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SUBJECT: Forwards evaluations for SEP Topics XV-1,XV-2,XV-3,XV-4, XV-5,XV-6,XV-7,XV-8,XV-10,XV-12,XV-14,XV-15 & XV-17 rel DBE-accidents & transients. 26 SEE Rpts.

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June 29, 1981

Docket No. 50-244
LS05-81-06-122

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Mr. John E. Maier
Vice President
Electric and Stream Production
Rochester Gas & Electric Corp.
89 East Avenue
Rochester, New York 14649

Dear Mr. Maier:

SUBJECT: GINNA - SEP TOPICS XV-1, XV-2, XV-3, XV-4, XV-5, XV-6, XV-7, XV-8, XV-10, XV-12, XV-14, XV-15, XV-17 (DESIGN BASIS EVENTS - ACCIDENTS AND TRANSIENTS)

Enclosed are topic evaluations for the SEP Topics on the accidents and transients listed above. Evaluations of the remaining events and of the radiological consequences of these events will be issued separately. These evaluations compare your facility with the criteria currently used by the regulatory staff for licensing new facilities.

Please inform us if your as-built facility differs from the licensing basis assumed in our assessment within 30 days of receipt of this letter.

These evaluations will be basic inputs to the integrated safety assessment for your facility unless you identify changes needed to reflect as-built conditions at your facility. These topic assessments may be revised in the future if your facility design is changed or if NRC criteria relating to these topics are modified before the integrated assessment is complete.

Sincerely,

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing

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TOPIC: XV-1 Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow and Inadvertent Opening of a Steam Generator Relief or Safety Valve.

15.1.1 & 15.1.2 Excessive Heat Removal Due to Feedwater System Malfunctions

Excessive heat removal, i.e., a heat removal rate in excess of the heat generation rate in the core, from the reactor coolant to the steam generator feedwater is caused by one of the following events:

(i) Feedwater system malfunction that results in a decrease in feedwater temperature.

(ii) Feedwater system malfunction that results in an increase in feedwater flow.

This group of accidents is analyzed to assure that the consequences of these moderate frequency events are acceptable as per criteria set by Standard Review Plan (SRP) Section 15.1. These two events discussed above are reviewed separately.

15.1.1 Decrease In Feedwater Temperature

Introduction

Excess heat removal can cause descrease in moderator temperature which increases core reactivity, leading to increase in power and decrease in shutdown margin. The increase in power can result in overpressurization of the primary and possibly lead to clad failures.

An example of excess heat removal by a decrease in feedwater temperature is the transient associated with the accidental opening of the condensate bypass valve which diverts feedwater flow around the low pressure feedwater heaters. In the

event of accidental opening of the bypass valve there would be a sudden reduction in feedwater inlet temperature causing an increase in subcooling of the primary. The increased subcooling will create a greater load demand on the primary and may lead to a reactor trip. The feedwater control system responds to keep steam generator level constant. However, if the control system does not function properly, the continued addition of cold feedwater may depressurize the primary system to the safety injection actuation setpoint. The safety injection signal will isolate the feedwater lines by venting the supply air to all feedwater control valves causing the valves to close, trip the main feedwater pumps, and close the feedwater discharge valves. The auxiliary feedwater system will take over heat removal needs until the reactor is cooled sufficiently to switch to the residual heat removal system.

Evaluation

Two cases were analyzed in Section 14.1.10 of the Ginna FSAR (Reference 1) to demonstrate the plant behavior in the event of a sudden feedwater temperature reduction resulting from accidental opening of the condensate bypass valve. The results were obtained by means of a detailed digital simulation of the plant including core kinetics, reactor coolant system, and the steam system. Both cases were assumed to occur from full power. The two cases analyzed are with automatic control by the rod control system and with no automatic control.

A zero moderator coefficient of reactivity was assumed for the uncontrolled case as this represents the condition where the plant has the least uncontrolled capability. For this case, the analysis showed a fairly rapid decrease in the reactor coolant average temperature and pressurizer pressure as the secondary heat extraction exceeds the core power generation. These parameters are summarized below in Table 1. The fixed low pressure trip would occur at 1880 psia at about 160 seconds from the initiation of valve opening. There is a considerable margin to DNB because of the accompanying large reduction in average coolant temperature. The DNBR at the time of trip is approximately 1.8.

The automatic controlled case was analyzed with a large negative moderator coefficient, which also acts to increase power. The core power increases and so it reduces the rate of decrease in coolant average temperature and pressurizer pressure. These parameters are tabulated below. The steady state conditions are reached with a minimum DNBR greater than 1.5. The plant would actually be tripped from the overpower protection set at 109% power. The trip setpoint used in the analysis is 118% power. The results of this analysis showed the maximum power attained in this transient was 115% power.

TABLE 1

Transient Response to Opening of

Condensate Bypass Valve

Parameters	No Control $\alpha_{mod} = 0$	With automatic control $\alpha_{mod} = -3.5x10-4$
1. Change in Pressurizer Pressure at 150 seconds	-450 psi	-35 psi
2. Change in Coolant Average Temperature at 150 seconds	-38°F	-4.6°F

Conclusions

As part of the SEP review of Ginna, the analysis of excessive heat removal events was reviewed against the specific criteria of Standard Review Plan (SRP) Section 15.1. Deviations from specific criteria are noted below:

1. The above analysis was made without reference to the specific computer code used for such analysis.
2. The transients were assumed to occur from full power for this analysis. According to SRP Section 15.1, the analysis should be done for the reactor initially at 102% of the rated core thermal power to account for a 2% power measurement uncertainty.
3. The analysis does not justify that the transient associated with the accidental opening of the condensate bypass valve is limiting.

However, the above deviations do not pose a safety problem since this event is bounded by the analyses performed for cycle 8 reload (Ref. 2).

15.1.2 Increase in Feedwater Flow

Introduction

The addition of excess feedwater is another means of increasing core power above full power. The overpower-overtemperature protection prevents any power increase which could lead to a DNBR less than 1.30.

Evaluation

The consequences of a step increase in feedwater flow to one steam generator from zero to full power flow at no load were analyzed in the Ginna FSAR (ref. 1). The calculations were based on conservative assumptions of constant feedwater temperature of 70°F, the most negative reactivity moderator coefficient assumed at the end of life. These calculations assume no credit for heat capacity of the reactor coolant system and steam generator shell thick metal. The maximum reactivity insertion rate was calculated to be 4.1×10^{-4} which is less than the maximum reactivity insertion rate analyzed for rod withdrawal from startup condition. If the accident occurs with the plant just critical at no load, the reactor will be tripped by the power range neutron flux level trip low setting set at approximately 25%. There is a large margin to DNB for the above calculated reactivity insertion rate.

The addition of cold feedwater after a reactor trip is interrupted by the actuation of safety injection on low pressurizer pressure and level. The safety injection signal will trip the main feedwater pumps and close the feedwater pump discharge valves as well as close the main feedwater control valves.

The licensee has evaluated the accidental opening of both feedwater control valves at full power. The consequences of this event were found to be less severe

than those resulting from the opening of the condensate bypass valve at full power as described in Section 15.1.1.

Conclusions

As part of SEP review of Ginna this topic has been evaluated against the criteria of SRP Section 15.1 and found to generally conform with the requirements of the SRP. The consequences of this event are bounded by those of Section 15.1.1.

15.1.3 Increase in Steam Flow

Introduction

A radid increase in steam generator steam flow causes a power mismatch between the reactor core power and the steam generator load demand. The reactor control system is designed to accomodate a 10% step load increase and a 5% per minute ramp load increase without a reactor trip in the range of 15 to 100% full power. Any loading rate in excess of these values may cause the reactor to be tripped by the reactor protection system. If the load increase exceeds the capability of the reactor coolant system, the transient is terminated in sufficient time to prevent the DNBR from decreasing below 1.3. An excessive load increase event could result from either an adminstrative violation such as excessive loading by the operator or an equipment malfunction in the steam bypass control or turbine speed control. In case of excessive loading by the operator or by system demand, the turbine load limiter limits maximum turbine load to 100% rated load.

During power operation, steam bypass to the condenser is controlled by reactor coolant conditions, i.e., abnormally high reactor coolant temperature indicates

a need for steam bypass. A single controller malfunction does not cause steam bypass because an interlock is provided which blocks the control signal to the valves unless a large turbine load decrease has occurred.

Increases in steam load to more than rated load are analyzed as steamline ruptures in FSAR. However, the reactor protection system will trip the reactor in time to prevent DNBR less than 1.30, regardless of the magnitude or rate of load increase.

Evaluation

Two cases have been analyzed in the FSAR to demonstrate the plant behavior in the event of excessive load increase. Both transients were assumed to occur from full power. These transients are (a) without automatic control and (b) with automatic reactor control. A zero moderator coefficient of reactivity was assumed for both cases which represents the condition where the plant has the least uncontrolled transient capability. The FSAR has presented results for 10% step increase in turbine load with and without automatic control. The results are similar but not as limiting as those of condensate bypass valve opening discussed in Section 15.1.1.

Without automatic control, the reactor coolant average temperature and pressurizer pressure decrease rapidly, as the secondary heat extraction exceeds the core power generation. The fixed low pressure trip occurs at about 150 seconds. At that time, reactor coolant temperature decreases by 45°F and pressurizer pressure decreases by 520 psi. There is a considerable margin to DNB because of the accompanying large reduction in coolant average temperature. The DNBR at the time of trip is approximately 1.8. The core power level remains essentially constant at full power.

The second case with automatic reactor control functioning is presented in the FSAR. The core power increases to about 112% of full power in 55 seconds before it levels off at 110% power at about 90 seconds. The increase in core power reduces the rate of decrease in coolant average temperature and pressurizer pressure. The average coolant temperature decreases by 3.3°F in about 30 seconds and then increases steadily showing an increase of 1°F at 100 seconds. The pressurizer pressure drops by 38 psi at 30 seconds and then increases sharply as much as 50 psi above normal pressure at 65 seconds. With no trip actuation steady state condition are reached with a minimum DNBR greater than 1.66.

Conclusions

The excessive load increase considered in this section will cause no radioactive release. Both transients show the same general behavior. A core power increase is accompanied by an average coolant temperature decrease and without a power increase there is a larger reduction in coolant average temperature. This has the effect of maintaining considerable margin to a limiting DNBR of 1.30.

As part of SEP review of Ginna this topic has been evaluated against the criteria of SRP Section 15.1.3, and found to conform with the requirements of the SRP.

15.1.4 Inadvertent Opening of Steam Generator Relief/Safety Valve

Introduction

A rupture of a steam pipe is assumed to include any accident which results in an uncontrolled steam release from a steam generator. The licensee has analyzed the effects of relief-safety valve opening as part of the spectrum of steam line

break. This particular event of inadvertent opening of the valve was considered as small line break in the FSAR (ref. 1). Since this transient is one of six limiting transients considered by the licensee, it was also reanalyzed for cycle 8 reload (ref. 2).

Evaluation

The transient analysis for the cycle 8 reload of the Ginna plant with Exxon Nuclear Fuel was performed using the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS PWR 2). The PTS PWR 2 code is a digital computer program developed to model the behavior of pressurized water reactors under normal and abnormal operating conditions. The analysis was done for a steam release rate of 273 lbs/sec assuming one loop operation at hot shutdown condition. The flow rate is based on the release from a steam generator safety valve since the safety valve was determined to have the largest venting capacity of any of the steam system valves in question. The minimum capability for boron injection was assumed corresponding to the most restrictive single failure in the safety injection system.

Additionally, the licensee assumed offsite power remains available and the most reactive control rod would remain withdrawn from the core. The small steamline break analysis showed that the worst expected shutdown margin at the end of cycle 8 is adequate to prevent return to criticality during such an event. The results presented in Reference 2 show the pressurizer pressure drops by 1300 psi in 180 seconds following steam line break. At that time, the reactor coolant temperature drops by approximately 80°F. Following the break, the maximum reactivity occurs at 175 seconds but does not reach criticality.

The Ginna plant would be protected from unacceptable consequences of this accident by several design features. The safety injection system would supply borated water and insert negative reactivity to the system. A low pressurizer pressure signal initiates the safety injection system. Borated water starts entering the injection lines after the pressurizer pressure has come down to the shutoff head (1400 psia)of the injection pumps. Borated water from the safety injection system reaches the core in 175 seconds. Provisions for isolation of the main steam system and feedwater system would reduce the severity of the accident by limiting the cooldown.

Results of the licensee analysis showed that the core does not go critical after the inadvertent opening of a steam generator safety valve.

The staff's review of this event is included in Amendment 19 to POL License No. DPR-18 (Reference 3). The review states that the consequences of a steam line break are acceptable for large break even though the plant would return to criticality. The small steam line break does not return to criticality. The staff has reviewed the PTS PWR 2 code and found its use acceptable (Ref. 3) for determining margins to the peak linear heat generation rate and departure from nucleate boiling design limits.

Conclusions

As part of the SEP review of Ginna, the analysis has been reviewed against the criteria of SRP Section 15.1.4. Based on our review of initial conditions and assumptions in the analysis, we conclude that the licensee analysis is in conformance to licensing criteria.

REFERENCES

1. R.E. Ginna Nuclear Power Plant, Unit 1, Final Facility Description and Safety Analysis Report, Rochester Gas and Electric Corporation.

2. XN-NF-77-40, "Plant Transient Analysis for the R.E. Ginna, Unit 1, Nuclear Power Plant," November 1977.

3. Amendment No. 19 to Provisional Operating License No. DPR-18, Letter from D.L. Ziemann (NRC) to L.D. White, Jr., (R G & E) dated May 1, 1978.

TOPIC: XV-2 Spectrum of Steam System Piping Failures (PWR)

Steamline Break Inside Containment

Introduction

A steamline break in the secondary system results in an initial increase in steam flow which increases heat removal from the primary coolant system. This increased heat removal from the reactor coolant system causes a reduction in the coolant temperature and pressure. Reactor power increases because of the negative moderator temperature coefficient of reactivity feedback from the cooldown. This cooldown results in an insertion of positive reactivity, which could cause a return to criticality even after the reactor is scrammed. If the most reactive rod is assumed stuck in the withdrawn position there is an increased possibility that the reactor will become critical and return to power without the addition of negative reactivity to the core.

