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RS-00-158

December 21, 2000

United States Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Subject: Response to Request for Additional Information Regarding the License Amendment Request to Permit Uprated Power Operations at Byron and Braidwood Stations

References:

(1) Letter from R. M. Krich (Commonwealth Edison Company) to U.S. NRC, "Request for a License Amendment to Permit Uprated Power Operations at Byron and Braidwood Stations," dated July 5, 2000

(2) Letter from G. F. Dick (U.S. NRC) to O. D. Kingsley (Commonwealth Edison Company), "Byron and Braidwood - Request for Additional Information Regarding the Power Uprate Request," dated October 19, 2000

(3) Letter from R. M. Krich (Commonwealth Edison Company) to U.S. NRC, "Response to Request for Additional Information Regarding the License Amendment Request to Permit Uprated Power Operations at Byron and Braidwood Stations," dated November 27, 2000

(4) Letter from G. F. Dick (U.S. NRC) to O. D. Kingsley (Commonwealth Edison Company), "Byron and Braidwood - Request for Additional Information Regarding the Power Uprate Request," dated November 21, 2000

In Reference 1, we submitted proposed changes to Facility Operating License Nos. NPF-72, NPF-77, NPF-37 and NPF-66, and Appendix A, Technical Specifications (TS), for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively. The proposed changes

A001

would revise the maximum power level specified in each unit's license and the TS definition of rated thermal power.

On September 20, 2000, representatives of Commonwealth Edison (ComEd) Company and the NRC met to discuss technical issues associated with this license amendment request. In Reference 2, the NRC requested that we formally document the information discussed during this meeting along with some additional information in order to complete its evaluation. In a subsequent teleconference on November 8, 2000, the NRC also requested that additional information be provided regarding the documents reviewed in support of the Power Uprate Environmental Assessment. Our response to these requests for additional information was submitted to the NRC in Reference 3.

In Reference 4, the NRC forwarded a second request for additional information to ComEd which addresses issues in a technical area not addressed by Reference 2. Our response to this request for additional information is included in Attachment 1. The NRC requested that this response be submitted by December 21, 2000.

Should you have any questions or concerns regarding this information, please contact Mr. J. A. Bauer at (630) 663-7287.

Respectfully,

R. M. Krich
Director – Licensing

Attachment 1: Response to Request for Additional Information Regarding a License Amendment Request to Permit Uprated Power Operations at Byron and Braidwood Stations

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Braidwood Station
NRC Senior Resident Inspector – Byron Station
Office of Nuclear Safety – Illinois Department of Nuclear Safety

STATE OF ILLINOIS	)	
COUNTY OF DUPAGE	)	
IN THE MATTER OF	)	
COMMONWEALTH EDISON (COMED) COMPANY	)	Docket Numbers
BYRON STATION UNITS 1 AND 2	)	STN 50-454 AND STN 50-455
BRAIDWOOD STATION UNITS 1 AND 2	)	STN 50-456 AND STN 50-457

SUBJECT: Response to a Request for Additional Information Regarding a Previous ComEd License Amendment Request to Permit Uprated Power Operations at Byron and Braidwood Stations

AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information and belief.

R. M. Krich
Director – Licensing

Subscribed and sworn to before me, a Notary Public in and for the State above named, this 21st day of December, 2000

Lisa Ann Delimata
Notary Public

OFFICIAL SEAL
LISA ANN DELIMATA
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES:04/07/01

ATTACHMENT 1

Response to Request for Additional Information Regarding a License Amendment Request to Permit Uprated Power Operations at Byron and Braidwood Stations

In a letter from G. F. Dick (U.S. NRC) to O. D. Kingsley (Commonwealth Edison (ComEd) Company), "Byron and Braidwood - Request for Additional Information Regarding the Power Uprate Request," dated November 21, 2000, the NRC requested that the below additional information be provided.

Note that procedures referenced in this submittal will refer to only Byron Station procedures for simplicity. In all cases, a parallel Braidwood Station procedure exists for use at Braidwood Station.

