4.9.1.6 Prior to start-up, following a refueling shutdown or a cold shutdown greater than 30 days, conduct a test to demonstrate that the motor driven EFW pumps can pump water from the condensate tanks to the Steam Generators.

*For the purpose of this requirement, an OPERABLE flow path shall mean an unobstructed path from the water source to the pump and from the pump to a Steam Generator.

4.9.1.7 Acceptance Criteria

These tests shall be considered satisfactory if control board indication and visual observation of the equipment demonstrates that all components have operated properly except for the tests required by Specification 4.9.1.1.

4.9.2 Decay Heat Removal Capability - Periodic Testing (Reactor Coolant System Temperature 250°F or less).*

4.9.2.1 On a daily basis, verify operability of the means for decay heat removal required by specification 3.4.2 by observation of console status indication.

*These requirements supplement the requirements of 4.5.2.2 and 4.5.4.

Bases

The 31-day testing frequency will be sufficient to verify that the turbine driven and two motor-driven EFW pumps are operable and that the associated valves are in the correct alignment. ASME Section XI, Article IWP-3210 specifies requirements and acceptance standards for the testing of nuclear safety related pumps. Compliance with the normal acceptance criteria assures that the EFW pumps are operating as expected. The test frequency of 31 days (nominal) has been demonstrated by the B&W Emergency Feedwater Reliability Study to assure an appropriate level of reliability. In the case of the EFW System flow, the flow shall be considered acceptable if under the worst case single pump failure, a minimum of 500 gpm can be delivered when steam generator pressure is 1050 psig and one steam generator is isolated. A flow of 500 gpm, at 1050 psig head, ensures that sufficient flow can be delivered to either Steam Generator. The surveillance requirements ensure that the overall EFW System functional capability is maintained.

Daily verification of the operability of the required means for decay heat removal ensures that sufficient decay heat removal capability will be maintained.

GPU Nuclear

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5211-85-2088
May 14, 1985

Office of Nuclear Reactor Regulation
Attn: John F. Stolz, Chief
Operating Reactors Branch No. 4
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Stolz:

Three Mile Island Nuclear Station, Unit I, (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
Technical Specification Change Request No. 142 (Rev. 1)

Enclosed are three originals and forty conformed copies of Technical Specification Change Request No. 142 (Rev. 1).

Also enclosed is one signed copy of the Certificate of Service for this request to the chief executives of the township and county in which the facility is located, as well as the Bureau of Radiation Protection.

Pursuant to 10 CFR 50.91(a)(1), we enclose our analyses, using the standards in 10 CFR 50.92 of significant hazards considerations. As stated above, pursuant to 10 CFR 50.91(a) of the regulations, we have provided a copy of this letter, the proposed change in Technical Specifications, and our analyses of significant hazards considerations to Thomas Gerusky, the designated representative of the Commonwealth of Pennsylvania.

Pursuant to the provisions of 10 CFR 170.21, a check for $150.00 was provided with our letter of March 8, 1985 as payment of the fee associated with Technical Specification Change Request No. 142.

Sincerely,

8505210281

H. D. Hukill
Director, TMI-1

HDH/MRK/spb

Enclosures: 1) Technical Specification Change Request No. 142 (Rev. 1)
2) Certificate of Service for Technical Specification Change Request No. 142 (Rev. 1)

cc: J. Thoma

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER & LIGHT COMPANY

AND

PENNSYLVANIA ELECTRIC COMPANY

THREE MILE ISLAND NUCLEAR STATION, UNIT 1

Operating License No. DPR-50
Docket No. 50-289
Technical Specification Change Request No. 142 (Rev. 1)

This Technical Specification Change Request is submitted in support of Licensee's request to change Appendix A to Operating License No. DPR-50 for Three Mile Island Nuclear Station, Unit 1. As a part of this request, proposed replacement pages for Appendix A are also included.

GPU NUCLEAR CORPORATION

BY H D Hukill
Director, TMI-1

Sworn and Subscribed
to before me this 14th
day of May, 1985.

Darla Jean Berry
Notary Public

DARLA JEAN BERRY, NOTARY PUBLIC
MIDDLETOWN BORO, DAUPHIN COUNTY
MY COMMISSION EXPIRES JUNE 17, 1985

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

DOCKET NO. 50-289
LICENSE NO. DPR-50

GPU NUCLEAR CORPORATION

This is to certify that a copy of Technical Specification Change Request No. 142 (Rev. 1) to Appendix A of the Operating License for Three Mile Island Nuclear Station Unit 1, has, on the date given below, been filed with executives of Londonderry Township, Dauphin County, Pennsylvania; Dauphin County, Pennsylvania; and the Pennsylvania Department of Environmental Resources, Bureau of Radiation Protection, by deposit in the United States mail, addressed as follows:

Mr. Jay H. Kopp, Chairman
Board of Supervisors of
Londonderry Township
R. D. #1, Geyers Church Road
Middletown, PA 17057

Mr. John E. Minnich, Chairman
Board of County Commissioners
of Dauphin County
Dauphin County Courthouse
Harrisburg, PA 17120

Mr. Thomas Gerusky, Director
PA. Dept. of Environmental Resources
Bureau of Radiation Protection
P.O. Box 2063
Harrisburg, PA 17120

GPU NUCLEAR CORPORATION

BY H D Hukill
Director, TMI-1

DATE: May 14, 1985

I. TECHNICAL SPECIFICATION CHANGE REQUEST (TSCR) NO. 142 (Rev. 1)

The Licensee requests that the revised pages attached replace the following pages of the existing Technical Specifications:

ii, iv, 3-25, 3-26, 3-26a, 3-26b, 3-26c, 4-52, 4-52a

This Request No. 142 (Rev. 1) supersedes the previous Request No. 142 in its entirety.