Evaluation

Assuming the most pessimistic combination of circumstances which could lead to power generation following a steam line break, the core is ultimately shut down by the boric acid in the Safety Injection System. The following systems in the Ginna plant provide the necessary protection to mitigate the consequences of a steamline rupture:

(1) Safety Injection System actuation on a) two out of three pressurizer low pressure signals, b) two out of three low pressure signals in any steam line, and c) two out of three high containment pressure signals.

(2) Reactor trip upon receiving high neutron flux signal, overtemperature ΔT signal, or upon actuation of the safety injection system.

(3) Redundant isolation of the main feedwater lines. In addition to the normal control action which will close the main feedwater valves, any safety injection signal will rapidly close all feedwater control valves, trip the main feedwater pumps, and close the feedwater pump discharge valves.

(4) Trip of the fast acting steam line isolation valves designed to close in less than 5 seconds with no flow on (a) one out of two high steam flow signals in that steam line in coincidence with any safety injection signal, and (b) two out of three high containment pressure signals.

Each steam line has a fast closing isolation valve and a check valve. These four valves (two for each steam generator) prevent blowdown of more than one steam generator for any break location even if one valve fails to close.

A spectrum of pipe breaks with various combinations of break sizes and initial plant conditions was analyzed in Reference 1. The following seven combinations were considered:

(A) Complete severance of a pipe outside the containment, downstream of the steam flow measuring nozzle (1.4 ft^2) at initial no load conditions with offsite power available and two loops in operation.

(B) Complete severance of a pipe inside the containment (4.37 ft^2) at the outlet of the steam generator at initial no load conditions with outside power available and two loops in operation.

(C) Case (A) above with only one loop in operation.

(D) Case (B) above with only one loop in operation.

(E) Case (A) above with loss of offsite power simultaneous with the steam break.

(F) Case (B) above with loss of offsite power simultaneous with the steam break.

(G) A break equivalent to steam release through one steam generator safety valve with offsite power available.

All above cases assumed initial hot shutdown conditions with the rods inserted (except for one stuck rod) at time zero. The steamline break at hot zero power condition is the worst case since the steam generator secondary side water inventory is maximum at this time, prolonging the duration and increasing the magnitude of the primary loop cooldown.

The analysis did not specifically account for auxiliary feedwater. However, the steam generator heat transfer code, using constant heat transfer coefficients, continued to calculate heat transfer from the primary to the secondary side after the broken steam generator had been estimated to be empty. If auxiliary flow was specifically accounted for, its effect would be negligible during the initial portion of the transient and would have minimal effect during later portions of the transient since by the time the broken steam generator empties, the total system reactivity is negative and core power is decreasing.

The limiting case based on the results of previous analysis (Reference 1) was identified and analyzed for cycle 8 reload (Reference 2). The latest analysis (cycle 8) considered the double ended guillotine rupture of a steam line inside containment. The analysis of the limiting MSLB was performed using the PTS PWR 2 (Reference 3) code for a break at hot standby conditions with offsite power available. Major assumptions include taking credit for minimum boron injection capacity and assuming the most reactive control rod stuck out of the core on reactor scram.

The results of the analysis indicate the plant reaches a peak average core power of 22% of rated power approximately 90 seconds after accident initiation. The minimum DNBR, determined using the modified Macbeth critical heat flux correlation, was 1.58.

The main steamline break analysis was reviewed and approved by the staff (Reference 4) without generic approval of the analytical methods. Approval of the plant specific analysis recognized the margin to DNB and the conservatism of the scram characteristics assumed in the analysis.

The effects of a postulated MSLB on other systems consistent with the intent of APCSB 3-1 and MEB 3-1 is addressed in SEP Topics III-5A and III-5B. Analysis of the containment response to a postulated MSLB, which may require an analysis considering different principal assumptions and/or single failures is considered under SEP Topics VI-2D and VI-3.

Conclusions

As part of the SEP review of Ginna, the MSLB analysis was reviewed against the criteria of SRP Section 15.1.5. The initial conditions, core kinetics,

power level, and operating conditions have been reviewed and found to generally conform with the requirements of the SRP.

Steamline Break Outside Containment

Introduction, Evaluation, and Conclusions

The rupture of a steam line break outside containment has been determined by the licensee (Reference 1) to have less severe consequences on the primary system than a MSLB inside containment. This is due to the fact that the steamlines have nozzles located inside containment. The normal function of the nozzles is to measure steam flow, but under accident conditions, the nozzles act as flow restrictors for breaks outside containment. By limiting the flow rate from the broken loop steam generators for outside containment pipe breaks, the nozzles reduce the cooldown of the primary system.

REFERENCES

1. Rochester Gas & Electric Corporation Proposed Change to Shutdown Margin Requirements in the Ginna Nuclear Plant Technical Specifications, Amendment Application dated September 22, 1975.

2. XN-NF-77-40, "Plant Transient Analysis for the R. E. Ginna, Unit 1, Nuclear Power Plant," November, 1977.

3. Kahn, J. D., Description of the Exxon Nuclear Power Plant Transient Simulation Model for Pressurized Water Reactors (PTS PWR), XN-74-5, Revision 1, May 1975.

4. Amendment No. 19 to Provisional Operating License No. DPR-18, Letter from D. L. Zieman (NRC) to L. D. White, Jr., (R G & E) dated May 1, 1978.

TOPIC XV-3: LOSS OF EXTERNAL LOAD, TURBINE TRIP, LOSS OF CONDENSER VACUUM, CLOSURE OF MAIN STEAM ISOLATION VALVE (BWR), AND STEAM PRESSURE REGULATOR FAILURE (CLOSED)

LOSS OF EXTERNAL LOAD

Introduction

Loss of external electrical load, due to a spectrum of electrical system conditions, is effected by the opening of the main generator breaker. The plant design does not provide a direct reactor trip from the breaker opening circuit. In the event of total loss of the electrical load the reactor will be tripped by high pressurizer pressure or high pressurizer level signal in the reactor trip system.

Review Criteria

The review criteria for the transients, such as loss of load, that result in an unplanned decrease in heat removal by the secondary system are presented in the SRP Section numbered 15.2.1, 2, 3, 4, 5.

Evaluation

The licensee has performed analyses of bounding conditions for total loss of load at beginning-of-life and end-of-life of the core (Ref. 1). The initial conditions include reactor power, coolant temperature and pressure all at maximum values with the plant at full power (102%) which leads to maximum power difference and minimum margin to core protection limits at the initiation of the loss of load.

First, the reactor control system was assumed in the normal automatic mode with the control rods in the minimum incremental worth region and the most reactive rod held out of the core. The steam bypass to the condenser was assumed unavailable while credit is taken for the effects of the pressurizer spray and relief valves in reducing or limiting coolant pressure, thus delaying the high pressurizer pressure reactor trip. Credit is also taken for the effects of control rod insertion by the reactor control system.

The results of the analysis, with the reactor in automatic control, indicate that the integrity of the core is maintained by high pressurizer pressure reactor trip in 12 seconds with peak pressure of 2511 psia and a minimum DNB ratio of 1.83, reached in 13 seconds. However, the licensee's analysis and results are limited to plant response only during the transient. A single failure in the systems required for the long term mitigation and decay heat removal from the core is not identified.

The licensee was requested to provide a single failure analysis to determine the limiting failure concurrent with a turbine trip (Ref. 2), that may affect the ability of the plant to mitigate the consequences of the transient. The

turbine trip event bounds the loss of load event, and therefore, the analysis with the worst single failure should demonstrate the adequacy of the plant design to mitigate the consequences of the loss of load transient.

The licensee has reported (Ref. 3) that a single failure analysis is not available and that the objective of the turbine trip transient analysis is to show that the primary pressure relieving devices can limit the pressure to acceptable levels, and that no core damage occurs during the transient.

The SEP staff evaluated the ability of the Ginna plant to reach safe shutdown assuming a loss of offsite power and a single active failure under Topic VII-3. Once the transient effects of overpressurization are over, the results of the above evaluation can be applied and the single failure criterion of the SRP satisfied.

The licensee has also analyzed the loss of load transient with the plant operating at full power in manual control. Credit is not taken for control rod insertion, pressurizer spray, relief valves or steam bypass.

The results of this analysis, with the reactor in manual control, indicate the reactor also tripped by high pressurizer pressure with peak pressure of 2415 psia. The minimum DNB ratio is not reported, however, it is greater than 1.3 based on a comparison with the case of automatic control.

Conclusions

As part of the SEP review for Ginna we have evaluated the analysis of the loss of external load and the licensee's response to our request for additional information (Ref. 3) against the criteria of SRP Section 15.2.1. Based on this evaluation we conclude that the criteria are satisfied.

TURBINE TRIP

Evaluation

The licensee has performed an analysis of the consequences of an instantaneous turbine trip by closure of the turbine stop valves. However, the event is identified as a loss of load transient (Ref. 3).

The turbine trip event is different from the loss of external load in that the fast closure of the turbine stop valves causes an abrupt interruption of steam to the turbine, creating a more severe overpressure condition in the primary system.

The licensee has analyzed the turbine trip transient with the plant operating at full power (102%) and in manual control. Credit is not taken for pressurizer spray, relief valves or steam bypass to the condenser.

The results of the analysis indicate that the reactor is tripped by high pressurizer pressure reactor trip in about 12 seconds, the mimimum DNB ratio 1.83 is reached in about 13 seconds and the pressurizer reaches a pressure of 2511 psia following the reactor trip. These results are very conservative for this plant since the licensee has in place a simultaneous turbine trip-reactor trip (anticipatory trip) associated with turbine stop valve closure. The reactor trip, therefore, would be effected in less than one second following turbine trip, as opposed to the 12 seconds assumed in the analysis, and would result in less severe conditions to the core and vessel.

The discussion presented under Loss of External Load on single failure considerations is also applicable to this event.

Conclusions

As part of the SEP review for Ginna we have evaluated the analysis of turbine trip and the licensee's response to our request for additional information (Ref. 3). Based on this evaluation we conclude that the criteria of SRP Section 15.2.1 are met.

LOSS OF CONDENSER VACUUM

Evaluation

Loss of condenser vacuum can be effected from failure of the circulating water system or excessive air leakage through turbine gland packing. In the event of loss of condenser vacuum the turbine will be tripped and therefore this event is bounded by the turbine trip event coincident with loss of condenser vacuum.

Conclusions

A turbine trip event coincident with loss of condenser vacuum has been analyzed by the licensee and the results of the analysis have been evaluated against the criteria of SRP Section 15.2.1. Our conclusions are included in the evaluation for the turbine trip event.

REFERENCES

(1) XN-NF-77-40, "Plant Transient Analysis for the R. E. Ginna, Unit 1, Nuclear Power Plant," November, 1977.

(2) Letter from D. Ziemann (NRC) to L.D. White (RG&E) dated February 25, 1980.

TOPIC XV-4: LOSS OF NON-EMERGENCY A-C POWER TO THE STATION AUXILIARIES

Introduction

Loss of all a-c power to station auxiliaries, while the plant is at power, will cause loss of main feedwater and thus, loss of capability of the secondary system to remove heat generated in the primary. Additional power generation, however, will be abruptly interrupted as the reactor is tripped by a fast undervoltage or underfrequency reactor trip signal, generated from loss of a-c power. Forced coolant circulation in the primary will also be interrupted as a result of loss of power to the reactor coolant pumps and natural circulation would have to be relied upon to carry the coolant through the steam generators for removal of the core decay heat. Therefore, a source of feedwater (auxiliary) will be required to be made available to remove the decay heat. A source of primary coolant make-up supply may also become necessary to maintain the required inventory level in the primary, if inventory is lost from actuation of pressurizer relief valves.

Evaluation

The licensee has performed an analysis of the consequences of loss of all a-c power to the station auxiliaries (Ref. 1). The analysis assumes the loss of normal feedwater on the secondary side and loss of forced circulation of the primary coolant. Both normal feedwater and forced circulation are supplied motive power by the non-emergency a-c power system. Hence, loss of that power supply results in the loss of both systems.

The principal discussion on the method of analysis for this transient appears in the analysis for loss of normal feedwater (Ref. 1). In that analysis the loss of feedwater is assumed coincident with natural circulation in the primary coolant system.

The initial conditions for the analysis include the reactor at full power and the steam generators at the lowest water level during reactor trip. The steam bypass to the condenser was assumed unavailable and credit is taken for only one motor driven auxiliary feedwater pump actuated automatically one minute after the event.

The results of the analysis indicate that following reactor trip and turbine trip, the steam generator water level will not reduce below the level at which sufficient heat transfer area is available to dissipate core decay heat without water relief from the primary relief, or safety valves.

Conclusions

As part of the SEP review for Ginna the analysis has been evaluated against the criteria of SRP Section 15.2.6 and we have concluded that it is in conformance with these criteria.

REFERENCES

(1) R. E. Ginna Nuclear Power Plant, Unit 1, Final Facility Description and Safety Analysis Report, Rochester Gas and Electric Corporation, as supplemented.

TOPIC: XV-5 Loss of Normal Feedwater Flow

Introduction

A loss of normal feedwater could be caused by pump failures, valve malfunctions, or a loss of offsite power. The result of this loss of feedwater flow would be increasing reactor coolant temperature and pressure due to a decrease in heat removal and decreasing water level in the steam generator. If the reactor were not tripped during the accident, fuel damage could possibly occur from a sudden loss of heat sink.

Evaluation

Reactor protection is provided by (a) trip on low steam generator water level, (b) trip on steam flow-feedwater flow mismatch in coincidence with low water level in either steam generator, (c) two motor driven auxiliary feedwater pumps, and (d) one turbine driven pump. The motor driven pumps are automatically initiated by 2 of 3 low-low steam generator level in either steam generator, trip of both main feedwater pumps, or safety injection initiation. The steam admission valve to the turbine driven pump is automatically opened on 2 out of 3 low-low water level in both steam generators or loss of voltage on both 4kv buses. Ginna also has a standby auxiliary feedwater system (SBAFWS) with two motor driven pumps which are powered from emergency buses. The SBAFWS is manually actuated. The motor driven auxiliary feedwater pumps are supplied by the diesel if a loss of offsite power occurs and the turbine driven pump utilizes steam from the secondary systems.