NRC Question Set H

H.1 *During normal (planned) refueling operations at the Byron and Braidwood plants, entire cores are routinely discharged to the spent fuel pool (SFP). Please provide the following information regarding the SFP cooling assuming a single active failure during the normal (planned) full core offload refueling outages.*

a. *Justifications to demonstrate that the worst single active failure (e.g. failure of an emergency diesel generator, electrical bus, SFP cooling pump, etc.) has been identified.*

H.1.a Response

The Byron Station and Braidwood Station SFP cooling system consists of two independent trains. Each cooling train has its own pump, heat exchanger, piping and valves, and power from an independent power source; however, both trains have a common discharge return to the SFP. A single active failure of any component, other than the common SFP discharge throttle valve, (i.e., valve no. 0FC8754), would only affect one cooling train. This is the original design of the SFP cooling system and was approved during the original licensing of Byron and Braidwood Stations. The power uprate effort made no physical changes to the SFP cooling system. The existing basis and single failure evaluation as described in the Byron and Braidwood Stations Updated Final Safety Analysis Report (UFSAR) were also not impacted. The current basis for the SFP cooling system is to maintain one train of cooling following a single active failure which would render the other cooling train inoperable. As discussed in the response to questions H.2 and H.3, one train of SFP cooling can accommodate the increased heat load from uprated discharged fuel.

b. *Prior to core offload, how many trains of the SFP cooling system are available and operable?*

H.1.b Response

Both trains of the SFP cooling system are normally available prior to core offload. Planned SFP cooling system maintenance is scheduled for non-outage periods. Both trains of SFP cooling are planned to be available throughout an outage, with the exception of Non-Engineered Safety Feature (ESF) electrical bus outage activities which are typically 24 hours in duration and scheduled when core offload/reload activities are not occurring. As stated in the Byron/Braidwood UFSAR, Section 9.1.3.2, the Unit 1 and Unit 2 spent fuel pool pumps are normally powered from respective units non-ESF buses. In an emergency, such as loss of offsite power, the pumps could be powered from one of the two ESF buses by manually aligning the non-ESF bus with one of the ESF buses.

One train of SFP cooling is verified to be operating during the equipment attendant daily rounds. The essential service water system provides cooling to the component cooling water system which in turn provides cooling to the SFP cooling system heat exchangers. Typically, only one SFP cooling train is operating due to the cool essential service water temperature during the typical spring or fall refueling outages at Byron Station and Braidwood Station. To maintain the temperature and clarity of the SFP, and habitability of the Fuel Handling Building (FHB) during fuel movement, the operator will normally start the second SFP cooling train to maintain the temperature of SFP water ≤ 90°F in accordance with Byron Station Operating Procedure (BOP) Spent Fuel Pool Cooling (FC)-1, "Fuel Pool Cooling System Start-Up and SFP Purification System Operation."

c. *A curve to show the SFP temperature as a function of time during the planned refueling outage with a full core offloaded to the SFP.*

H.1.c Response

Figure 1, "Bulk Spent Fuel Pool Temperature Following Full Core Offload with One Train of Cooling," is attached showing the SFP temperature as a function of time. The attached curve is a mark-up of the curve developed for the recent SFP rerack License Amendment Request (LAR) showing the impact due to the power uprate. This LAR was approved by the NRC in a letter from G. F. Dick (U.S. NRC) to O. D. Kingsley (ComEd), "Byron and Braidwood – Issuance of Amendments on Spent Fuel Storage Racks," dated March 1, 2000, (i.e., Reference 2 from the "Question Set H References" listed below).

H.2 *ComEd considered a full core off load, which causes the SFP temperature to exceed the Standard Review Plan (SRP) guidance for SFP temperature limit for a duration over 500 hours, during normal (planned) refueling outages as a temporary condition. In order to determine whether adequate SFP cooling exists to ensure that the intent of SRP, Section 9.1.3, guidance is met, the staff requests a discussion of the rationale for establishing a full core off load during planned (normal) refueling outages as a temporary condition. Please provide and justify the acceptance criteria for defining a full core off load during normal refueling outages as a temporary condition.*