II. REASON FOR CHANGE

By letter dated July 18, 1984, NRC provided its evaluation of GPUN's submittal (LIL 341) of January 26, 1984 regarding Decay Heat Removal and requested that GPUN include certain aspects of Standard Technical Specifications (STS) into the TMI-1 Technical Specifications or provide justification as to why the requirements are not necessary.

This TSCR provides additional technical specification requirements and clarification for maintaining decay heat removal capability below 250°F reactor coolant temperature as described in GPUN's letter of October 10, 1984, in order to assure redundant or diverse decay heat removal capability without reliance upon administrative requirements or management directives alone and incorporates NRC staff comment on TSCR No. 142.

III. SAFETY EVALUATION JUSTIFYING CHANGE

The changes made through this proposed revision incorporate additional requirements and clarification which meet the intent of STS (NUREG-0103, Rev. 4) in order to assure DHR capability for plant conditions with reactor coolant temperature less than 250°F. For this reason, these changes will have a beneficial effect on plant safety. Exceptions or modifications to certain aspects of the STS which are described in NRC's letter of July 18, 1984 are justified as follows:

1) This change is structured to conform to the TMI-1 T.S. format, applicable to the plant conditions which correspond most closely to STS modes of operation and meets the intent of STS to assure DHR capability.

2) Surveillance of RCP operation at power is performed continuously by the Reactor Protection System. Without a RCP in operation in each loop, the RPS will trip the reactor as specified in TMI-1 T.S. Table 2.3-1. This accomplishes the STS goal of verifying RCP operation. If an RCP is not in operation in each loop, the operator will know because the reactor will trip. Also, equipment status is turned over as part of the normal shift change process. Any inoperable RCP would be part of this turnover process when at power.

Surveillance requirements for reactor protection system instrumentation are given in TMI-1 T.S. 4.1.1. Procedures for these requirements are specified by TMI-1 T.S. 6.8 to be implemented and maintained requiring review and approval prior to implementation and periodic review as set forth in administrative procedures. Therefore, additional surveillance requirements for verification of RCP operation at power are not included as part of this change.

3) This change includes surveillance specifications that the means for decay heat removal below 250°F which are required to be operable be verified operable daily. Detailed surveillance procedures are required by Specification 6.8 in order to implement these surveillance requirements. However, during shutdown plant conditions, reactor coolant pump operability is not required inasmuch as natural circulation is an acceptable means for decay heat removal as discussed in Section III.6. Therefore, a surveillance requirement to document verification of the operability of reactor coolant pumps every 7 days is unnecessary.

   Major equipment which is in operation to maintain the conditions of the Reactor Coolant System, such as decay heat removal loops and reactor coolant loops, is under continual observation as a part of the normal control room operator duties. Equipment status is turned over as part of the normal shift change process. Therefore, a specific Technical Specification surveillance requirement to document the verification of operation of such equipment is unnecessary.

4) This change recognizes heat losses to the Reactor Building atmosphere as an acceptable means of decay heat removal at decay heat generation rates below 188 KW with the RCS full and below 100 KW with the RCS drained down for maintenance (TMI-1 calculation, C3320-85-001). When decay heat load is very low, as with the present plant conditions at TMI-1, heat loss to ambient is sufficient to provide adequate decay heat removal capability. This cooling method requires no active components but relies upon basic heat transfer principles.

5) This change requires equipment to be in operation only when needed to circulate reactor coolant in order to maintain the reactor coolant system in a subcooled condition. While STS Sections 3.4.1.3, 3.4.1.4 and 3.9.8.1 require equipment to be in operation allowing only a maximum of one hour down time with certain stipulations applied, this change allows equipment to be secured for longer periods where conditions permit.

   Depending on the decay heat generation rate, shutdown of the forced circulation equipment may allow reactor coolant system temperature to increase very slowly some time after shutdown compared to the conditions immediately following plant shutdown from power operation. During these conditions of slow temperature increase, continuous operation of a decay heat removal loop would not be necessary and suspension of operation would be permitted under 3.4.2.2.

6) This change recognizes the acceptability of cooldown by natural circulation as an acceptable means of decay heat removal. The adequacy of natural circulation cooldown as a stable means of decay heat removal is presented in TMI-1 FSAR Table 14.1-12.

7) This change allows a limited period of up to 7 days for which the requirement for operability of a redundant or diverse means of DHR may be suspended with reactor coolant temperature 250°F or less. This is to provide for preventive or corrective maintenance that may be necessary to ensure the continued reliability of the preferred means of DHR capability. This provides the assurance also that appropriate action will be taken to restore the preferred means of DHR capability to operable status in a timely manner, without prohibiting maintenance which is needed to decrease the likelihood of actual in-service failures. A period of up to 7 days is justified considering the low probability and minimal consequences of such a system failure.

8) This change recognizes that the DHR system is not the only system capable of providing a flow of borated cooling water through the reactor vessel below 250°F. In addition to the equipment allowed by STS in fulfilling the requirements for DHR capability, this change allows the use of a flow path from the BWST with BWST level greater than 44 ft. as an alternate flow path whenever such means are determined to be capable of maintaining RCS in a subcooled condition for at least 7 days. The length of time such an alternate flow path would be available for decay heat removal is predictable using calculations based on actual plant data or through plant testing at the time the system is to be declared operable.

   Surveillance requirements are included to verify the operability of the circulating path daily whenever the flow path is required to be operable.