The analysis, as discussed in the FSAR, has been performed to show that following a loss of normal feedwater, the auxiliary feedwater system is adequate to remove stored and residual heat to prevent water relief through the pressurizer relief valves. Conservative initial conditions (102% initial power, low steam generator water level, only one motor driven auxiliary feedwater pump available at one minute after the accidents, and low heat transfer coefficient in the steam generator) were assumed.

The results of the analysis in the FSAR indicate that at no time is the tube sheet uncovered in the steam generator receiving auxiliary feedwater flow and at no time is there water relief from the pressurizer. The capacity of one motor driven auxiliary feedwater pump is such that the water level in the steam generator being fed does not recede below the lowest level at which sufficient heat transfer area is available to dissipate core residual heat without water relief from the primary system relief or safety valve.

Conclusions

As part of our SEP review for Ginna the analysis has been evaluated against the criteria of SRP section 15.2.7 and found to generally conform with the requirements of the SRP.

This event is bounded by the analysis performed for loss of Non-Emergency A-C power to the station auxiliaries (SEP Topic XV-4).

TOPIC XV-6: FEEDWATER SYSTEM PIPE BREAKS

Introduction

A feedwater line break can result in either a reactor system cooldown (such as that from a steamline break) or a reactor coolant system heatup (by reducing feedwater flow to the affected steam generator). Feedwater break cooldowns would be less severe than those analyzed in the steamline rupture analysis. Thus, the analysis has been performed to demonstrate that the system is capable of sustaining a feedwater line rupture under initial conditions and assumptions which result in the most severe heatup of the primary system.

Evaluation

The licensee performed an analysis of a heatup resulting from a feedwater line rupture in Reference 1. This event was not part of the original licensing basis for this facility, which was submitted in the FSAR (Ref. 2).

The primary system transients resulting from the hypothetical double ended rupture of a main feedwater line was analyzed. Conservative initial conditions (102% initial power, only one motor driven auxiliary feedwater pump available at ten minutes after reactor trip to allow sufficient time for operator realignment of the auxiliary feedwater system i.e., isolation of AFW to the affected steam generator, a conservative core residual heat generation, a zero quality blowdown, low heat transfer coefficient, reactor trip after all steam generator liquid emptied) were assumed.

Reactor protection in the event of a feedwater line break is provided by (a) reactor trip, (b) two motor driven auxiliary feedwater pumps, (c) one turbine driven auxiliary feedwater pump, and (d) a standby auxiliary feedwater system (SBAFWS) with two motor driven pumps which are powered from emergency buses. The SBAFWS is manually actuated. The motor driven auxiliary feedwater pumps are supplied by the diesel if a loss of offsite power occurs and the turbine driven pump utilizes steam from the secondary systems.

The results of the analysis in Reference 1 demonstrated that under the severely limiting assumptions used to maximize the time delay to reactor trip and minimize heat removal from the steam generator blowing down, the system is capable of removing decay heat following the blowdown without exceeding the safety limit for reactor coolant system pressure. Maximum reactor coolant pressure remains below 2600 psia which is well below the SRP Section 15.2.8 guidelines of 110% of design pressure since the design pressure of Ginna is 2500 psia. Assuming one of the motor driven pumps fails to operate and neglecting the pumping capacity of the turbine driven auxiliary feedwater pump, the energy removal capability of the secondary system exceeds the residual energy generation in the primary system within 30 minutes of the occurrence of the assumed rupture. At no time does the reactor coolant thermal expansion rate exceed the pressurizer safety valve capacity. Therefore, no overpressurization of the reactor coolant system will occur. Total discharge of liquid from the reactor coolant system through the pressurizer safety valve is 1394 ft^3 due to thermal expansion, or less than 25% of the initial reactor coolant system liquid volume. Therefore, the core remains flooded.

The licensee has indicated that the MARVEL code was used for this analysis, although the specific revision was not known. The MARVEL code has been used extensively by Westinghouse for plant transient analysis. The code is still under staff review, however, the staff has previously accepted results obtained with this code in licensing actions.

Conclusions

As part of our SEP review for Ginna the analysis has been evaluated against the criteria of SRP Section 15.2.8 and found to generally conform with the requirements of the SRP. Therefore, the staff concludes that the analysis of this event is acceptable.

REFERENCES

1. Letter from K. W. Amish (RG&E) to J. F. O'Leary (NRC) dated May 24, 1974.

2. R. E. Ginna Nuclear Power Plant, Unit 1, Final Facility Description and Safety Analysis Report, Rochester Gas and Electric Corporation, as supplemented.

TOPIC XV-7: REACTOR COOLANT PUMP ROTOR SEIZURE AND REACTOR COOLANT PUMP SHAFT BREAK

LOSS OF FORCED COOLANT FLOW

Introduction

The loss of forced coolant flow event considers the cases of both a partial and complete loss-of-coolant flow. A loss of forced coolant flow could result from mechanical or electrical failures in one or both of the reactor coolant pumps or from failures in the power supply to the pumps. A decrease in reactor coolant flow while the reactor is at power would result in a degradation of core heat transfer and an increase in the primary coolant temperature due to reduced heat transfer in the steam generators. This increase in primary coolant temperature could result in DNB and subsequent fuel damage if adequate safety features are not provided.

Evaluation

The licensee provided the latest analysis for a loss of forced coolant flow event in Reference 3. This analysis, performed using the PTSPWR2 code, (Ref. 1) considered only the case of two pumps coasting down which was determined to be the limiting case from the results of previous analysis (Ref. 2). The licensee assumed beginning-of-cycle values for the moderator and fuel coefficients with a factor of 1.2 applied to the Doppler coefficient. The most reactive rod was assumed stuck out of the core following reactor scram.

The Ginna plant is protected against the consequences of this event by the reactor protection system which scrams the reactor on a variety of signals which are specifically related to the loss of forced coolant flow. The reactor trips of special interest are the reactor trip on either undervoltage or underfrequency at the bus supplying power to the pumps and the reactor trip on low flow. The low flow trip is designed, for most Westinghouse plants, to provide suitable protection in the event of the loss of a single reactor coolant pump. The undervoltage/underfrequency scram signal provides additional protection in the event both pumps fail since the signal would be generated more quickly than the low flow derived signal. The licensee, however, has conservatively assumed in the analysis that the reactor scrams on the low flow signal.

The results of the analysis indicate a minimum DNBR of 1.61 is reached 4.7 seconds after the accident initiation. The staff approved the analysis performed by the licensee (Ref. 3) in Reference 4. As part of this evaluation the staff also approved the consideration of only the case where both pumps are assumed to coast down.

Conclusions

As part of the SEP review of Ginna we have reviewed the analysis of a loss of forced coolant flow against the specific criteria of SRP Section 15.3.1. We conclude that the acceptance criteria are satisfied.

REACTOR COOLANT PUMP ROTOR SEIZURE

Introduction

In the event of a reactor coolant pump rotor seizure the primary coolant flow through the affected loop will be abruptly interrupted and the loss of flow could cause severe overheating in the primary system. The other pump would continue to circulate coolant through the unaffected loop. However, if offsite power is lost the other pump will be tripped and its contribution will be limited to only coastdown flow, thus making the accident more severe. The coastdown flow is available from the energy stored in the inertia of the pump's flywheel. For the pump seizure event the reactor will be tripped by the loss of flow signal in the reactor protection system.

Evaluation

The licensee has performed an analysis of a locked pump rotor (Ref. 3). The initial conditions include the reactor at rated power with begining of cycle kinetics and 1.2 multiplier applied to the Doppler coefficient. The steam bypass to the condenser was assumed unavailable and the effects of the pressurizer spray and relief valves in reducing or limiting coolant pressure were neglected. Also, the feedwater pumps were assumed to be tripped. The unaffected main coolant (RCP) pump, however, is assumed to continue to circulate coolant through the unaffected loop during the transient, without loss of offsite power.

The results of the analysis indicated that the reactor is tripped by the loss of flow reactor trip in .8 seconds with an increase in system temperature and pressure of 13°F and 53 psia respectively. The minimum DNB ratio was calculated to be 1.23 with less than 1% of the fuel rods failed.

Standard Review Plan Section 15.3.3 requires that the analysis of a locked rotor event consider loss of offsite power and coastdown.

We believe that an analysis assuming loss of offsite power and coastdown of the unaffected pump, will result in DNB ratio lower than the licensee has calculated (1.23) and could give rise to more than 1% fuel rods failure.

The licensee has provided a comparison of the results for a locked rotor event with and without coastdown of the second pump. Assuming an instantaneous loss of offsite power, the flow is reduced approximately 4% at the time of minimum DNB ratio, and the DNBR is decreased by only 0.1. Although some additional clad failures may result, the core cooling capability should not be affected.

This evaluation was performed with the RETRAN code. The licensee also provided a comparison of the predictions for a loss of flow event for RETRAN and the Exxon code used to perform the reference analysis. The results showed good agreement, so the above sensitivity can be applied to the reference case. Both RETRAN and the Exxon code are under staff review, but results calculated by the codes have been previously accepted in licensing evaluations.

Conclusion

As part of the SEP review of Ginna we have evaluated the analysis of the locked pump rotor event against the criteria of SRP Section 15.3.3. Based on this evaluation we concluded that the locked rotor event has been adequately considered by the licensee.

REACTOR COOLANT PUMP SHAFT BREAK

Evaluation

In the event of a reactor coolant pump shaft break, the sudden decrease in coolant flow will result in degradation of core heat transfer and will cause some fuel damage. The decrease in coolant flow from this event is not as severe, early in the transient, as is the decrease in flow from the bounding event of the pump rotor seizure. Early during the pump shaft break transient the impeller of the affected pump will continue to rotate in the forward direction aiding the circulation of the coolant. However, the flow through the affected loop will reverse later during the transient and will result in a lower core flow rate.

The licensee has not performed an analysis of the pump shaft break transient. However, since the core flow, early during this transient, is greater than the flow in the pump rotor seizure event the consequences are less severe that the pump rotor seizure. The consequences later during the transient, when the impeller reverses direction may be more severe than the pump rotor seizure event. Since the loss of flow events are thermally limiting early in the transient, lower steady-state core flow should not adversely affect long-term core cooling.

Conclusion

As part of the SEP review of Ginna, we have evaluated the reactor coolant pump shaft break event and based on this evaluation we concluded that analyses are not required.

REFERENCES

1. Kahn, J. D., Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTSPWR), XM-74-5, Revision 1, May 1975.

2. R. E. Ginna Nuclear Power Plant, Unit 1, Final Facility Description and Safety Analysis Report, Rochester Gas and Electric Corporation, as supplemented.

3. XN-NF-77-40, "Plant Transient Analysis for the R. E. Ginna, Unit 1, Nuclear Power Plant," November, 1977.

4. Amendment No. 19 to Provisional Operating License No. DPR-18, Letter from D. L. Ziemann (NRC) to L. D. White, Jr., (RG&E) dated May 1, 1978.

R. E. GINNA

TOPIC XV-8, CONTROL ROD MISOPERATION

Introduction

Control rod misoperation can occur through operator error or malfunction of the rod control system. Situations considered are rod misalignment, rod withdrawal, and rod drop. These events can result in power distribution and reactivity changes.

Review Criteria

The guidance presently used by the staff for evaluating these events is presented in Standard Review Plan Sections 15.4.1, 15.4.2 and 15.4.3. The criterion to be satisfied is that fuel thermal limits not be exceeded, i.e. that the departure from nucleate boiling (DNB) ratio is greater than 1.3.

Evaluation

Rod Misalignment

Rod misalignment can lead to changes in power distributions and local peaking. During the operating license review for Ginna the staff performed a review of rod misalignment that led to power distribution control technical specifications, such as quadrant tilt limits, that have since been applied to all Westinghouse reactors. The staff concludes that the analysis performed and the provisions made at the time are still acceptable.

Rod Withdrawal

An uncontrolled rod group withdrawal at power is a reactivity addition which results in an increase in the core heat flux and power level. Since heat extraction from the steam generator lags behind the core power increase until the steam generator pressure increases to the relief or safety valve setpoint, there is an increase in the primary coolant temperature.

The licensee has performed analyses of an uncontrolled rod withdrawal (Reference 1) using the PTSPWR2 Code to demonstrate the adequacy of the reactor protection system. The analysis was done assuming both a fast rod withdrawal (6. X 10^{-4} ΔK/sec) and a slow rod withdrawal (50. X 10^{-6} ΔK/sec) at 100% power. The reactivity insertion rates chosen were based on the rates determined to be limiting from previous analysis (Reference 2).

The analyses performed by the licensee show that for the high reactivity insertion rate the reactor is tripped on reaching the high neutron flux trip setpoint. For this case the DNBR decreased from 2.00 initially to 1.77, and the primary system pressure increased from 2220 psia to 2235 psia with the core average temperature increasing by less than 1°F. As can be seen from the results the neutron flux level in the core rises rapidly for the high reactivity insertion rate while the core heat flux and primary coolant temperature lag behind due to the thermal capacitance of the fuel and primary system coolant.

For the case assuming a slow reactivity insertion rate the reactor protection system scrams the reactor upon first reaching the overtemperature ΔT trip setpoint. The results of analysis for slow rod withdrawal demonstrate a similar margin to DNB as for fast rod withdrawal with the reactor trip occurring after a longer period. In the event of a slow rod withdrawal the core heat flux remains more closely matched to the neutron flux. Thus by the time the trip setpoint is reached the primary system pressure and temperature will have experienced a greater increase although remaining below acceptable limits.

For lower initial power conditions, the margin to DNB is only slightly lower, DNBR of 1.69 for 60% initial power. This analysis was reviewed and accepted by the staff for Cycle 8 operation of the reactor. The reload involved a change of fuel supplier, who used methods only recently developed for analysis of PWR transients. These methods are used for a number of Westinghouse designed reactors, and are considered current. Because of these factors, the staff concludes that the uncontrolled rod withdrawal analysis is acceptable.