H.2 Response

During a normal refueling outage, the full reactor core is initially off-loaded to the SFP. The core off-load typically takes approximately 36 hours. Two-thirds of the assemblies are subsequently reloaded in the reactor vessel starting approximately 10 – 20 hours after the core off-load is completed. The core reload again takes approximately 36 hours. This total period of time (i.e., approximately four days) is considered the temporary condition in which the full core decay heat load is being removed by the SFP cooling system. This is consistent with UFSAR, Section 9.1.3.1, "Spent Fuel Pool Cooling." This position was approved by the NRC in Reference 2. In addition, during core offload and reload, the SFP water volume is in "communication" with the reactor cavity water volume through the transfer canal. The reactor cavity is cooled via the Residual Heat Removal (RHR) system in the shutdown cooling mode during this time period.

UFSAR, Section 9.1.3.1, states that, "Refueling operations are routinely performed in either an approximate one-third core offload or a full core temporary offload where approximately two-thirds of the fuel assemblies are returned to the reactor vessel, along with the new fuel, prior to the end of the outage. A third refueling mode, dual unit discharge, would be an abnormal circumstance, but is also considered in the analysis. The one-third core off-load and the full core off-load cases are analyzed assuming single failure of one train of spent fuel pool cooling. The dual-unit discharge case is analyzed assuming both trains of spent fuel pool cooling are available."

The below table summarizes the acceptance criteria for SFP temperature given in the Standard Review Plan (SRP), NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," June 1987, and the analysis results given in the "pre-uprate" UFSAR, and the proposed post-uprate UFSAR revision.

Table H.2
Spent Fuel Pool Temperature Analysis and Requirements

Scenario	Single Active Failure Assumed*	SRP Requirements	UFSAR Pre-Uprate Analysis	UFSAR Uprate Analysis
1/3 Core Offload	Yes	Less than 140°F	138.3°F	141.2°F
Full- core Offload**	Yes	N/A	157.1°F	162.7°F
Full- core Offload	No	Less than 212°F	N/A	133.8°F
Back-to Back Refueling	No	Less than 212°F	137.2°F	139.4°F

* Yes – assumes single train of SFP cooling
No – assumes two trains of SFP cooling

** Temporary full Core offload during refueling in which two-thirds of the core is returned to reactor vessel approximately four days after start of core offload. Calculated bulk SFP water temperature exceeds 140°F during this temporary condition with a single active failure but will remain below 140°F with two trains of SFP cooling.

H.3 *During normal (planned) refueling outages with a full core discharged to the SFP, the calculated SFP peak temperature (assuming a single active failure) is 162.7°F which exceeds the guidance of 140 °F described in Standard Review Plan (SRP) Section 9.1.3 for SFP temperature, and the guidance of 150°F described in American Concrete Institute (ACI) 349-97, "Nuclear Safety Structures," for concrete structures. The durations of the SFP temperature exceeding the SRP guidance and ACI guidance are expected to be approximately 580 hours and 120 hours, respectively (Figure 5.8.2 of Attachment E of ComEd's submittal of March 23, 1999, "Request for an Amendment to Technical Specifications to Support Installation of New Spent Fuel Pool Storage Racks at Byron and Braidwood Stations"). Please provide the following information:*

a. A *detailed discussion to justify why the higher pool temperature of 162.7°F is acceptable during planned refueling outages.*

H.3.a Response

A full core off-load produces a maximum bulk pool temperature of 162.7°F assuming a single active failure resulting in the loss of one train of SFP cooling. This temperature is calculated using conservative assumptions and is based on the final fuel off load with the SFP filled to capacity. For the case of a full core discharge with two heat exchangers operable, the maximum temperature 100 hours after shutdown would be 133.8°F. The temperatures during a normal refueling are not expected to peak above 140°F and the SFP temperature alarm is set at 149°F to alert operators of abnormal condition, such as a loss of SFP cooling. Assuming a single failure following a full core offload, the calculated SFP temperature

exceeds the SRP guidance for approximately four days. However, the full core is in the SFP a small time period when compared to the time the one-third core will be in the SFP. The guidance to maintain the SFP below 140°F will be met during the time the pool contains fuel with the exception of the approximately four days between full core offload and the subsequent reload. The SRP guidance allows the SFP to exceed 140°F following an abnormal full core offload. The potential risk and subsequent impact of losing a single train of SFP cooling during the period when the "temporary" full core offload is in the SFP, is consistent with this SRP acceptance criteria. Due to an unforeseen circumstance, if the discharged full core needs to remain in the SFP for longer than the nominal four days between core offload and reload, the unavailable SFP cooling train, which was assumed to be unavailable due to a single active failure, would be repaired and made available. This second SFP cooling train would be put into operation to reduce the SFP temperature to less than 140°F for long term operations.