These changes as discussed above are justified in that the requirements embodied in this TSCR provide for continuous decay heat removal capability, provide additional guidance, and specify a level of redundancy while allowing systems to be taken out of service for proper maintenance to be performed.

IV. NO SIGNIFICANT HAZARDS CONSIDERATIONS

These proposed changes provide additional operational requirements to assure redundant or diverse decay heat removal capability. Additional limiting conditions for operation and additional surveillance requirements are included. Therefore, operation of TMI-1 in accordance with this TSCR:

1) does not involve a significant increase in the probability or consequences of an accident previously evaluated,

2) does not create the possibility of a new or different kind of accident from any accident previously evaluated, and

3) does not involve a significant reduction in a margin of safety.

Therefore, significant safety hazards are not associated with this change.

V. IMPLEMENTATION

It is requested that the amendment authorizing this change become effective 120 days after receipt, to allow for the necessary procedural revisions to be put in place.

VI. AMENDMENT FEE (10 CFR 170.21)

Pursuant to the provisions of 10 CFR 170.21, a check for $150.00 was provided with our letter of March 8, 1985 as payment of the fee associated with TSCR No. 142.

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3.4 DECAY HEAT REMOVAL CAPABILITY

Applicability

Applies to the operating status of systems and components that function to remove decay heat when one or more fuel bundles are located in the reactor vessel.

Objective

To define the conditions necessary to assure continuous capability of decay heat removal.**

Specification

3.4.1 Reactor Coolant System temperature greater than 250°F.

3.4.1.1 With the Reactor Coolant System temperature greater than 250°F, three independent EFW pumps and associated flow paths shall be OPERABLE with:

a. Two EFW pumps, each capable of being powered from an OPERABLE emergency bus, and one EFW pump capable of being powered from an OPERABLE steam supply system. Specification 3.0.1 applies.

b. With one pump or flow path* inoperable, restore the inoperable pump or flow path to OPERABLE status within 72 hours or be in COLD SHUTDOWN within the next 12 hours. With more than one EFW pump or flow path* inoperable, restore the inoperable pumps or flow paths* to OPERABLE status or be subcritical within 1 hour, in at least HOT SHUTDOWN within the next 6 hours, and in COLD SHUTDOWN within the following 6 hours.

c. Four of six turbine bypass valves OPERABLE.

d. The condensate storage tanks (CST) OPERABLE with a minimum of 150,000 gallons of condensate available in each CST. With a CST inoperable, restore the CST to operability within 72 hours or be in at least HOT SHUTDOWN within the next 6 hours, and COLD SHUTDOWN within the next 30 hours. With more than one CST inoperable, restore the inoperable CST to OPERABLE status or be subcritical within 1 hour, in at least HOT SHUTDOWN within the next 6 hours, and in COLD SHUTDOWN within the following 6 hours. Specification 3.0.1 applies.

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*For the purpose of this requirement, an OPERABLE flow path shall mean an unobstructed path from the water source to the pump and from the pump to a steam generator.

**These requirements supplement the requirements of Sections 3.1.1.1.c, 3.1.1.2, 3.3.1 and 3.8.3.

3.4.1.2 With the Reactor Coolant System temperature greater than 250°F, all eighteen (18) main steam safety valves shall be OPERABLE or, if any are not OPERABLE, the maximum overpower trip setpoint (see Table 2.3-1) shall be reset as follows:

Maximum Number of Safety Valves Disabled on Any Steam Generator	Maximum Overpower Trip Setpoint (% of Rated Power)
1	92.4
2	79.4
3	66.3

With more than 3 main steam safety valves inoperable, restore at least fifteen (15) main steam safety valves to OPERABLE status within 4 hours or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3.4.2 Reactor Coolant System temperature 250°F or less.

3.4.2.1 With Reactor Coolant temperature 250°F or less, at least two of the following means for maintaining decay heat removal capability shall be OPERABLE and at least one shall be in operation except as allowed by Specifications 3.4.2.2, 3.4.2.3 and 3.4.2.4.

a. Decay Heat Removal String "A".

b. Decay Heat Removal String "B".

c. Reactor Coolant Loop "A", its associated OTSG, and its associated emergency feedwater flowpath.

d. Reactor Coolant Loop "B", its associated OTSG, and its associated emergency feedwater flowpath.

3.4.2.2 Operation of the means for decay heat removal may be suspended provided the core outlet temperature is maintained below saturation temperature.

3.4.2.3 The number of means for decay heat removal required to be operable per 3.4.2.1 may be reduced to one provided that one of the following conditions is satisfied:

a. The Reactor is in a Refueling Shutdown condition with the Fuel Transfer Canal water level greater than 23 feet above the reactor vessel flange.

b. Reactor coolant temperature is less than 140°F with BWST level greater than 44 feet and an associated flow path through the RCS OPERABLE such that core outlet temperature can be maintained subcooled for at least 7 days.

c. Equipment Maintenance on one of the means for decay heat removal specified by 3.4.2.1 is required and the equipment outage does not exceed 7 days.

3.4.2.4 Specification 3.4.2.1 does not apply when either of the following conditions exist:

a. Decay heat generation is less than 188 KW with the RCS full.

b. Decay heat generation is less than 100 KW with the RCS drained down for maintenance.

3.4.2.5 With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible.

Bases

A reactor shutdown following power operation requires removal of core decay heat. Normal decay heat removal is by the steam generators with the steam dump to the condenser when RCS temperature is above 250°F and by the decay heat removal system below 250°F. Core decay heat can be continuously dissipated up to 15 percent of full power via the steam bypass to the condenser as feedwater in the steam generator is converted to steam by heat absorption. Normally, the capability to return feedwater flow to the steam generators is provided by the main feedwater system.