Control Rod Drop

A control rod drop results in a decrease in reactor power, pressure and temperature. Depending on the worth of the rod, thermal limits can be approached because of the distorted power distribution. If the rod control system is controlling rod motion, other control rods may be withdrawn in response to the decreasing power and temperature, which exacerbates the maldistributions caused by the dropped rod.

Rod drop protection at Ginna is provided by a turbine runback and rod block. These features act to prevent adverse thermal consequences upon detection of rod bottom lights, indicating a fully inserted rod. The rod block prevents withdrawal of other control rods, and the turbine runback reduces generator load, and thus the temperature and pressure drop.

The licensee has provided (Ref. 2) an analysis that shows that for a rod drop event, these protection features ensure margin to departure from nucleate boiling (DNB). The staff reviewed and approved that analysis during the Ginna licensing review. However, the staff also noted that later generation Westinghouse reactors have provided a safety grade negative flux rate trip for rod drop event protection and that for the older Westinghouse plant, such as Ginna, the existing protection is not safety grade. THis question as to the adequacy of the rod block turbine runback protection for rod drop events is being pursued generically.

Conclusion

Based on this review, the staff concludes that the analyses of these events is acceptable and that Topic XV-8 is complete. If the staff concludes that additional protection is necessary as a result of the generic review of rod drop protection for the older Westinghouse plants it will be handled on a case by case basis, separate from the SEP.

REFERENCES

1. "Plant Transient Analysis for the R. E. Ginna Unit 1 Nuclear Power Plant", XN-NF-77-40, November, 1977.

2. R. E. Ginna Nuclear Power Plant Unit 1, Final Facility Description and Safety Analysis Report, Rochester Gas and Electric Corportation, as supplemented.

TOPIC XV-10: CHEMICAL AND VOLUME CONTROL SYSTEMS MALFUNCTION THAT RESULTS IN A DECREASE IN BORON CONCENTRATION IN THE REACTOR COOLANT (PWR)

Introduction

The Chemical and Volume Control System (CVCS) supplies reactor makeup water to the reactor coolant system via the reactor makeup control system. A boric acid blend system is provided to permit the operator to match the boron concentration of reactor coolant makeup water to that in the reactor coolant system.. Reactivity can, therefore, be added to the reactor coolant if the boron dilution process becomes uncontrolled.

Review Criteria

The review criteria for boron dilution events is presented in SRP Section 15.4.6. Since operator action is generally relied upon to terminate the event, minimium time intervals of 30 minutes during refueling and 15 minutes during other operational modes, must be available from the time an alarm alerts the operator to loss of shutdown margin.

Evaluation

The licensee has performed an analysis for an uncontrolled boron dilution event (Ref. 1). The analysis assumes plant modes of refueling, startup and power operation. The rate of addition of unborated water makeup is limited to the capacity of the makeup water pumps at the maximum combined flow of 120 gpm for both pumps.

DILUTION DURING REFUELING

The plant conditions for this event include one residual heat removal pump operating, boron concentration of 2000 ppm, all control rods inserted, a minimum water volume in the reactor coolant system of 2724 ft^3 (this corresponds to the volume necessary to fill the reactor vessel above the nozzles) and the maximum dilution flow of 120 gpm.

The results of the analysis indicate that the boron concentration must be reduced from 2000 ppm to approximately 1080 ppm before the reactor will go critical, and it would require 1.75 hours to reach criticality. The operator, therefore, will have sufficient time to recognize the high neutron count rate alarms in the containment and the control room and isolate the reactor makeup water source by closing valves and stopping the pumps.

DILUTION DURING STARTUP

The plant conditions for this event include the reactor coolant system filled with borated water at 2000 ppm, reactor coolant pumps are operating and the volume of the reactor coolant is approximately 5247 ft^3 which is the reactor coolant system excluding the pressurizer. All control rods are also assumed inserted.

The results of the analysis indicate that 3.4 hours of uncontrolled dilution would be required for the reactor to reach criticality.

DILUTION AT POWER

The plant conditions for this event include the plant at full power in automatic control, a boron dilution flow of 120 gpm at 579°F and reactivity addition rate of $1.2 \times 10^{-5}\delta k$/sec. The plants response to this event is similar to that for a rod withdrawal event with a low reactivity addition rate.

The results of this analysis indicate that with continued dilution the control rods will reach the minimum rod insertion limit in approximately six minutes. However, before the minimum insertion limit is reached, two alarms will be actuated to warn the operator of the accident condition. The first alarm alerts the operator to initiate normal boration, and the second alarm, at a lower setting from the first, alerts the operator to follow emergency boration procedures. In the event that the operator does not take immediate action it would take 15 minutes of continued dilution before the total shutdown margin is lost.

DILUTION AT SHUTDOWN CONDITIONS

This condition was not considered in the FSAR. However, in response to a generic concern on Westinghouse plants, the potential for boron dilution at shutdown while on RHR was evaluated. The concern that had been raised was that for certain dilution rates and shutdown margins, adequate time for operator action (15 minutes) might not be available unless certain restrictions were applied.

The licensee assessed the effect for the Ginna plant with a dilution rate of 120 gpm and Technical Specification shutdown margins.

For the plant-specific conditions at Ginna, the additional restrictions established by Westinghouse were not necessary to ensure 15 minutes for operator action before loss of shutdown margin. The means available to the operator for detection of the dilution event are the audible count rate indication, CVCS status indications and the high source range neutron flux at shutdown alarm. Upon determination that a dilution event is in progress, the operator is directed to secure the volume control tank suction and align the charging pump suction to the RWST. This will stop the addition of water via the makeup subsystem and provide borated water from the tank.

Conclusions

As part of the SEP review of Ginna the analysis has been evaluated against the criteria of SRP Section 15.4.6 and we have concluded that the analysis for uncontrolled boron dilution during refueling, startup, shutdown and at power is acceptable.

REFERENCES

1. R. E. Ginna Nuclear Power Plant, Unit 1, Final Facility Description and Safety Analysis Report, Rochester Gas and Electric Corporation, as supplemented.

TOPIC: XV-12, SPECTRUM OF CONTROL ROD EJECTION ACCIDENTS

Introduction

The ejection of a control rod from the core can be caused by a failure of the control rod housing such that reactor system pressure expels the control rod. Rod ejection results in a rapid increase in reactivity, energy production and a corresponding pressure increase.

Review Criteria

The guidelines presently used by the staff for review of this event are presented in Standard Review Plan Section 15.4.8 and Regulatory Guide 1.77.

Evaluation

The licensee submitted an analysis of this event in XN-NF-77-53, December 1977. This analysis, which was reviewed and approved by the staff for Cycle 8 operation of the R. E. Ginna reactor, used techniques developed by Exxon Nuclear Company which have been employed and approved for the reloads of several other reactors. The analysis showed a peak fuel enthalpy of 171 cal/gm whereas the Regulatory Guide allows 280 cal/gm. Since the methods used are current, and the results produced show considerable margin to the criteria for this accident contained in Regulatory Guide 1.77, we consider the analysis on record for Ginna to be acceptable.

Exxon methods for analysis of the rod ejection accident are described in XN-NF-78-44, "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," January, 1979. This report is under review in the Core Performance Branch. Should this review modify our conclusions concerning adequacy of Exxon methods for analysis of this accident, the staff would make appropriate revisions to these conclusions for Ginna.

Conclusion

Based on this review, we consider Topic XV-12 to be complete unless the ongoing review of rod ejection methodology identifies any inadequacies in the methods.

TOPIC XV-14: INADVERTENT OPERATION OF ECCS AND CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS) MALFUNCTION THAT INCREASES REACTOR COOLANT INVENTORY

Introduction

Inadvertent actuation of ECCS or a CVCS malfunction that increases coolant inv inventory can lead to an increase in system pressure and pressurizer level. Reactivity effects due to a decrease in boron concentration from a CVCS malfunction are considered under SEP Topic XV-10.

The overpressure protection features were reviewed for plant operations both at power and at low primary system temperature.

Acceptance criteria for the inadvertent operation of ECCS and CVCS are listed in Sub-Section II of the SRP Section 15.5.1 and 2.

Evaluation

During power operations for the Ginna facility, the HPSI pumps could not deliver flow at full operating pressure since the pump shutoff head is approximately 1500 psi. The three positive displacement charging pumps can deliver a maximum of 180 gpm (one of the three pumps is normally operating at 46 gpm). There are alarms to alert the operator for high pressurizer level, high pressurizer pressure and low volume control tank level. Reactor trip would occur on high pressurizer pressure or level. The steam volume in the pressurizer is 320 ft^3. It would take several minutes to fill this volume. Since only one charging pump is normally running while at power, the operator would have adequate time and indication to terminate the transient. In addition, automatic plant protection features, such as the pressurizer PORVs and safety valves, would also be available to help control the inventory and pressure increase.

The overpressure consequences during operation at low primary system temperature have been analyzed by the licensee. That analysis was reviewed and approved previously (Ref. 1) by the staff in conjunction with modifications to the plant design and technical specifications.

Conclusion

As part of the SEP review of Ginna we have reviewed the inadvertent operation of the ECCS and CVCS malfunction which result in increased reactor coolant inventory against the specific criteria of SRP Section 15.5.1 and 15.5.2 and concluded that the acceptance criteria are satisfied.

REFERENCES

1. Amendment 26 to the POL to DPR-18, dated 04/18/79.

TOPIC XV-15: INADVERTENT OPENING OF A PWR PRESSURIZER SAFETY/RELIEF VALVE

Introduction

The inadvertent opening of a pressurizer safety or relief valve or the failure to close following a overpressurization transient, results in a reactor coolant inventory decrease and a decrease in reactor coolant system pressure. If the valve is not closed, the continuing pressure decrease leads to a reactor trip and safety injection.

Evaluation

The reactor coolant system is protected from transient overpressure conditions as per the requirement of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. This protection is accomplished by several means including reactor trip, safety valves, and Power Operated Relief Valves (PORV). The Ginna plant is equipped with two PORV's. The PORV's are designed to prevent the lifting of the pressurizer code safety valves and to allow the reactor to remain on the line for load rejection transients. The PORV's (with a reduced setpoint) are also used to prevent overpressurization of the reactor vessel during operation at low temperature.

Failure of a PORV to reclose following an overpressure transient or inadvertent opening during operation initiates the design basis event.

The PORV used in the Ginna plant is a spring loaded valve with an air actuated opening. This overcomes the spring force on the valve stem and opens the valve. Closure of the valve is initiated by venting air off the control diaphragm causing the spring force to positively seat the valve closed. The valve will close on loss of air.

The ASME code safety valve for Ginna is set to open at 2485 psig and the PORV is set to open at reactor pressure of 2335 psig. Normal primary coolant system pressure is 2235 psig, so there is considerable margin for operation without approaching the safety valve setpoints.

Before the accident at Three Mile Island, inadvertent opening of a PORV or safety valve was considered only as a small break LOCA, and no specific analyses of PORV opening and its unique response characteristics were done. Generic analyses have been performed by Westinghouse (Ref. 1) in response to post TMI requirements for the inadvertent opening of PORV's. The computer code used for this analysis was WFLASH. This code was verified by comparison to the most applicable experimental data.

Two transients with breaks in the pressurizer vapor space were analyzed, a 0.008 ft^2 and a 0.034 ft^2 break. The 0.008 ft^2 break closely represents the flow area of one PORV of a typical Westinghouse plant. The other break is approximately the flow area of 3 PORV's and would give the largest insurge of flow into the pressurizer. Most Westinghouse plants including Ginna have 2 PORV's, so the maximum vapor space break would be smaller than the 0.034 ft^2 break. However, a few plants have 3 PORV's so this break size was considered. The larger break causes the system pressure to decrease more rapidly than the smaller break. The system also stabilizes much more quickly and the key events happen earlier in time. The analyses showed that in no case did the core uncover.

Conclusion

As part of the SEP review for Ginna, it is difficult to evaluate against the criteria of SRP Section 15.6.1 since this accident condition has not been addressed individually either in the FSAR or in a separate report. In general, the generic analysis (Ref. 1) conforms with the requirements of the SRP except that the analysis method used by Westinghouse for small break LOCA analysis for compliance with Appendix K should be revised, documented, and submitted for NRC approval. This recommendation was made by NRC staff in NUREG-0611 (Ref. 2).

Both the revised small break evaluation model and plant specific calculations are included in the TMI Action Plan, which is being performed outside the SEP. Therefore, we conclude that, for SEP, this topic is complete.

REFERENCES

1. "Report on Small Break Accidents For Westinghouse NSSS System," WCAP-9600, June, 1979.

2. "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants," NUREG-0611, January, 1980.

TOPIC XV-17: STEAM GENERATOR TUBE FAILURE

Introduction

In the event of a steam generator tube rupture, primary coolant will be discharged to the secondary until the primary pressure reduces below the secondary pressure (1100 psia) in the steam generator. This discharge of the primary coolant will result in radioactive release to the environment from direct leakage of the secondary system to the environment or from the overpressure atmospheric reliefs.

Review Criteria

The review criteria for steam generator tube ruptures are presented in SRP Section 15.6.3. Since the major safety concern for this event is the radiological consequences, the systems review is focused on the isolation of the affected generator, on operator actions, and on the potential for fuel damage.

Evaluation

The licensee has performed an analysis to assess the consequences of a steam generator tube rupture (Ref. 1). The initial plant response predicted in the analysis is a rapidly falling pressure and level in the pressurizer which initiates safety injection and reactor trip. The main effect of this early initiation of safety injection and reactor trip is a reduction in primary system pressure, which is required for isolation of the affected steam generator by closure of its steam line isolation valve.

The licensee expects that if the steam line isolation valve is closed after approximately one hour the leakage to the secondary would be about 25% of that assumed in the analysis. The analysis assumes that all of the reactor coolant activity is transferred to the steam system and that the affected steam generator is not isolated for 4-1/2 hours during the cooldown period. The licensee has modified his plant design to initiate safety injection from low pressurizer pressure alone, as opposed to the design identified in the FSAR (Ref. 2) which requires low pressurizer pressure coincident with low pressurizer level. The FSAR analysis is still applicable since this change would only result in a potentially sooner initiation of safety injection. The radiological consequences of a tube rupture are assessed in a separate evaluation.