In addition, the concrete temperature will not be uniformly elevated to the maximum bulk pool temperature of 162.7°F. The heat that is transferred from the SFP water produces a thermal gradient across the thickness of the concrete wall/slab. The average temperature associated with this gradient will be below the ACI limit of 150°F. Only a thin layer of concrete on the interior surface may experience temperatures above 150°F for a short duration.

b. *The effects of the higher pool temperature during this duration on equipment and systems.*

H.3.b Response

The SFP cooling system is designed to remain functional during and following a seismic event, and the fuel pool cooling system is designed for 200°F. The SFP has been analyzed to withstand the stresses associated with a steady state temperature gradient across the thickness of the concrete wall/slab from 70°F to 162.7°F as shown in Attachment A (i.e., Calculation No. 8.1.12-BRW-96-754 and supplemental Document Change Request (DCR) #990668). The impact of the SFP temperature of 162.7°F on the SFP structure is evaluated and discussed in the response to question H.3.c below.

c. *Detailed discussion of the thermal stress analyses (e.g., assumptions, analytical models, etc.) of the pool structures should be provided in Section 9.5.3, "Spent Fuel Pool."*

H.3.c Response

The current analysis of the spent fuel pool structure for increased temperatures, as approved by the NRC in Reference 2 the "Question Set H References" listed below, is used as a basis to determine the consequences of higher spent fuel pool temperatures due to power uprate.

The increased temperatures reflecting these conditions are evaluated using latest plant analysis methods and Reference 1. This analysis (i.e., Calculation No. 8.1.12-BRW-96-754 and supplemental Document Change Request (DCR) #990668) is included as Attachment A, "Evaluation of Spent Fuel Pool for Elevated Temperature." Although this is a Braidwood Station calculation, it is also applicable to the Byron Station SFP. We do not plan to revise Section 9.5.3 of the Licensing Report to incorporate this analysis.

The plant analysis identifies the thermal expansion (i.e., axial expansion) of the walls and slab as the load contributing to the maximum (i.e., limiting) bending moment in the pool walls. Assuming that this moment is proportional to the increase in the average temperature of the wall (note that the thermal gradient is treated separately), the maximum rebar stress is estimated for the case when the peak temperature is 162.7°F for full core discharge and the ambient temperature on the exterior of the SFP is 70°F.

- Pre-Uprate Rebar Stress

  The increase in the average concrete wall temperature is (158°F - 70°F)/2 or 44°F and the maximum rebar stress is 50.9 ksi. The allowable stress for the SFP grade 60 reinforcing steel is 54.0 ksi.

- Post-Uprate Rebar Stress

  The increase in the average concrete wall temperature for the power uprate conditions only is 46.4°F, i.e., (162.7°F - 70°F)/2. This is an increase in the average temperature of 5.45%. A proportionate increase in the maximum rebar stress results in a stress of 53.7 ksi. As previously noted, the allowable stress for the SFP grade 60 reinforcing steel is 54.0 ksi.

Thermal gradients were evaluated and do not change the reinforcement stresses when cracking of the cross-section is considered.

The maximum temperature of 162.7°F in the spent fuel pool only occurs for a full core offload and assumes a single active failure. Since this is considered a temporary offload, the SFP temperature for the long term remains below 150°F. The power uprate scenario for the full core offloads results in the maximum temperature condition. The revised post-uprate maximum steel reinforcing stress is 5.5% higher than the pre-uprate analysis; however, the stress levels remain within allowable limits. These local overstresses do not significantly reduce the factor of safety for the SFP structure. As noted in the response to question H.2, if the discharged full core needs to remain in the SFP, due to an unforeseen circumstance, for longer than the nominal four days between core offload and reload, the unavailable SFP cooling train, which was assumed to be unavailable due to a single active failure, would be repaired and made available. This second SFP cooling train would be put into operation to reduce the SFP temperature to less than 140°F for long term operations. This will limit the increased thermal stress to a relatively short period of time.