The main steam safety valves will be able to relieve to atmosphere the total steam flow if necessary. If Main Steam Safety Valves are inoperable, the power level must be reduced, as stated in Technical Specification 3.4.1.2 such that the remaining safety valves can prevent overpressure on a turbine trip.

In the unlikely event of complete loss of off-site electrical power to the station, decay heat removal is by either the steam-driven emergency feedwater pump, or two half-sized motor-driven pumps. Steam discharge is to the atmosphere via the main steam safety valves and controlled atmospheric relief valves, and in the case of the turbine driven pump, from the turbine exhaust. (1)

Both motor-driven pumps are required initially to remove decay heat with one eventually sufficing. The minimum amount of water in the condensate storage tanks, contained in Technical Specification 3.4.1.1, will allow cooldown to 250°F with steam being discharged to the atmosphere. After cooling to 250°F, the decay heat removal system is used to achieve further cooling.

When the RCS is below 250°F, a single DHR string, or single OTSG and its associated emergency feedwater flowpath is sufficient to provide removal of decay heat at all times following the cooldown to 250°F. The requirement to maintain two OPERABLE means of decay heat removal ensures that a single failure does not result in a complete loss of decay heat removal capability. The requirement to keep a system in operation as necessary to maintain the system subcooled at the core outlet provides the guidance to ensure that steam conditions which could inhibit core cooling do not occur.

Limited reduction in redundancy is allowed for preventive or corrective maintenance on the primary means for decay heat removal to ensure that maintenance necessary to assure the continued reliability of the systems may be accomplished.

As decay heat loads are reduced through decay time or fuel off loading, alternate flow paths will provide adequate cooling for a time sufficient to take compensatory action if the normal means of heat removal is lost.

With the reactor vessel head removed and 23 feet of water above the reactor vessel flange, a large heat sink is available for core cooling. The BWST with level at 44 feet provides an equivalent reservoir available as a heat sink. Operability of the BWST is to be determined using calculations based on actual plant data or through plant testing at the time the system is to be declared operable. At such times that either of these means is determined to be operable, removal of the redundant or diverse cooling system is permitted.

Following extensive outages or major core off loading, the decay heat generation being removed from the Reactor Vessel is so low that ambient losses are sufficient to maintain core cooling and no other means of heat removal is required. The system is passive and requires no redundant or diverse backup system. Decay heat generation is calculated in accordance with ANSI 5.1-1979 to determine when this situation exists.

An unlimited emergency feedwater supply is available from the river via either of the two motor-driven reactor building emergency cooling water pumps for an indefinite period of time.

The requirements of Technical Specification 3.4.1.1 assure that before the reactor is heated to above 250°F, adequate auxiliary feedwater capability is available. One turbine driven pump full capacity (920 gpm) and the two half-capacity motor-driven pumps (460 gpm each) are specified. However, only one half-capacity motor-driven pump is necessary to supply auxiliary feedwater flow to the steam generators in the onset of a small break loss-of-coolant accident.

REFERENCES

(1) FSAR Section 10.2.1.3.

4.9 DECAY HEAT REMOVAL CAPABILITY - PERIODIC TESTING

Applicability

Applies to the periodic testing of systems or components which function to remove decay heat.

Objective

To verify that systems/components required for decay heat removal are capable of performing their design function.

Specification

4.9.1 Emergency Feedwater System - Periodic Testing (Reactor Coolant System Temperature greater than 250°F.)

4.9.1.1 Whenever the Reactor Coolant System temperature is greater than 250°F, the EFW pumps shall be tested in the recirculation mode in accordance with the requirements and acceptance criteria of ASME Section XI Article IWP-3210. The test frequency shall be at least every 31 days of plant operation at Reactor Coolant Temperature above 250°F.

4.9.1.2 During testing of the EFW System when the reactor is in STARTUP, HOT STANDBY or POWER OPERATION, if one steam generator flow path* is made inoperable, a dedicated qualified individual who is in communication with the control room shall be continuously stationed at the EFW local manual valves (See Table 4.9-1). On instruction from the Control Room Operator, the individual shall realign the valves from the test mode to their operational alignment.

4.9.1.3 At least once per 31 days each valve listed in Table 4.9-1 shall be verified to be in the status specified in Table 4.9-1, when required to be OPERABLE.

4.9.1.4 On a quarterly basis, verify that the manual control (HIC-849/850) valve station functions properly.

4.9.1.5 On a quarterly basis, EFV-30A and B shall be checked for proper operation by cycling each valve over its full stroke.

4.9.1.6 Prior to start-up, following a refueling shutdown or a cold shutdown greater than 30 days, conduct a test to demonstrate that the motor driven EFW pumps can pump water from the condensate tanks to the Steam Generators.

*For the purpose of this requirement, an OPERABLE flow path shall mean an unobstructed path from the water source to the pump and from the pump to a Steam Generator.

4.9.1.7 Acceptance Criteria

These tests shall be considered satisfactory if control board indication and visual observation of the equipment demonstrates that all components have operated properly except for the tests required by Specification 4.9.1.1.

4.9.2 Decay Heat Removal Capability - Periodic Testing (Reactor Coolant System Temperature 250°F or less).*

4.9.2.1 On a daily basis, verify operability of the means for decay heat removal required by specification 3.4.2 by observation of console status indication.

*These requirements supplement the requirements of 4.5.2.2 and 4.5.4.