Conclusion

As part of the SEP review for Ginna we have evaluated the reactor systems aspects of the licensee's analysis against the criteria of SRP Section 15.6.3.

Based on this evaluation we have concluded that the results of the reactor systems aspects of analysis are acceptable.

REFERENCES

1. R. E. Ginna Nuclear Power Plant, Unit 1, Final Facility Description and Safety Analysis Report, Rochester Gas and Electric Corporation, as supplemented.

TOPIC: XV-1 Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow and Inadvertent Opening of a Steam Generator Relief or Safety Valve.

15.1.1 & 15.1.2 Excessive Heat Removal Due to Feedwater System Malfunctions

Excessive heat removal, i.e., a heat removal rate in excess of the heat generation rate in the core, from the reactor coolant to the steam generator feedwater is caused by one of the following events:

(i) Feedwater system malfunction that results in a decrease in feedwater temperature.

(ii) Feedwater system malfunction that results in an increase in feedwater flow.

This group of accidents is analyzed to assure that the consequences of these moderate frequency events are acceptable as per criteria set by Standard Review Plan (SRP) Section 15.1. These two events discussed above are reviewed separately.

15.1.1 Decrease In Feedwater Temperature

Introduction

Excess heat removal can cause descrease in moderator temperature which increases core reactivity, leading to increase in power and decrease in shutdown margin. The increase in power can result in overpressurization of the primary and possibly lead to clad failures.

An example of excess heat removal by a decrease in feedwater temperature is the transient associated with the accidental opening of the condensate bypass valve which diverts feedwater flow around the low pressure feedwater heaters. In the

event of accidental opening of the bypass valve there would be a sudden reduction in feedwater inlet temperature causing an increase in subcooling of the primary. The increased subcooling will create a greater load demand on the primary and may lead to a reactor trip. The feedwater control system responds to keep steam generator level constant. However, if the control system does not function properly, the continued addition of cold feedwater may depressurize the primary system to the safety injection actuation setpoint. The safety injection signal will isolate the feedwater lines by venting the supply air to all feedwater control valves causing the valves to close, trip the main feedwater pumps, and close the feedwater discharge valves. The auxiliary feedwater system will take over heat removal needs until the reactor is cooled sufficiently to switch to the residual heat removal system.

Evaluation

Two cases were analyzed in Section 14.1.10 of the Ginna FSAR (Reference 1) to demonstrate the plant behavior in the event of a sudden feedwater temperature reduction resulting from accidental opening of the condensate bypass valve. The results were obtained by means of a detailed digital simulation of the plant including core kinetics, reactor coolant system, and the steam system. Both cases were assumed to occur from full power. The two cases analyzed are with automatic control by the rod control system and with no automatic control.

A zero moderator coefficient of reactivity was assumed for the uncontrolled case as this represents the condition where the plant has the least uncontrolled capability. For this case, the analysis showed a fairly rapid decrease in the reactor coolant average temperature and pressurizer pressure as the secondary heat extraction exceeds the core power generation. These parameters are summarized below in Table 1. The fixed low pressure trip would occur at 1880 psia at about 160 seconds from the initiation of valve opening. There is a considerable margin to DNB because of the accompanying large reduction in average coolant temperature. The DNBR at the time of trip is approximately 1.8.

The automatic controlled case was analyzed with a large negative moderator coefficient, which also acts to increase power. The core power increases and so it reduces the rate of decrease in coolant average temperature and pressurizer pressure. These parameters are tabulated below. The steady state conditions are reached with a minimum DNBR greater than 1.5. The plant would actually be tripped from the overpower protection set at 109% power. The trip setpoint used in the analysis is 118% power. The results of this analysis showed the maximum power attained in this transient was 115% power.

TABLE 1

Transient Response to Opening of

Condensate Bypass Valve

Parameters	No Control $\alpha_{mod} = 0$	With automatic control $\alpha_{mod} = -3.5x10-4$
1. Change in Pressurizer Pressure at 150 seconds	-450 psi	-35 psi
2. Change in Coolant Average Temperature at 150 seconds	-38°F	-4.6°F

Conclusions

As part of the SEP review of Ginna, the analysis of excessive heat removal events was reviewed against the specific criteria of Standard Review Plan (SRP) Section 15.1. Deviations from specific criteria are noted below:

1. The above analysis was made without reference to the specific computer code used for such analysis.
2. The transients were assumed to occur from full power for this analysis. According to SRP Section 15.1, the analysis should be done for the reactor initially at 102% of the rated core thermal power to account for a 2% power measurement uncertainty.
3. The analysis does not justify that the transient associated with the accidental opening of the condensate bypass valve is limiting.

However, the above deviations do not pose a safety problem since this event is bounded by the analyses performed for cycle 8 reload (Ref. 2).

15.1.2 Increase in Feedwater Flow

Introduction

The addition of excess feedwater is another means of increasing core power above full power. The overpower-overtemperature protection prevents any power increase which could lead to a DNBR less than 1.30.

Evaluation

The consequences of a step increase in feedwater flow to one steam generator from zero to full power flow at no load were analyzed in the Ginna FSAR (ref. 1). The calculations were based on conservative assumptions of constant feedwater temperature of 70°F, the most negative reactivity moderator coefficient assumed at the end of life. These calculations assume no credit for heat capacity of the reactor coolant system and steam generator shell thick metal. The maximum reactivity insertion rate was calculated to be 4.1×10^{-4} which is less than the maximum reactivity insertion rate analyzed for rod withdrawal from startup condition. If the accident occurs with the plant just critical at no load, the reactor will be tripped by the power range neutron flux level trip low setting set at approximately 25%. There is a large margin to DNB for the above calculated reactivity insertion rate.

The addition of cold feedwater after a reactor trip is interrupted by the actuation of safety injection on low pressurizer pressure and level. The safety injection signal will trip the main feedwater pumps and close the feedwater pump discharge valves as well as close the main feedwater control valves.

The licensee has evaluated the accidental opening of both feedwater control valves at full power. The consequences of this event were found to be less severe

than those resulting from the opening of the condensate bypass valve at full power as described in Section 15.1.1.

Conclusions

As part of SEP review of Ginna this topic has been evaluated against the criteria of SRP Section 15.1 and found to generally conform with the requirements of the SRP. The consequences of this event are bounded by those of Section 15.1.1.

15.1.3 Increase in Steam Flow

Introduction

A radid increase in steam generator steam flow causes a power mismatch between the reactor core power and the steam generator load demand. The reactor control system is designed to accomodate a 10% step load increase and a 5% per minute ramp load increase without a reactor trip in the range of 15 to 100% full power. Any loading rate in excess of these values may cause the reactor to be tripped by the reactor protection system. If the load increase exceeds the capability of the reactor coolant system, the transient is terminated in sufficient time to prevent the DNBR from decreasing below 1.3. An excessive load increase event could result from either an adminstrative violation such as excessive loading by the operator or an equipment malfunction in the steam bypass control or turbine speed control. In case of excessive loading by the operator or by system demand, the turbine load limiter limits maximum turbine load to 100% rated load.

During power operation, steam bypass to the condenser is controlled by reactor coolant conditions, i.e., abnormally high reactor coolant temperature indicates

a need for steam bypass. A single controller malfunction does not cause steam bypass because an interlock is provided which blocks the control signal to the valves unless a large turbine load decrease has occurred.

Increases in steam load to more than rated load are analyzed as steamline ruptures in FSAR. However, the reactor protection system will trip the reactor in time to prevent DNBR less than 1.30, regardless of the magnitude or rate of load increase.

Evaluation

Two cases have been analyzed in the FSAR to demonstrate the plant behavior in the event of excessive load increase. Both transients were assumed to occur from full power. These transients are (a) without automatic control and (b) with automatic reactor control. A zero moderator coefficient of reactivity was assumed for both cases which represents the condition where the plant has the least uncontrolled transient capability. The FSAR has presented results for 10% step increase in turbine load with and without automatic control. The results are similar but not as limiting as those of condensate bypass valve opening discussed in Section 15.1.1.

Without automatic control, the reactor coolant average temperature and pressurizer pressure decrease rapidly, as the secondary heat extraction exceeds the core power generation. The fixed low pressure trip occurs at about 150 seconds. At that time, reactor coolant temperature decreases by 45°F and pressurizer pressure decreases by 520 psi. There is a considerable margin to DNB because of the accompanying large reduction in coolant average temperature. The DNBR at the time of trip is approximately 1.8. The core power level remains essentially constant at full power.

The second case with automatic reactor control functioning is presented in the FSAR. The core power increases to about 112% of full power in 55 seconds before it levels off at 110% power at about 90 seconds. The increase in core power reduces the rate of decrease in coolant average temperature and pressurizer pressure. The average coolant temperature decreases by 3.3°F in about 30 seconds and then increases steadily showing an increase of 1°F at 100 seconds. The pressurizer pressure drops by 38 psi at 30 seconds and then increases sharply as much as 50 psi above normal pressure at 65 seconds. With no trip actuation steady state condition are reached with a minimum DNBR greater than 1.66.

Conclusions

The excessive load increase considered in this section will cause no radioactive release. Both transients show the same general behavior. A core power increase is accompanied by an average coolant temperature decrease and without a power increase there is a larger reduction in coolant average temperature. This has the effect of maintaining considerable margin to a limiting DNBR of 1.30.

As part of SEP review of Ginna this topic has been evaluated against the criteria of SRP Section 15.1.3, and found to conform with the requirements of the SRP.

15.1.4 Inadvertent Opening of Steam Generator Relief/Safety Valve

Introduction

A rupture of a steam pipe is assumed to include any accident which results in an uncontrolled steam release from a steam generator. The licensee has analyzed the effects of relief-safety valve opening as part of the spectrum of steam line

break. This particular event of inadvertent opening of the valve was considered as small line break in the FSAR (ref. 1). Since this transient is one of six limiting transients considered by the licensee, it was also reanalyzed for cycle 8 reload (ref. 2).

Evaluation

The transient analysis for the cycle 8 reload of the Ginna plant with Exxon Nuclear Fuel was performed using the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS PWR 2). The PTS PWR 2 code is a digital computer program developed to model the behavior of pressurized water reactors under normal and abnormal operating conditions. The analysis was done for a steam release rate of 273 lbs/sec assuming one loop operation at hot shutdown condition. The flow rate is based on the release from a steam generator safety valve since the safety valve was determined to have the largest venting capacity of any of the steam system valves in question. The minimum capability for boron injection was assumed corresponding to the most restrictive single failure in the safety injection system.

Additionally, the licensee assumed offsite power remains available and the most reactive control rod would remain withdrawn from the core. The small steamline break analysis showed that the worst expected shutdown margin at the end of cycle 8 is adequate to prevent return to criticality during such an event. The results presented in Reference 2 show the pressurizer pressure drops by 1300 psi in 180 seconds following steam line break. At that time, the reactor coolant temperature drops by approximately 80°F. Following the break, the maximum reactivity occurs at 175 seconds but does not reach criticality.

The Ginna plant would be protected from unacceptable consequences of this accident by several design features. The safety injection system would supply borated water and insert negative reactivity to the system. A low pressurizer pressure signal initiates the safety injection system. Borated water starts entering the injection lines after the pressurizer pressure has come down to the shutoff head (1400 psia)of the injection pumps. Borated water from the safety injection system reaches the core in 175 seconds. Provisions for isolation of the main steam system and feedwater system would reduce the severity of the accident by limiting the cooldown.

Results of the licensee analysis showed that the core does not go critical after the inadvertent opening of a steam generator safety valve.

The staff's review of this event is included in Amendment 19 to POL License No. DPR-18 (Reference 3). The review states that the consequences of a steam line break are acceptable for large break even though the plant would return to criticality. The small steam line break does not return to criticality. The staff has reviewed the PTS PWR 2 code and found its use acceptable (Ref. 3) for determining margins to the peak linear heat generation rate and departure from nucleate boiling design limits.

Conclusions

As part of the SEP review of Ginna, the analysis has been reviewed against the criteria of SRP Section 15.1.4. Based on our review of initial conditions and assumptions in the analysis, we conclude that the licensee analysis is in conformance to licensing criteria.

REFERENCES

1. R.E. Ginna Nuclear Power Plant, Unit 1, Final Facility Description and Safety Analysis Report, Rochester Gas and Electric Corporation.

2. XN-NF-77-40, "Plant Transient Analysis for the R.E. Ginna, Unit 1, Nuclear Power Plant," November 1977.

3. Amendment No. 19 to Provisional Operating License No. DPR-18, Letter from D.L. Ziemann (NRC) to L.D. White, Jr., (R G & E) dated May 1, 1978.

SEP TOPIC XV-2

TOPIC: XV-2 Spectrum of Steam System Piping Failures (PWR)

Steamline Break Inside Containment

Introduction

A steamline break in the secondary system results in an initial increase in steam flow which increases heat removal from the primary coolant system. This increased heat removal from the reactor coolant system causes a reduction in the coolant temperature and pressure. Reactor power increases because of the negative moderator temperature coefficient of reactivity feedback from the cooldown. This cooldown results in an insertion of positive reactivity, which could cause a return to criticality even after the reactor is scrammed. If the most reactive rod is assumed stuck in the withdrawn position there is an increased possibility that the reactor will become critical and return to power without the addition of negative reactivity to the core.

Evaluation

Assuming the most pessimistic combination of circumstances which could lead to power generation following a steam line break, the core is ultimately shut down by the boric acid in the Safety Injection System. The following systems in the Ginna plant provide the necessary protection to mitigate the consequences of a steamline rupture:

(1) Safety Injection System actuation on a) two out of three pressurizer low pressure signals, b) two out of three low pressure signals in any steam line, and c) two out of three high containment pressure signals.

(2) Reactor trip upon receiving high neutron flux signal, overtemperature ΔT signal, or upon actuation of the safety injection system.

(3) Redundant isolation of the main feedwater lines. In addition to the normal control action which will close the main feedwater valves, any safety injection signal will rapidly close all feedwater control valves, trip the main feedwater pumps, and close the feedwater pump discharge valves.

(4) Trip of the fast acting steam line isolation valves designed to close in less than 5 seconds with no flow on (a) one out of two high steam flow signals in that steam line in coincidence with any safety injection signal, and (b) two out of three high containment pressure signals.