Considering an ambient temperature of 70 °F on the exterior of the pool, the average concrete temperatures of the walls and bottom slab, considering a peak SFP water temperature, are well below 150°F.

Reference 1 provides the following limitations for "normal operations or other long term period. The temperatures shall not exceed 150°F ... except for local areas... which are allowed to have increased temperatures not to exceed 200°F." The definition of normal operations of the spent fuel pool is discussed in UFSAR Section 9.1.3.1, "Spent Fuel Pool Cooling". A full core off load is a temporary condition. Refueling operations are routinely performed in either an approximate one-third core offload or a full core temporary offload where approximately two-thirds of the fuel assemblies are routinely returned to the reactor vessel, along with new fuel, prior to the end of the outage. Since the decay heat of the fuel (i.e., the spent fuel pool heat load) decreases exponentially, the SFP temperature remains below 150°F for long term operations.

The maximum concrete temperature will be less than the 200°F limit for local areas during normal operations and therefore meets the acceptance criteria of Reference 1.

The changes in the spent fuel pool temperature loading due to power uprate result in concrete temperatures and reinforcement stresses which meet the structural acceptance criteria.

H.4 *In order to determine whether adequate controls exist to ensure the guidance of Standard Review Plan, Section, 9.1.3, are met, the staff needs to understand the provisions established or to be established in plant operating procedures to monitor and control the SFP water temperature during full-core offload events. Please provide the following information:*

a. *The frequency that the local temperature indicators for SFP water temperature will be monitored.*

H.4.a Response

In accordance with procedure BOP 199-Equipment Attendant (EA) A2, "U-2 Aux Bldg Equipment Attendant Daily Logs," the SFP temperature is monitored locally once per day. Also, as indicated in the response to question H.1.b, when the temperature or clarity of the pool starts to decrease, or when the FHB habitability needs improvement, the operator will check the local temperature indicators to determine the pool temperature, and start the other SFP cooling train, if necessary, to bring the pool temperature to ≤ 90°F. Experience has shown that visibility in the SFP starts to degrade when the water temperature increases above approximately 105°F. At this point, fuel handling personnel may request that operations personnel start the second train of SFP cooling to improve the visibility.

b. *The set-point of the high water temperature alarm for the SFP.*

H.4.b Response

As specified in the Byron Annunciator Response procedure BAR 1/2-1-B1, "Spent fuel Pit Temp High," SFP high temperature is annunciated in the Main Control Room (MCR) by Annunciator 1/2-1-B1, "Spent Fuel Pit Temp High," at 149°F.

c. *Information supporting a determination that there is sufficient time for operators to intervene in order to ensure that the temperature limit of 150°F will not be exceeded.*

H.4.c Response

As discussed in the response to question H.4.a, the spent fuel pool temperature is typically maintained below 90°F when fuel is being discharged into the SFP. The spent fuel pool water temperature will increase if there is a loss of cooling caused by spent fuel cooling pump trip. Loss of a Spent Fuel Pool Cooling Pump is annunciated in the MCR by Annunciator 1/2-1-A1, "Spent Fuel Pit Pump Trip." The annunciator response procedure directs monitoring of the SFP temperature and directs restart of the pump if no visible damage; otherwise directs starting of the opposite unit's pump. If no spent fuel pit cooling can be started due to power supply failure, the operator will locally align alternate power supply to spent fuel cooling pumps. An alternate cooling path by aligning the refueling water purification pumps to the spent fuel pit heat exchanger can also be used to cool the spent fuel pool.