Bases

The 31-day testing frequency will be sufficient to verify that the turbine driven and two motor-driven EFW pumps are operable and that the associated valves are in the correct alignment. ASME Section XI, Article IWP-3210 specifies requirements and acceptance standards for the testing of nuclear safety related pumps. Compliance with the normal acceptance criteria assures that the EFW pumps are operating as expected. The test frequency of 31 days (nominal) has been demonstrated by the B&W Emergency Feedwater Reliability Study to assure an appropriate level of reliability. In the case of the EFW System flow, the flow shall be considered acceptable if under the worst case single pump failure, a minimum of 500 gpm can be delivered when steam generator pressure is 1050 psig and one steam generator is isolated. A flow of 500 gpm, at 1050 psig head, ensures that sufficient flow can be delivered to either Steam Generator. The surveillance requirements ensure that the overall EFW System functional capability is maintained.

Daily verification of the operability of the required means for decay heat removal ensures that sufficient decay heat removal capability will be maintained.

GPU Nuclear

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5211-85-2088
May 14, 1985

Office of Nuclear Reactor Regulation
Attn: John F. Stolz, Chief
Operating Reactors Branch No. 4
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Stolz:

Three Mile Island Nuclear Station. Unit I, (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
Technical Specification Change Request No. 142 (Rev. 1)

Enclosed are three originals and forty conformed copies of Technical Specification Change Request No. 142 (Rev. 1).

Also enclosed is one signed copy of the Certificate of Service for this request to the chief executives of the township and county in which the facility is located, as well as the Bureau of Radiation Protection.

Pursuant to 10 CFR 50.91(a)(1), we enclose our analyses, using the standards in 10 CFR 50.92 of significant hazards considerations. As stated above, pursuant to 10 CFR 50.91(a) of the regulations, we have provided a copy of this letter, the proposed change in Technical Specifications, and our analyses of significant hazards considerations to Thomas Gerusky, the designated representative of the Commonwealth of Pennsylvania.

Pursuant to the provisions of 10 CFR 170.21, a check for $150.00 was provided with our letter of March 8, 1985 as payment of the fee associated with Technical Specification Change Request No. 142.

Sincerely,

H. D. Hukill
Director, TMI-1

HDH/MRK/spb

Enclosures: 1) Technical Specification Change Request No. 142 (Rev. 1)
2) Certificate of Service for Technical Specification Change Request No. 142 (Rev. 1)

cc: J. Thoma

~~8505210281~~

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER & LIGHT COMPANY

AND

PENNSYLVANIA ELECTRIC COMPANY

THREE MILE ISLAND NUCLEAR STATION, UNIT 1

Operating License No. DPR-50
Docket No. 50-289
Technical Specification Change Request No. 142 (Rev. 1)

This Technical Specification Change Request is submitted in support of Licensee's request to change Appendix A to Operating License No. DPR-50 for Three Mile Island Nuclear Station, Unit 1. As a part of this request, proposed replacement pages for Appendix A are also included.

GPU NUCLEAR CORPORATION

BY H D Hukill
Director, TMI-1

Sworn and Subscribed
to before me this 14th
day of May, 1985.

Darla Jean Berry
Notary Public

DARLA JEAN BERRY, NOTARY PUBLIC
MIDDLETOWN BORO, DAUPHIN COUNTY
MY COMMISSION EXPIRES JUNE 17, 1985
Member, Pennsylvania Association of Notaries

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

DOCKET NO. 50-289
LICENSE NO. DPR-50

GPU NUCLEAR CORPORATION

This is to certify that a copy of Technical Specification Change Request No. 142 (Rev. 1) to Appendix A of the Operating License for Three Mile Island Nuclear Station Unit 1, has, on the date given below, been filed with executives of Londonderry Township, Dauphin County, Pennsylvania; Dauphin County, Pennsylvania; and the Pennsylvania Department of Environmental Resources, Bureau of Radiation Protection, by deposit in the United States mail, addressed as follows:

Mr. Jay H. Kopp, Chairman
Board of Supervisors of
Londonderry Township
R. D. #1, Geyers Church Road
Middletown, PA 17057

Mr. John E. Minnich, Chairman
Board of County Commissioners
of Dauphin County
Dauphin County Courthouse
Harrisburg, PA 17120

Mr. Thomas Gerusky, Director
PA. Dept. of Environmental Resources
Bureau of Radiation Protection
P.O. Box 2063
Harrisburg, PA 17120

GPU NUCLEAR CORPORATION

BY H D Hukill
Director, TMI-1

DATE: May 14, 1985

I. TECHNICAL SPECIFICATION CHANGE REQUEST (TSCR) NO. 142 (Rev. 1)

The Licensee requests that the revised pages attached replace the following pages of the existing Technical Specifications:

ii, iv, 3-25, 3-26, 3-26a, 3-26b, 3-26c, 4-52, 4-52a

This Request No. 142 (Rev. 1) supersedes the previous Request No. 142 in its entirety.

II. REASON FOR CHANGE

By letter dated July 18, 1984, NRC provided its evaluation of GPUN's submittal (LIL 341) of January 26, 1984 regarding Decay Heat Removal and requested that GPUN include certain aspects of Standard Technical Specifications (STS) into the TMI-1 Technical Specifications or provide justification as to why the requirements are not necessary.

This TSCR provides additional technical specification requirements and clarification for maintaining decay heat removal capability below 250°F reactor coolant temperature as described in GPUN's letter of October 10, 1984, in order to assure redundant or diverse decay heat removal capability without reliance upon administrative requirements or management directives alone and incorporates NRC staff comment on TSCR No. 142.