Each steam line has a fast closing isolation valve and a check valve. These four valves (two for each steam generator) prevent blowdown of more than one steam generator for any break location even if one valve fails to close.

A spectrum of pipe breaks with various combinations of break sizes and initial plant conditions was analyzed in Reference 1. The following seven combinations were considered:

(A) Complete severance of a pipe outside the containment, downstream of the steam flow measuring nozzle (1.4 ft^2) at initial no load conditions with offsite power available and two loops in operation.

(B) Complete severance of a pipe inside the containment (4.37 ft^2) at the outlet of the steam generator at initial no load conditions with outside power available and two loops in operation.

(C) Case (A) above with only one loop in operation.

(D) Case (B) above with only one loop in operation.

(E) Case (A) above with loss of offsite power simultaneous with the steam break.

(F) Case (B) above with loss of offsite power simultaneous with the steam break.

(G) A break equivalent to steam release through one steam generator safety valve with offsite power available.

All above cases assumed initial hot shutdown conditions with the rods inserted (except for one stuck rod) at time zero. The steamline break at hot zero power condition is the worst case since the steam generator secondary side water inventory is maximum at this time, prolonging the duration and increasing the magnitude of the primary loop cooldown.

The analysis did not specifically account for auxiliary feedwater. However, the steam generator heat transfer code, using constant heat transfer coefficients, continued to calculate heat transfer from the primary to the secondary side after the broken steam generator had been estimated to be empty. If auxiliary flow was specifically accounted for, its effect would be negligible during the initial portion of the transient and would have minimal effect during later portions of the transient since by the time the broken steam generator empties, the total system reactivity is negative and core power is decreasing.

The limiting case based on the results of previous analysis (Reference 1) was identified and analyzed for cycle 8 reload (Reference 2). The latest analysis (cycle 8) considered the double ended guillotine rupture of a steam line inside containment. The analysis of the limiting MSLB was performed using the PTS PWR 2 (Reference 3) code for a break at hot standby conditions with offsite power available. Major assumptions include taking credit for minimum boron injection capacity and assuming the most reactive control rod stuck out of the core on reactor scram.

The results of the analysis indicate the plant reaches a peak average core power of 22% of rated power approximately 90 seconds after accident initiation. The minimum DNBR, determined using the modified Macbeth critical heat flux correlation, was 1.58.

The main steamline break analysis was reviewed and approved by the staff (Reference 4) without generic approval of the analytical methods. Approval of the plant specific analysis recognized the margin to DNB and the conservatism of the scram characteristics assumed in the analysis.

The effects of a postulated MSLB on other systems consistent with the intent of APCSB 3-1 and MEB 3-1 is addressed in SEP Topics III-5A and III-5B. Analysis of the containment response to a postulated MSLB, which may require an analysis considering different principal assumptions and/or single failures is considered under SEP Topics VI-2D and VI-3.

Conclusions

As part of the SEP review of Ginna, the MSLB analysis was reviewed against the criteria of SRP Section 15.1.5. The initial conditions, core kinetics,

power level, and operating conditions have been reviewed and found to generally conform with the requirements of the SRP.

Steamline Break Outside Containment

Introduction, Evaluation, and Conclusions

The rupture of a steam line break outside containment has been determined by the licensee (Reference 1) to have less severe consequences on the primary system than a MSLB inside containment. This is due to the fact that the steamlines have nozzles located inside containment. The normal function of the nozzles is to measure steam flow, but under accident conditions, the nozzles act as flow restrictors for breaks outside containment. By limiting the flow rate from the broken loop steam generators for outside containment pipe breaks, the nozzles reduce the cooldown of the primary system.

REFERENCES

1. Rochester Gas & Electric Corporation Proposed Change to Shutdown Margin Requirements in the Ginna Nuclear Plant Technical Specifications, Amendment Application dated September 22, 1975.

2. XN-NF-77-40, "Plant Transient Analysis for the R. E. Ginna, Unit 1, Nuclear Power Plant," November, 1977.

3. Kahn, J. D., Description of the Exxon Nuclear Power Plant Transient Simulation Model for Pressurized Water Reactors (PTS PWR), XN-74-5, Revision 1, May 1975.

4. Amendment No. 19 to Provisional Operating License No. DPR-18, Letter from D. L. Zieman (NRC) to L. D. White, Jr., (R G & E) dated May 1, 1978.

TOPIC XV-3: LOSS OF EXTERNAL LOAD, TURBINE TRIP, LOSS OF CONDENSER VACUUM, CLOSURE OF MAIN STEAM ISOLATION VALVE (BWR), AND STEAM PRESSURE REGULATOR FAILURE (CLOSED)

LOSS OF EXTERNAL LOAD

Introduction

Loss of external electrical load, due to a spectrum of electrical system conditions, is effected by the opening of the main generator breaker. The plant design does not provide a direct reactor trip from the breaker opening circuit. In the event of total loss of the electrical load the reactor will be tripped by high pressurizer pressure or high pressurizer level signal in the reactor trip system.

Review Criteria

The review criteria for the transients, such as loss of load, that result in an unplanned decrease in heat removal by the secondary system are presented in the SRP Section numbered 15.2.1, 2, 3, 4, 5.

Evaluation

The licensee has performed analyses of bounding conditions for total loss of load at beginning-of-life and end-of-life of the core (Ref. 1). The initial conditions include reactor power, coolant temperature and pressure all at maximum values with the plant at full power (102%) which leads to maximum power difference and minimum margin to core protection limits at the initiation of the loss of load.

First, the reactor control system was assumed in the normal automatic mode with the control rods in the minimum incremental worth region and the most reactive rod held out of the core. The steam bypass to the condenser was assumed unavailable while credit is taken for the effects of the pressurizer spray and relief valves in reducing or limiting coolant pressure, thus delaying the high pressurizer pressure reactor trip. Credit is also taken for the effects of control rod insertion by the reactor control system.

The results of the analysis, with the reactor in automatic control, indicate that the integrity of the core is maintained by high pressurizer pressure reactor trip in 12 seconds with peak pressure of 2511 psia and a minimum DNB ratio of 1.83, reached in 13 seconds. However, the licensee's analysis and results are limited to plant response only during the transient. A single failure in the systems required for the long term mitigation and decay heat removal from the core is not identified.

The licensee was requested to provide a single failure analysis to determine the limiting failure concurrent with a turbine trip (Ref. 2), that may affect the ability of the plant to mitigate the consequences of the transient. The

turbine trip event bounds the loss of load event, and therefore, the analysis with the worst single failure should demonstrate the adequacy of the plant design to mitigate the consequences of the loss of load transient.

The licensee has reported (Ref. 3) that a single failure analysis is not available and that the objective of the turbine trip transient analysis is to show that the primary pressure relieving devices can limit the pressure to acceptable levels, and that no core damage occurs during the transient.

The SEP staff evaluated the ability of the Ginna plant to reach safe shutdown assuming a loss of offsite power and a single active failure under Topic VII-3. Once the transient effects of overpressurization are over, the results of the above evaluation can be applied and the single failure criterion of the SRP satisfied.

The licensee has also analyzed the loss of load transient with the plant operating at full power in manual control. Credit is not taken for control rod insertion, pressurizer spray, relief valves or steam bypass.

The results of this analysis, with the reactor in manual control, indicate the reactor also tripped by high pressurizer pressure with peak pressure of 2415 psia. The minimum DNB ratio is not reported, however, it is greater than 1.3 based on a comparison with the case of automatic control.

Conclusions

As part of the SEP review for Ginna we have evaluated the analysis of the loss of external load and the licensee's response to our request for additional information (Ref. 3) against the criteria of SRP Section 15.2.1. Based on this evaluation we conclude that the criteria are satisfied.

TURBINE TRIP

Evaluation

The licensee has performed an analysis of the consequences of an instantaneous turbine trip by closure of the turbine stop valves. However, the event is identified as a loss of load transient (Ref. 3).

The turbine trip event is different from the loss of external load in that the fast closure of the turbine stop valves causes an abrupt interruption of steam to the turbine, creating a more severe overpressure condition in the primary system.

The licensee has analyzed the turbine trip transient with the plant operating at full power (102%) and in manual control. Credit is not taken for pressurizer spray, relief valves or steam bypass to the condenser.

The results of the analysis indicate that the reactor is tripped by high pressurizer pressure reactor trip in about 12 seconds, the mimimum DNB ratio 1.83 is reached in about 13 seconds and the pressurizer reaches a pressure of 2511 psia following the reactor trip. These results are very conservative for this plant since the licensee has in place a simultaneous turbine trip-reactor trip (anticipatory trip) associated with turbine stop valve closure. The reactor trip, therefore, would be effected in less than one second following turbine trip, as opposed to the 12 seconds assumed in the analysis, and would result in less severe conditions to the core and vessel.

The discussion presented under Loss of External Load on single failure considerations is also applicable to this event.

Conclusions

As part of the SEP review for Ginna we have evaluated the analysis of turbine trip and the licensee's response to our request for additional information (Ref. 3). Based on this evaluation we conclude that the criteria of SRP Section 15.2.1 are met.

LOSS OF CONDENSER VACUUM

Evaluation

Loss of condenser vacuum can be effected from failure of the circulating water system or excessive air leakage through turbine gland packing. In the event of loss of condenser vacuum the turbine will be tripped and therefore this event is bounded by the turbine trip event coincident with loss of condenser vacuum.

Conclusions

A turbine trip event coincident with loss of condenser vacuum has been analyzed by the licensee and the results of the analysis have been evaluated against the criteria of SRP Section 15.2.1. Our conclusions are included in the evaluation for the turbine trip event.

REFERENCES

(1) XN-NF-77-40, "Plant Transient Analysis for the R. E. Ginna, Unit 1, Nuclear Power Plant," November, 1977.

(2) Letter from D. Ziemann (NRC) to L.D. White (RG&E) dated February 25, 1980.

TOPIC XV-4: LOSS OF NON-EMERGENCY A-C POWER TO THE STATION AUXILIARIES

Introduction

Loss of all a-c power to station auxiliaries, while the plant is at power, will cause loss of main feedwater and thus, loss of capability of the secondary system to remove heat generated in the primary. Additional power generation, however, will be abruptly interrupted as the reactor is tripped by a fast undervoltage or underfrequency reactor trip signal, generated from loss of a-c power. Forced coolant circulation in the primary will also be interrupted as a result of loss of power to the reactor coolant pumps and natural circulation would have to be relied upon to carry the coolant through the steam generators for removal of the core decay heat. Therefore, a source of feedwater (auxiliary) will be required to be made available to remove the decay heat. A source of primary coolant make-up supply may also become necessary to maintain the required inventory level in the primary, if inventory is lost from actuation of pressurizer relief valves.

Evaluation

The licensee has performed an analysis of the consequences of loss of all a-c power to the station auxiliaries (Ref. 1). The analysis assumes the loss of normal feedwater on the secondary side and loss of forced circulation of the primary coolant. Both normal feedwater and forced circulation are supplied motive power by the non-emergency a-c power system. Hence, loss of that power supply results in the loss of both systems.

The principal discussion on the method of analysis for this transient appears in the analysis for loss of normal feedwater (Ref. 1). In that analysis the loss of feedwater is assumed coincident with natural circulation in the primary coolant system.

The initial conditions for the analysis include the reactor at full power and the steam generators at the lowest water level during reactor trip. The steam bypass to the condenser was assumed unavailable and credit is taken for only one motor driven auxiliary feedwater pump actuated automatically one minute after the event.

The results of the analysis indicate that following reactor trip and turbine trip, the steam generator water level will not reduce below the level at which sufficient heat transfer area is available to dissipate core decay heat without water relief from the primary relief, or safety valves.

Conclusions

As part of the SEP review for Ginna the analysis has been evaluated against the criteria of SRP Section 15.2.6 and we have concluded that it is in conformance with these criteria.

REFERENCES

(1) R. E. Ginna Nuclear Power Plant, Unit 1, Final Facility Description and Safety Analysis Report, Rochester Gas and Electric Corporation, as supplemented.

SEP TOPIC XV-5

TOPIC: XV-5 Loss of Normal Feedwater Flow

Introduction

A loss of normal feedwater could be caused by pump failures, valve malfunctions, or a loss of offsite power. The result of this loss of feedwater flow would be increasing reactor coolant temperature and pressure due to a decrease in heat removal and decreasing water level in the steam generator. If the reactor were not tripped during the accident, fuel damage could possibly occur from a sudden loss of heat sink.

Evaluation

Reactor protection is provided by (a) trip on low steam generator water level, (b) trip on steam flow-feedwater flow mismatch in coincidence with low water level in either steam generator, (c) two motor driven auxiliary feedwater pumps, and (d) one turbine driven pump. The motor driven pumps are automatically initiated by 2 of 3 low-low steam generator level in either steam generator, trip of both main feedwater pumps, or safety injection initiation. The steam admission valve to the turbine driven pump is automatically opened on 2 out of 3 low-low water level in both steam generators or loss of voltage on both 4kv buses. Ginna also has a standby auxiliary feedwater system (SBAFWS) with two motor driven pumps which are powered from emergency buses. The SBAFWS is manually actuated. The motor driven auxiliary feedwater pumps are supplied by the diesel if a loss of offsite power occurs and the turbine driven pump utilizes steam from the secondary systems.

The analysis, as discussed in the FSAR, has been performed to show that following a loss of normal feedwater, the auxiliary feedwater system is adequate to remove stored and residual heat to prevent water relief through the pressurizer relief valves. Conservative initial conditions (102% initial power, low steam generator water level, only one motor driven auxiliary feedwater pump available at one minute after the accidents, and low heat transfer coefficient in the steam generator) were assumed.

The results of the analysis in the FSAR indicate that at no time is the tube sheet uncovered in the steam generator receiving auxiliary feedwater flow and at no time is there water relief from the pressurizer. The capacity of one motor driven auxiliary feedwater pump is such that the water level in the steam generator being fed does not recede below the lowest level at which sufficient heat transfer area is available to dissipate core residual heat without water relief from the primary system relief or safety valve.