III. SAFETY EVALUATION JUSTIFYING CHANGE

The changes made through this proposed revision incorporate additional requirements and clarification which meet the intent of STS (NUREG-0103, Rev. 4) in order to assure DHR capability for plant conditions with reactor coolant temperature less than 250°F. For this reason, these changes will have a beneficial effect on plant safety. Exceptions or modifications to certain aspects of the STS which are described in NRC's letter of July 18, 1984 are justified as follows:

1) This change is structured to conform to the TMI-1 T.S. format, applicable to the plant conditions which correspond most closely to STS modes of operation and meets the intent of STS to assure DHR capability.

2) Surveillance of RCP operation at power is performed continuously by the Reactor Protection System. Without a RCP in operation in each loop, the RPS will trip the reactor as specified in TMI-1 T.S. Table 2.3-1. This accomplishes the STS goal of verifying RCP operation. If an RCP is not in operation in each loop, the operator will know because the reactor will trip. Also, equipment status is turned over as part of the normal shift change process. Any inoperable RCP would be part of this turnover process when at power.

Surveillance requirements for reactor protection system instrumentation are given in TMI-1 T.S. 4.1.1. Procedures for these requirements are specified by TMI-1 T.S. 6.8 to be implemented and maintained requiring review and approval prior to implementation and periodic review as set forth in administrative procedures. Therefore, additional surveillance requirements for verification of RCP operation at power are not included as part of this change.

3) This change includes surveillance specifications that the means for decay heat removal below 250°F which are required to be operable be verified operable daily. Detailed surveillance procedures are required by Specification 6.8 in order to implement these surveillance requirements. However, during shutdown plant conditions, reactor coolant pump operability is not required inasmuch as natural circulation is an acceptable means for decay heat removal as discussed in Section III.6. Therefore, a surveillance requirement to document verification of the operability of reactor coolant pumps every 7 days is unnecessary.

   Major equipment which is in operation to maintain the conditions of the Reactor Coolant System, such as decay heat removal loops and reactor coolant loops, is under continual observation as a part of the normal control room operator duties. Equipment status is turned over as part of the normal shift change process. Therefore, a specific Technical Specification surveillance requirement to document the verification of operation of such equipment is unnecessary.

4) This change recognizes heat losses to the Reactor Building atmosphere as an acceptable means of decay heat removal at decay heat generation rates below 188 KW with the RCS full and below 100 KW with the RCS drained down for maintenance (TMI-1 calculation, C3320-85-001). When decay heat load is very low, as with the present plant conditions at TMI-1, heat loss to ambient is sufficient to provide adequate decay heat removal capability. This cooling method requires no active components but relies upon basic heat transfer principles.

5) This change requires equipment to be in operation only when needed to circulate reactor coolant in order to maintain the reactor coolant system in a subcooled condition. While STS Sections 3.4.1.3, 3.4.1.4 and 3.9.8.1 require equipment to be in operation allowing only a maximum of one hour down time with certain stipulations applied, this change allows equipment to be secured for longer periods where conditions permit.

   Depending on the decay heat generation rate, shutdown of the forced circulation equipment may allow reactor coolant system temperature to increase very slowly some time after shutdown compared to the conditions immediately following plant shutdown from power operation. During these conditions of slow temperature increase, continuous operation of a decay heat removal loop would not be necessary and suspension of operation would be permitted under 3.4.2.2.

6) This change recognizes the acceptability of cooldown by natural circulation as an acceptable means of decay heat removal. The adequacy of natural circulation cooldown as a stable means of decay heat removal is presented in TMI-1 FSAR Table 14.1-12.

7) This change allows a limited period of up to 7 days for which the requirement for operability of a redundant or diverse means of DHR may be suspended with reactor coolant temperature 250°F or less. This is to provide for preventive or corrective maintenance that may be necessary to ensure the continued reliability of the preferred means of DHR capability. This provides the assurance also that appropriate action will be taken to restore the preferred means of DHR capability to operable status in a timely manner, without prohibiting maintenance which is needed to decrease the likelihood of actual in-service failures. A period of up to 7 days is justified considering the low probability and minimal consequences of such a system failure.

8) This change recognizes that the DHR system is not the only system capable of providing a flow of borated cooling water through the reactor vessel below 250°F. In addition to the equipment allowed by STS in fulfilling the requirements for DHR capability, this change allows the use of a flow path from the BWST with BWST level greater than 44 ft. as an alternate flow path whenever such means are determined to be capable of maintaining RCS in a subcooled condition for at least 7 days. The length of time such an alternate flow path would be available for decay heat removal is predictable using calculations based on actual plant data or through plant testing at the time the system is to be declared operable.

   Surveillance requirements are included to verify the operability of the circulating path daily whenever the flow path is required to be operable.

These changes as discussed above are justified in that the requirements embodied in this TSCR provide for continuous decay heat removal capability, provide additional guidance, and specify a level of redundancy while allowing systems to be taken out of service for proper maintenance to be performed.

IV. NO SIGNIFICANT HAZARDS CONSIDERATIONS

These proposed changes provide additional operational requirements to assure redundant or diverse decay heat removal capability. Additional limiting conditions for operation and additional surveillance requirements are included. Therefore, operation of TMI-1 in accordance with this TSCR:

1) does not involve a significant increase in the probability or consequences of an accident previously evaluated,

2) does not create the possibility of a new or different kind of accident from any accident previously evaluated, and

3) does not involve a significant reduction in a margin of safety.

Therefore, significant safety hazards are not associated with this change.

V. IMPLEMENTATION

It is requested that the amendment authorizing this change become effective 120 days after receipt, to allow for the necessary procedural revisions to be put in place.

VI. AMENDMENT FEE (10 CFR 170.21)

Pursuant to the provisions of 10 CFR 170.21, a check for $150.00 was provided with our letter of March 8, 1985 as payment of the fee associated with TSCR No. 142.

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3.4 DECAY HEAT REMOVAL CAPABILITY

Applicability

Applies to the operating status of systems and components that function to remove decay heat when one or more fuel bundles are located in the reactor vessel.

Objective

To define the conditions necessary to assure continuous capability of decay heat removal.**

Specification

3.4.1 Reactor Coolant System temperature greater than 250°F.

3.4.1.1 With the Reactor Coolant System temperature greater than 250°F, three independent EFW pumps and associated flow paths shall be OPERABLE with:

a. Two EFW pumps, each capable of being powered from an OPERABLE emergency bus, and one EFW pump capable of being powered from an OPERABLE steam supply system. Specification 3.0.1 applies.

b. With one pump or flow path* inoperable, restore the inoperable pump or flow path to OPERABLE status within 72 hours or be in COLD SHUTDOWN within the next 12 hours. With more than one EFW pump or flow path* inoperable, restore the inoperable pumps or flow paths* to OPERABLE status or be subcritical within 1 hour, in at least HOT SHUTDOWN within the next 6 hours, and in COLD SHUTDOWN within the following 6 hours.

c. Four of six turbine bypass valves OPERABLE.

d. The condensate storage tanks (CST) OPERABLE with a minimum of 150,000 gallons of condensate available in each CST. With a CST inoperable, restore the CST to operability within 72 hours or be in at least HOT SHUTDOWN within the next 6 hours, and COLD SHUTDOWN within the next 30 hours. With more than one CST inoperable, restore the inoperable CST to OPERABLE status or be subcritical within 1 hour, in at least HOT SHUTDOWN within the next 6 hours, and in COLD SHUTDOWN within the following 6 hours. Specification 3.0.1 applies.

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*For the purpose of this requirement, an OPERABLE flow path shall mean an unobstructed path from the water source to the pump and from the pump to a steam generator.

**These requirements supplement the requirements of Sections 3.1.1.1.c, 3.1.1.2, 3.3.1 and 3.8.3.

3.4.1.2 With the Reactor Coolant System temperature greater than 250°F, all eighteen (18) main steam safety valves shall be OPERABLE or, if any are not OPERABLE, the maximum overpower trip setpoint (see Table 2.3-1) shall be reset as follows:

Maximum Number of Safety Valves Disabled on Any Steam Generator	Maximum Overpower Trip Setpoint (% of Rated Power)
1	92.4
2	79.4
3	66.3

With more than 3 main steam safety valves inoperable, restore at least fifteen (15) main steam safety valves to OPERABLE status within 4 hours or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3.4.2 Reactor Coolant System temperature 250°F or less.

3.4.2.1 With Reactor Coolant temperature 250°F or less, at least two of the following means for maintaining decay heat removal capability shall be OPERABLE and at least one shall be in operation except as allowed by Specifications 3.4.2.2, 3.4.2.3 and 3.4.2.4.

a. Decay Heat Removal String "A".

b. Decay Heat Removal String "B".

c. Reactor Coolant Loop "A", its associated OTSG, and its associated emergency feedwater flowpath.

d. Reactor Coolant Loop "B", its associated OTSG, and its associated emergency feedwater flowpath.

3.4.2.2 Operation of the means for decay heat removal may be suspended provided the core outlet temperature is maintained below saturation temperature.

3.4.2.3 The number of means for decay heat removal required to be operable per 3.4.2.1 may be reduced to one provided that one of the following conditions is satisfied:

a. The Reactor is in a Refueling Shutdown condition with the Fuel Transfer Canal water level greater than 23 feet above the reactor vessel flange.

b. Reactor coolant temperature is less than 140°F with BWST level greater than 44 feet and an associated flow path through the RCS OPERABLE such that core outlet temperature can be maintained subcooled for at least 7 days.

c. Equipment Maintenance on one of the means for decay heat removal specified by 3.4.2.1 is required and the equipment outage does not exceed 7 days.

3.4.2.4 Specification 3.4.2.1 does not apply when either of the following conditions exist:

a. Decay heat generation is less than 188 KW with the RCS full.

b. Decay heat generation is less than 100 KW with the RCS drained down for maintenance.

3.4.2.5 With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible.

Bases

A reactor shutdown following power operation requires removal of core decay heat. Normal decay heat removal is by the steam generators with the steam dump to the condenser when RCS temperature is above 250°F and by the decay heat removal system below 250°F. Core decay heat can be continuously dissipated up to 15 percent of full power via the steam bypass to the condenser as feedwater in the steam generator is converted to steam by heat absorption. Normally, the capability to return feedwater flow to the steam generators is provided by the main feedwater system.

The main steam safety valves will be able to relieve to atmosphere the total steam flow if necessary. If Main Steam Safety Valves are inoperable, the power level must be reduced, as stated in Technical Specification 3.4.1.2 such that the remaining safety valves can prevent overpressure on a turbine trip.

In the unlikely event of complete loss of off-site electrical power to the station, decay heat removal is by either the steam-driven emergency feedwater pump, or two half-sized motor-driven pumps. Steam discharge is to the atmosphere via the main steam safety valves and controlled atmospheric relief valves, and in the case of the turbine driven pump, from the turbine exhaust. (1)

Both motor-driven pumps are required initially to remove decay heat with one eventually sufficing. The minimum amount of water in the condensate storage tanks, contained in Technical Specification 3.4.1.1, will allow cooldown to 250°F with steam being discharged to the atmosphere. After cooling to 250°F, the decay heat removal system is used to achieve further cooling.