Conclusions

As part of our SEP review for Ginna the analysis has been evaluated against the criteria of SRP section 15.2.7 and found to generally conform with the requirements of the SRP.

This event is bounded by the analysis performed for loss of Non-Emergency A-C power to the station auxiliaries (SEP Topic XV-4).

TOPIC XV-6: FEEDWATER SYSTEM PIPE BREAKS

Introduction

A feedwater line break can result in either a reactor system cooldown (such as that from a steamline break) or a reactor coolant system heatup (by reducing feedwater flow to the affected steam generator). Feedwater break cooldowns would be less severe than those analyzed in the steamline rupture analysis. Thus, the analysis has been performed to demonstrate that the system is capable of sustaining a feedwater line rupture under initial conditions and assumptions which result in the most severe heatup of the primary system.

Evaluation

The licensee performed an analysis of a heatup resulting from a feedwater line rupture in Reference 1. This event was not part of the original licensing basis for this facility, which was submitted in the FSAR (Ref. 2).

The primary system transient resulting from the hypothetical double ended rupture of a main feedwater line was analyzed. Conservative initial conditions (102% initial power, only one motor driven auxiliary feedwater pump available at ten minutes after reactor trip to allow sufficient time for operator realignment of the auxiliary feedwater system i.e., isolation of AFW to the affected steam generator, a conservative core residual heat generation, a zero quality blowdown, low heat transfer coefficient, reactor trip after all steam generator liquid emptied) were assumed.

Reactor protection in the event of a feedwater line break is provided by (a) reactor trip, (b) two motor driven auxiliary feedwater pumps, (c) one turbine driven auxiliary feedwater pump, and (d) a standby auxiliary feedwater system (SBAFWS) with two motor driven pumps which are powered from emergency buses. The SBAFWS is manually actuated. The motor driven auxiliary feedwater pumps are supplied by the diesel if a loss of offsite power occurs and the turbine driven pump utilizes steam from the secondary systems.

The results of the analysis in Reference 1 demonstrated that under the severely limiting assumptions used to maximize the time delay to reactor trip and minimize heat removal from the steam generator blowing down, the system is capable of removing decay heat following the blowdown without exceeding the safety limit for reactor coolant system pressure. Maximum reactor coolant pressure remains below 2600 psia which is well below the SRP Section 15.2.8 guidelines of 110% of design pressure since the design pressure of Ginna is 2500 psia. Assuming one of the motor driven pumps fails to operate and neglecting the pumping capacity of the turbine driven auxiliary feedwater pump, the energy removal capability of the secondary system exceeds the residual energy generation in the primary system within 30 minutes of the occurrence of the assumed rupture. At no time does the reactor coolant thermal expansion rate exceed the pressurizer safety valve capacity. Therefore, no overpressurization of the reactor coolant system will occur. Total discharge of liquid from the reactor coolant system through the pressurizer safety valve is 1394 ft^3 due to thermal expansion, or less than 25% of the initial reactor coolant system liquid volume. Therefore, the core remains flooded.

The licensee has indicated that the MARVEL code was used for this analysis, although the specific revision was not known. The MARVEL code has been used extensively by Westinghouse for plant transient analysis. The code is still under staff review, however, the staff has previously accepted results obtained with this code in licensing actions.

Conclusions

As part of our SEP review for Ginna the analysis has been evaluated against the criteria of SRP Section 15.2.8 and found to generally conform with the requirements of the SRP. Therefore, the staff concludes that the analysis of this event is acceptable.

REFERENCES

1. Letter from K. W. Amish (RG&E) to J. F. O'Leary (NRC) dated May 24, 1974.

2. R. E. Ginna Nuclear Power Plant, Unit 1, Final Facility Description and Safety Analysis Report, Rochester Gas and Electric Corporation, as supplemented.

TOPIC XV-7: REACTOR COOLANT PUMP ROTOR SEIZURE AND REACTOR COOLANT PUMP SHAFT BREAK

LOSS OF FORCED COOLANT FLOW

Introduction

The loss of forced coolant flow event considers the cases of both a partial and complete loss-of-coolant flow. A loss of forced coolant flow could result from mechanical or electrical failures in one or both of the reactor coolant pumps or from failures in the power supply to the pumps. A decrease in reactor coolant flow while the reactor is at power would result in a degradation of core heat transfer and an increase in the primary coolant temperature due to reduced heat transfer in the steam generators. This increase in primary coolant temperature could result in DNB and subsequent fuel damage if adequate safety features are not provided.

Evaluation

The licensee provided the latest analysis for a loss of forced coolant flow event in Reference 3. This analysis, performed using the PTSPWR2 code, (Ref. 1) considered only the case of two pumps coasting down which was determined to be the limiting case from the results of previous analysis (Ref. 2). The licensee assumed beginning-of-cycle values for the moderator and fuel coefficients with a factor of 1.2 applied to the Doppler coefficient. The most reactive rod was assumed stuck out of the core following reactor scram.

The Ginna plant is protected against the consequences of this event by the reactor protection system which scrams the reactor on a variety of signals which are specifically related to the loss of forced coolant flow. The reactor trips of special interest are the reactor trip on either undervoltage or underfrequency at the bus supplying power to the pumps and the reactor trip on low flow. The low flow trip is designed, for most Westinghouse plants, to provide suitable protection in the event of the loss of a single reactor coolant pump. The undervoltage/underfrequency scram signal provides additional protection in the event both pumps fail since the signal would be generated more quickly than the low flow derived signal. The licensee, however, has conservatively assumed in the analysis that the reactor scrams on the low flow signal.

The results of the analysis indicate a minimum DNBR of 1.61 is reached 4.7 seconds after the accident initiation. The staff approved the analysis performed by the licensee (Ref. 3) in Reference 4. As part of this evaluation the staff also approved the consideration of only the case where both pumps are assumed to coast down.

Conclusions

As part of the SEP review of Ginna we have reviewed the analysis of a loss of forced coolant flow against the specific criteria of SRP Section 15.3.1. We conclude that the acceptance criteria are satisfied.

REACTOR COOLANT PUMP ROTOR SEIZURE

Introduction

In the event of a reactor coolant pump rotor seizure the primary coolant flow through the affected loop will be abruptly interrupted and the loss of flow could cause severe overheating in the primary system. The other pump would continue to circulate coolant through the unaffected loop. However, if offsite power is lost the other pump will be tripped and its contribution will be limited to only coastdown flow, thus making the accident more severe. The coastdown flow is available from the energy stored in the inertia of the pump's flywheel. For the pump seizure event the reactor will be tripped by the loss of flow signal in the reactor protection system.

Evaluation

The licensee has performed an analysis of a locked pump rotor (Ref. 3). The initial conditions include the reactor at rated power with begining of cycle kinetics and 1.2 multiplier applied to the Doppler coefficient. The steam bypass to the condenser was assumed unavailable and the effects of the pressurizer spray and relief valves in reducing or limiting coolant pressure were neglected. Also, the feedwater pumps were assumed to be tripped. The unaffected main coolant (RCP) pump, however, is assumed to continue to circulate coolant through the unaffected loop during the transient, without loss of offsite power.

The results of the analysis indicated that the reactor is tripped by the loss of flow reactor trip in .8 seconds with an increase in system temperature and pressure of 13°F and 53 psia respectively. The minimum DNB ratio was calculated to be 1.23 with less than 1% of the fuel rods failed.

Standard Review Plan Section 15.3.3 requires that the analysis of a locked rotor event consider loss of offsite power and coastdown.

We believe that an analysis assuming loss of offsite power and coastdown of the unaffected pump, will result in DNB ratio lower than the licensee has calculated (1.23) and could give rise to more than 1% fuel rods failure.

The licensee has provided a comparison of the results for a locked rotor event with and without coastdown of the second pump. Assuming an instantaneous loss of offsite power, the flow is reduced approximately 4% at the time of minimum DNB ratio, and the DNBR is decreased by only 0.1. Although some additional clad failures may result, the core cooling capability should not be affected.

This evaluation was performed with the RETRAN code. The licensee also provided a comparison of the predictions for a loss of flow event for RETRAN and the Exxon code used to perform the reference analysis. The results showed good agreement, so the above sensitivity can be applied to the reference case. Both RETRAN and the Exxon code are under staff review, but results calculated by the codes have been previously accepted in licensing evaluations.

Conclusion

As part of the SEP review of Ginna we have evaluated the analysis of the locked pump rotor event against the criteria of SRP Section 15.3.3. Based on this evaluation we conclude that the locked rotor event has been adequately considered by the licensee.

REACTOR COOLANT PUMP SHAFT BREAK

Evaluation

In the event of a reactor coolant pump shaft break, the sudden decrease in coolant flow will result in degradation of core heat transfer and will cause some fuel damage. The decrease in coolant flow from this event is not as severe, early in the transient, as is the decrease in flow from the bounding event of the pump rotor seizure. Early during the pump shaft break transient the impeller of the affected pump will continue to rotate in the forward direction aiding the circulation of the coolant. However, the flow through the affected loop will reverse later during the transient and will result in a lower core flow rate.

The licensee has not performed an analysis of the pump shaft break transient. However, since the core flow, early during this transient, is greater than the flow in the pump rotor seizure event the consequences are less severe that the pump rotor seizure. The consequences later during the transient, when the impeller reverses direction may be more severe than the pump rotor seizure event. Since the loss of flow events are thermally limiting early in the transient, lower steady-state core flow should not adversely affect long-term core cooling.

Conclusion

As part of the SEP review of Ginna, we have evaluated the reactor coolant pump shaft break event and based on this evaluation we concluded that analyses are not required.

REFERENCES

1. Kahn, J. D., Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTSPWR), XM-74-5, Revision 1, May 1975.

2. R. E. Ginna Nuclear Power Plant, Unit 1, Final Facility Description and Safety Analysis Report, Rochester Gas and Electric Corporation, as supplemented.

3. XN-NF-77-40, "Plant Transient Analysis for the R. E. Ginna, Unit 1, Nuclear Power Plant," November, 1977.

4. Amendment No. 19 to Provisional Operating License No. DPR-18, Letter from D. L. Ziemann (NRC) to L. D. White, Jr., (RG&E) dated May 1, 1978.

R. E. GINNA

TOPIC XV-8, CONTROL ROD MISOPERATION

Introduction

Control rod misoperation can occur through operator error or malfunction of the rod control system. Situations considered are rod misalignment, rod withdrawal, and rod drop. These events can result in power distribution and reactivity changes.

Review Criteria

The guidance presently used by the staff for evaluating these events is presented in Standard Review Plan Sections 15.4.1, 15.4.2 and 15.4.3. The criterion to be satisfied is that fuel thermal limits not be exceeded, i.e. that the departure from nucleate boiling (DNB) ratio is greater than 1.3.

Evaluation

Rod Misalignment

Rod misalignment can lead to changes in power distributions and local peaking. During the operating license review for Ginna the staff performed a review of rod misalignment that led to power distribution control technical specifications, such as quadrant tilt limits, that have since been applied to all Westinghouse reactors. The staff concludes that the analysis performed and the provisions made at the time are still acceptable.

Rod Withdrawal

An uncontrolled rod group withdrawal at power is a reactivity addition which results in an increase in the core heat flux and power level. Since heat extraction from the steam generator lags behind the core power increase until the steam generator pressure increases to the relief or safety valve setpoint, there is an increase in the primary coolant temperature.

The licensee has performed analyses of an uncontrolled rod withdrawal (Reference 1) using the PTSPWR2 Code to demonstrate the adequacy of the reactor protection system. The analysis was done assuming both a fast rod withdrawal ($6. \times 10^{-4}$ ΔK/sec) and a slow rod withdrawal ($50. \times 10^{-6}$ ΔK/sec) at 100% power. The reactivity insertion rates chosen were based on the rates determined to be limiting from previous analysis (Reference 2).

The analyses performed by the licensee show that for the high reactivity insertion rate the reactor is tripped on reaching the high neutron flux trip setpoint. For this case the DNBR decreased from 2.00 initially to 1.77, and the primary system pressure increased from 2220 psia to 2235 psia with the core average temperature increasing by less than 1°F. As can be seen from the results the neutron flux level in the core rises rapidly for the high reactivity insertion rate while the core heat flux and primary coolant temperature lag behind due to the thermal capacitance of the fuel and primary system coolant.

For the case assuming a slow reactivity insertion rate the reactor protection system scrams the reactor upon first reaching the overtemperature ΔT trip setpoint. The results of analysis for slow rod withdrawal demonstrate a similar margin to DNB as for fast rod withdrawal with the reactor trip occurring after a longer period. In the event of a slow rod withdrawal the core heat flux remains more closely matched to the neutron flux. Thus by the time the trip setpoint is reached the primary system pressure and temperature will have experienced a greater increase although remaining below acceptable limits.

For lower initial power conditions, the margin to DNB is only slightly lower, DNBR of 1.69 for 60% initial power. This analysis was reviewed and accepted by the staff for Cycle 8 operation of the reactor. The reload involved a change of fuel supplier, who used methods only recently developed for analysis of PWR transients. These methods are used for a number of Westinghouse designed reactors, and are considered current. Because of these factors, the staff concludes that the uncontrolled rod withdrawal analysis is acceptable.

Control Rod Drop

A control rod drop results in a decrease in reactor power, pressure and temperature. Depending on the worth of the rod, thermal limits can be approached because of the distorted power distribution. If the rod control system is controlling rod motion, other control rods may be withdrawn in response to the decreasing power and temperature, which exacerbates the maldistributions caused by the dropped rod.

Rod drop protection at Ginna is provided by a turbine runback and rod block. These features act to prevent adverse thermal consequences upon detection of rod bottom lights, indicating a fully inserted rod. The rod block prevents withdrawal of other control rods, and the turbine runback reduces generator load, and thus the temperature and pressure drop.

The licensee has provided (Ref. 2) an analysis that shows that for a rod drop event, these protection features ensure margin to departure from nucleate boiling (DNB). The staff reviewed and approved that analysis during the Ginna licensing review. However, the staff also noted that later generation Westinghouse reactors have provided a safety grade negative flux rate trip for rod drop event protection and that for the older Westinghouse plant, such as Ginna, the existing protection is not safety grade. THis question as to the adequacy of the rod block turbine runback protection for rod drop events is being pursued generically.

Conclusion

Based on this review, the staff concludes that the analyses of these events is acceptable and that Topic XV-8 is complete. If the staff concludes that additional protection is necessary as a result of the generic review of rod drop protection for the older Westinghouse plants it will be handled on a case by case basis, separate from the SEP.

REFERENCES

1. "Plant Transient Analysis for the R. E. Ginna Unit 1 Nuclear Power Plant", XN-NF-77-40, November, 1977.

2. R. E. Ginna Nuclear Power Plant Unit 1, Final Facility Description and Safety Analysis Report, Rochester Gas and Electric Corportation, as supplemented.

TOPIC XV-10: CHEMICAL AND VOLUME CONTROL SYSTEMS MALFUNCTION THAT RESULTS IN A DECREASE IN BORON CONCENTRATION IN THE REACTOR COOLANT (PWR)

Introduction

The Chemical and Volume Control System (CVCS) supplies reactor makeup water to the reactor coolant system via the reactor makeup control system. A boric acid blend system is provided to permit the operator to match the boron concentration of reactor coolant makeup water to that in the reactor coolant system. Reactivity can, therefore, be added to the reactor coolant if the boron dilution process becomes uncontrolled.

Review Criteria

The review criteria for boron dilution events is presented in SRP Section 15.4.6. Since operator action is generally relied upon to terminate the event, minimium time intervals of 30 minutes during refueling and 15 minutes during other operational modes, must be available from the time an alarm alerts the operator to loss of shutdown margin.

Evaluation

The licensee has performed an analysis for an uncontrolled boron dilution event (Ref. 1). The analysis assumes plant modes of refueling, startup and power operation. The rate of addition of unborated water makeup is limited to the capacity of the makeup water pumps at the maximum combined flow of 120 gpm for both pumps.

DILUTION DURING REFUELING

The plant conditions for this event include one residual heat removal pump operating, boron concentration of 2000 ppm, all control rods inserted, a minimum water volume in the reactor coolant system of 2724 ft^3 (this corresponds to the volume necessary to fill the reactor vessel above the nozzles) and the maximum dilution flow of 120 gpm.

The results of the analysis indicate that the boron concentration must be reduced from 2000 ppm to approximately 1080 ppm before the reactor will go critical, and it would require 1.75 hours to reach criticality. The operator, therefore, will have sufficient time to recognize the high neutron count rate alarms in the containment and the control room and isolate the reactor makeup water source by closing valves and stopping the pumps.

DILUTION DURING STARTUP

The plant conditions for this event include the reactor coolant system filled with borated water at 2000 ppm, reactor coolant pumps are operating and the volume of the reactor coolant is approximately 5247 ft^3 which is the reactor coolant system excluding the pressurizer. All control rods are also assumed inserted.

TOPIC XV-10: CHEMICAL AND VOLUME CONTROL SYSTEMS MALFUNCTION THAT RESULTS IN A DECREASE IN BORON CONCENTRATION IN THE REACTOR COOLANT (PWR)

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DILUTION DURING STARTUP

The plant conditions for this event include the reactor coolant system filled with borated water at 2000 ppm, reactor coolant pumps are operating and the volume of the reactor coolant is approximately 5247 ft^3 which is the reactor coolant system excluding the pressurizer. All control rods are also assumed inserted.

The results of the analysis indicate that 3.4 hours of uncontrolled dilution would be required for the reactor to reach criticality.

DILUTION AT POWER

The plant conditions for this event include the plant at full power in automatic control, a boron dilution flow of 120 gpm at 579°F and reactivity addition rate of $1.2 \times 10^{-5}\delta k$/sec. The plants response to this event is similar to that for a rod withdrawal event with a low reactivity addition rate.

The results of this analysis indicate that with continued dilution the control rods will reach the minimum rod insertion limit in approximately six minutes. However, before the minimum insertion limit is reached, two alarms will be actuated to warn the operator of the accident condition. The first alarm alerts the operator to initiate normal boration, and the second alarm, at a lower setting from the first, alerts the operator to follow emergency boration procedures. In the event that the operator does not take immediate action it would take 15 minutes of continued dilution before the total shutdown margin is lost.

DILUTION AT SHUTDOWN CONDITIONS

This condition was not considered in the FSAR. However, in response to a generic concern on Westinghouse plants, the potential for boron dilution at shutdown while on RHR was evaluated. The concern that had been raised was that for certain dilution rates and shutdown margins, adequate time for operator action (15 minutes) might not be available unless certain restrictions were applied.

The licensee assessed the effect for the Ginna plant with a dilution rate of 120 gpm and Technical Specification shutdown margins.

For the plant-specific conditions at Ginna, the additional restrictions established by Westinghouse were not necessary to ensure 15 minutes for operator action before loss of shutdown margin. The means available to the operator for detection of the dilution event are the audible count rate indication, CVCS status indications and the high source range neutron flux at shutdown alarm. Upon determination that a dilution event is in progress, the operator is directed to secure the volume control tank suction and align the charging pump suction to the RWST. This will stop the addition of water via the makeup subsystem and provide borated water from the tank.

Conclusions

As part of the SEP review of Ginna the analysis has been evaluated against the criteria of SRP Section 15.4.6 and we have concluded that the analysis for uncontrolled boron dilution during refueling, startup, shutdown and at power is acceptable.

REFERENCES

1. R. E. Ginna Nuclear Power Plant, Unit 1, Final Facility Description and Safety Analysis Report, Rochester Gas and Electric Corporation, as supplemented.

TOPIC: XV-12, SPECTRUM OF CONTROL ROD EJECTION ACCIDENTS

Introduction

The ejection of a control rod from the core can be caused by a failure of the control rod housing such that reactor system pressure expels the control rod. Rod ejection results in a rapid increase in reactivity, energy production and a corresponding pressure increase.

Review Criteria

The guidelines presently used by the staff for review of this event are presented in Standard Review Plan Section 15.4.8 and Regulatory Guide 1.77.

Evaluation

The licensee submitted an analysis of this event in XN-NF-77-53, December 1977. This analysis, which was reviewed and approved by the staff for Cycle 8 operation of the R. E. Ginna reactor, used techniques developed by Exxon Nuclear Company which have been employed and approved for the reloads of several other reactors. The analysis showed a peak fuel enthalpy of 171 cal/gm whereas the Regulatory Guide allows 280 cal/gm. Since the methods used are current, and the results produced show considerable margin to the criteria for this accident contained in Regulatory Guide 1.77, we consider the analysis on record for Ginna to be acceptable.

Exxon methods for analysis of the rod ejection accident are described in XN-NF-78-44, "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," January, 1979. This report is under review in the Core Performance Branch. Should this review modify our conclusions concerning adequacy of Exxon methods for analysis of this accident, the staff would make appropriate revisions to these conclusions for Ginna.

Conclusion

Based on this review, we consider Topic XV-12 to be complete unless the ongoing review of rod ejection methodology identifies any inadequacies in the methods.

TOPIC XV-14: INADVERTENT OPERATION OF ECCS AND CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS) MALFUNCTION THAT INCREASES REACTOR COOLANT INVENTORY

Introduction

Inadvertent actuation of ECCS or a CVCS malfunction that increases coolant inventory can lead to an increase in system pressure and pressurizer level. Reactivity effects due to a decrease in boron concentration from a CVCS malfunction are considered under SEP Topic XV-10.

The overpressure protection features were reviewed for plant operations both at power and at low primary system temperature.

Acceptance criteria for the inadvertent operation of ECCS and CVCS are listed in Sub-Section II of the SRP Section 15.5.1 and 2.

Evaluation

During power operations for the Ginna facility, the HPSI pumps could not deliver flow at full operating pressure since the pump shutoff head is approximately 1500 psi. The three positive displacement charging pumps can deliver a maximum of 180 gpm (one of the three pumps is normally operating at 46 gpm). There are alarms to alert the operator for high pressurizer level, high pressurizer pressure and low volume control tank level. Reactor trip would occur on high pressurizer pressure or level. The steam volume in the pressurizer is 320 ft^3. It would take several minutes to fill this volume. Since only one charging pump is normally running while at power, the operator would have adequate time and indication to terminate the transient. In addition, automatic plant protection features, such as the pressurizer PORVs and safety valves, would also be available to help control the inventory and pressure increase.

The overpressure consequences during operation at low primary system temperature have been analyzed by the licensee. That analysis was reviewed and approved previously (Ref. 1) by the staff in conjunction with modifications to the plant design and technical specifications.

Conclusion

As part of the SEP review of Ginna we have reviewed the inadvertent operation of the ECCS and CVCS malfunction which result in increased reactor coolant inventory against the specific criteria of SRP Section 15.5.1 and 15.5.2 and concluded that the acceptance criteria are satisfied.

REFERENCES

1. Amendment 26 to the POL to DPR-18, dated 04/18/79.

TOPIC XV-15: INADVERTENT OPENING OF A PWR PRESSURIZER SAFETY/RELIEF VALVE

Introduction

The inadvertent opening of a pressurizer safety or relief valve or the failure to close following a overpressurization transient, results in a reactor coolant inventory decrease and a decrease in reactor coolant system pressure. If the valve is not closed, the continuing pressure decrease leads to a reactor trip and safety injection.

Evaluation

The reactor coolant system is protected from transient overpressure conditions as per the requirement of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. This protection is accomplished by several means including reactor trip, safety valves, and Power Operated Relief Valves (PORV). The Ginna plant is equipped with two PORV's. The PORV's are designed to prevent the lifting of the pressurizer code safety valves and to allow the reactor to remain on the line for load rejection transients. The PORV's (with a reduced setpoint) are also used to prevent overpressurization of the reactor vessel during operation at low temperature.

Failure of a PORV to reclose following an overpressure transient or inadvertent opening during operation initiates the design basis event.

The PORV used in the Ginna plant is a spring loaded valve with an air actuated opening. This overcomes the spring force on the valve stem and opens the valve. Closure of the valve is initiated by venting air off the control diaphragm causing the spring force to positively seat the valve closed. The valve will close on loss of air.

The ASME code safety valve for Ginna is set to open at 2485 psig and the PORV is set to open at reactor pressure of 2335 psig. Normal primary coolant system pressure is 2235 psig, so there is considerable margin for operation without approaching the safety valve setpoints.

Before the accident at Three Mile Island, inadvertent opening of a PORV or safety valve was considered only as a small break LOCA, and no specific analyses of PORV opening and its unique response characteristics were done. Generic analyses have been performed by Westinghouse (Ref. 1) in response to post TMI requirements for the inadvertent opening of PORV's. The computer code used for this analysis was WFLASH. This code was verified by comparison to the most applicable experimental data.

Two transients with breaks in the pressurizer vapor space were analyzed, a 0.008 ft^2 and a 0.034 ft^2 break. The 0.008 ft^2 break closely represents the flow area of one PORV of a typical Westinghouse plant. The other break is approximately the flow area of 3 PORV's and would give the largest insurge of flow into the pressurizer. Most Westinghouse plants including Ginna have 2 PORV's, so the maximum vapor space break would be smaller than the 0.034 ft^2 break. However, a few plants have 3 PORV's so this break size was considered. The larger break causes the system pressure to decrease more rapidly than the smaller break. The system also stabilizes much more quickly and the key events happen earlier in time. The analyses showed that in no case did the core uncover.

Conclusion

As part of the SEP review for Ginna, it is difficult to evaluate against the criteria of SRP Section 15.6.1 since this accident condition has not been addressed individually either in the FSAR or in a separate report. In general, the generic analysis (Ref. 1) conforms with the requirements of the SRP except that the analysis method used by Westinghouse for small break LOCA analysis for compliance with Appendix K should be revised, documented, and submitted for NRC approval. This recommendation was made by NRC staff in NUREG-0611 (Ref. 2).

Both the revised small break evaluation model and plant specific calculations are included in the TMI Action Plan, which is being performed outside the SEP. Therefore, we conclude that, for SEP, this topic is complete.

REFERENCES

1. "Report on Small Break Accidents For Westinghouse NSSS System," WCAP-9600, June, 1979.

2. "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants," NUREG-0611, January, 1980.

TOPIC XV-17: STEAM GENERATOR TUBE FAILURE

Introduction

In the event of a steam generator tube rupture, primary coolant will be discharged to the secondary until the primary pressure reduces below the secondary pressure (1100 psia) in the steam generator. This discharge of the primary coolant will result in radioactive release to the environment from direct leakage of the secondary system to the environment or from the overpressure atmospheric reliefs.

Review Criteria

The review criteria for steam generator tube ruptures are presented in SRP Section 15.6.3. Since the major safety concern for this event is the radiological consequences, the systems review is focused on the isolation of the affected generator, on operator actions, and on the potential for fuel damage.

Evaluation

The licensee has performed an analysis to assess the consequences of a steam generator tube rupture (Ref. 1). The initial plant response predicted in the analysis is a rapidly falling pressure and level in the pressurizer which initiates safety injection and reactor trip. The main effect of this early initiation of safety injection and reactor trip is a reduction in primary system pressure, which is required for isolation of the affected steam generator by closure of its steam line isolation valve.

The licensee expects that if the steam line isolation valve is closed after approximately one hour the leakage to the secondary would be about 25% of that assumed in the analysis. The analysis assumes that all of the reactor coolant activity is transferred to the steam system and that the affected steam generator is not isolated for 4-1/2 hours during the cooldown period. The licensee has modified his plant design to initiate safety injection from low pressurizer pressure alone, as opposed to the design identified in the FSAR (Ref. 2) which requires low pressurizer pressure coincident with low pressurizer level. The FSAR analysis is still applicable since this change would only result in a potentially sooner initiation of safety injection. The radiological consequences of a tube rupture are assessed in a separate evaluation.

Conclusion

As part of the SEP review for Ginna we have evaluated the reactor systems aspects of the licensee's analysis against the criteria of SRP Section 15.6.3.

Based on this evaluation we have concluded that the results of the reactor systems aspects of analysis are acceptable.

REFERENCES

1. R. E. Ginna Nuclear Power Plant, Unit 1, Final Facility Description and Safety Analysis Report, Rochester Gas and Electric Corporation, as supplemented.