When the RCS is below 250°F, a single DHR string, or single OTSG and its associated emergency feedwater flowpath is sufficient to provide removal of decay heat at all times following the cooldown to 250°F. The requirement to maintain two OPERABLE means of decay heat removal ensures that a single failure does not result in a complete loss of decay heat removal capability. The requirement to keep a system in operation as necessary to maintain the system subcooled at the core outlet provides the guidance to ensure that steam conditions which could inhibit core cooling do not occur.

Limited reduction in redundancy is allowed for preventive or corrective maintenance on the primary means for decay heat removal to ensure that maintenance necessary to assure the continued reliability of the systems may be accomplished.

As decay heat loads are reduced through decay time or fuel off loading, alternate flow paths will provide adequate cooling for a time sufficient to take compensatory action if the normal means of heat removal is lost.

With the reactor vessel head removed and 23 feet of water above the reactor vessel flange, a large heat sink is available for core cooling. The BWST with level at 44 feet provides an equivalent reservoir available as a heat sink. Operability of the BWST is to be determined using calculations based on actual plant data or through plant testing at the time the system is to be declared operable. At such times that either of these means is determined to be operable, removal of the redundant or diverse cooling system is permitted.

Following extensive outages or major core off loading, the decay heat generation being removed from the Reactor Vessel is so low that ambient losses are sufficient to maintain core cooling and no other means of heat removal is required. The system is passive and requires no redundant or diverse backup system. Decay heat generation is calculated in accordance with ANSI 5.1-1979 to determine when this situation exists.

An unlimited emergency feedwater supply is available from the river via either of the two motor-driven reactor building emergency cooling water pumps for an indefinite period of time.

The requirements of Technical Specification 3.4.1.1 assure that before the reactor is heated to above 250°F, adequate auxiliary feedwater capability is available. One turbine driven pump full capacity (920 gpm) and the two half-capacity motor-driven pumps (460 gpm each) are specified. However, only one half-capacity motor-driven pump is necessary to supply auxiliary feedwater flow to the steam generators in the onset of a small break loss-of-coolant accident.

REFERENCES

(1) FSAR Section 10.2.1.3.

4.9 DECAY HEAT REMOVAL CAPABILITY - PERIODIC TESTING

Applicability

Applies to the periodic testing of systems or components which function to remove decay heat.

Objective

To verify that systems/components required for decay heat removal are capable of performing their design function.

Specification

4.9.1 Emergency Feedwater System - Periodic Testing (Reactor Coolant System Temperature greater than 250°F.)

4.9.1.1 Whenever the Reactor Coolant System temperature is greater than 250°F, the EFW pumps shall be tested in the recirculation mode in accordance with the requirements and acceptance criteria of ASME Section XI Article IWP-3210. The test frequency shall be at least every 31 days of plant operation at Reactor Coolant Temperature above 250°F.

4.9.1.2 During testing of the EFW System when the reactor is in STARTUP, HOT STANDBY or POWER OPERATION, if one steam generator flow path* is made inoperable, a dedicated qualified individual who is in communication with the control room shall be continuously stationed at the EFW local manual valves (See Table 4.9-1). On instruction from the Control Room Operator, the individual shall realign the valves from the test mode to their operational alignment.

4.9.1.3 At least once per 31 days each valve listed in Table 4.9-1 shall be verified to be in the status specified in Table 4.9-1, when required to be OPERABLE.

4.9.1.4 On a quarterly basis, verify that the manual control (HIC-849/850) valve station functions properly.

4.9.1.5 On a quarterly basis, EFV-30A and B shall be checked for proper operation by cycling each valve over its full stroke.

4.9.1.6 Prior to start-up, following a refueling shutdown or a cold shutdown greater than 30 days, conduct a test to demonstrate that the motor driven EFW pumps can pump water from the condensate tanks to the Steam Generators.

*For the purpose of this requirement, an OPERABLE flow path shall mean an unobstructed path from the water source to the pump and from the pump to a Steam Generator.

4.9.1.7 Acceptance Criteria

These tests shall be considered satisfactory if control board indication and visual observation of the equipment demonstrates that all components have operated properly except for the tests required by Specification 4.9.1.1.

4.9.2 Decay Heat Removal Capability - Periodic Testing (Reactor Coolant System Temperature 250°F or less).*

4.9.2.1 On a daily basis, verify operability of the means for decay heat removal required by specification 3.4.2 by observation of console status indication.

*These requirements supplement the requirements of 4.5.2.2 and 4.5.4.

Bases

The 31-day testing frequency will be sufficient to verify that the turbine driven and two motor-driven EFW pumps are operable and that the associated valves are in the correct alignment. ASME Section XI, Article IWP-3210 specifies requirements and acceptance standards for the testing of nuclear safety related pumps. Compliance with the normal acceptance criteria assures that the EFW pumps are operating as expected. The test frequency of 31 days (nominal) has been demonstrated by the B&W Emergency Feedwater Reliability Study to assure an appropriate level of reliability. In the case of the EFW System flow, the flow shall be considered acceptable if under the worst case single pump failure, a minimum of 500 gpm can be delivered when steam generator pressure is 1050 psig and one steam generator is isolated. A flow of 500 gpm, at 1050 psig head, ensures that sufficient flow can be delivered to either Steam Generator. The surveillance requirements ensure that the overall EFW System functional capability is maintained.

Daily verification of the operability of the required means for decay heat removal ensures that sufficient decay heat removal capability will be maintained.