Adequate time is available to accomplish the above activities based on the following assumptions. As documented in the Power Uprate Licensing Report, Table 9.3.10-1, "Spent Fuel Pool Evaluation Results," for a full core discharge, the time to SFP boil is 3.055 hours after loss of all SFP cooling assuming an SFP maximum starting temperature of 162.7°F. This heatup rate is 16.13 °F/hr. Assuming this same heatup rate, it would take approximately 3.7 hours to heat up the SFP from 90°F to 150°F. This amount of time (i.e., 3.7 hours) is sufficient for operators to accomplish the above compensatory actions.

d. *The mitigative actions (i.e. prohibit fuel handling, aligning other systems to provide SFP cooling, etc.) to be taken in the event of a high SFP water temperature alarm.*

H.4.d Response

A high temperature condition in the SFP would result in actuation of MCR Annunciator 1/2-1-B1, "Spent Fuel Pit Temp High," at 149°F. The annunciator response procedure directs the operator to:

1) check adequate component cooling water flow to the SFP heat

exchangers;

2) verify that the pumps are supplying the in-service SFP cooling loop; and

3) throttle the SFP heat exchanger component cooling water outlet flow control valve to obtain a maximum of 5500 gpm and a minimum of 65 psig discharge pressure.

4) if temperature does not return to normal, operators are directed to start the opposite unit's SFP cooling loop.

As noted in the response to question H.4.c, alternate cooling to the SFP can be provided by aligning the refueling water purification pumps to the spent fuel pit heat exchanger. In addition, upon receiving a SFP high temperature alarm, it is expected that all fuel movement in the SFP will be stopped.

Question Set H References

1. American Concrete Institute (ACI) 349-97, Nuclear Safety Structures, Commentary on Appendix A - Thermal Considerations

2. Letter from G. F. Dick (USNRC) to O. D. Kingsley (ComEd), "Byron and Braidwood – Issuance of Amendments on Spent Fuel Storage Racks," dated March 1, 2000

NRC Question Set I

I.1 *For the loss of coolant accident (LOCA) and main steamline break (MSLB) Containment Analyses (Sections 6.4 and 6.5 of uprate report), please indicate key input parameters that are different from [the] updated final safety analysis report (UFSAR) besides power related and the effect on the peak containment pressure and temperature.*

I.1 – Response for LOCA:

Several tables comparing key parameters, including differences in the evaluation model, have been prepared. These tables compare only the key parameters which changed between the UFSAR and the uprate analysis. Further, parameters separated by a slash represent unit 1 and unit 2 values (e.g. U1/U2). A review shows that many of the parameters result in a penalty under uprated conditions. However, the change in the evaluation model from the one presented in WCAP-8264, "Topical Report, Westinghouse Mass and Energy Release Data for Containment Design," Revision 1, August 1975, to the newer model described in WCAP-10325, "Westinghouse LOCA Mass and Energy Release Model for Containment Design – March 1979 Version," May 1983, resulted in a net benefit for the Byron and Braidwood Stations uprate analysis. The model change was the dominant factor in the results obtained for the Byron and Braidwood Stations uprate analysis.

BYRON/BRAIDWOOD UPRATE

TABLE OF
THERMAL/HYDRAULIC SYSTEM PARAMETERS AND
INITIAL CONDITIONS FOR COMPARISION OF
UFSAR VS UPRATE

PARAMETERS	VALUE		DIRECTION OF CONSERVATISM FOR THE UPRATE
	UFSAR	UPRATE	
Core Thermal Power (MWt) w/o calorimetric error	3579	3586.6	PENALTY
Vessel Average Temperature (°F)	594.898	598.0	PENALTY
Initial Steam Generator Secondary Side Mass (lbm)	121550	136617.8 / 106484	PENALTY FOR BWI** / BENEFIT FOR W D5***
SG Total Dry Weight (Lbm)	801,637 / 702,000	808,000 / 703,000	PENALTY
Accumulator			
Water Volume (ft^3) per accumulator	950	1005.9	PENALTY
N_2 Cover Gas Pressure (psia)	600	661.7	PENALTY
Accumulator Nitrogen Gas Addition	Not Modeled	4509 Lbms Added	PENALTY
Safety Injection Delay, total (sec) (from beginning of event)	25.0	40.0	PENALTY
Fuel Array and Rod Diameter*	17X17 / 0.374	17X17 / 0.360	PENALTY
Core Stored Energy (Full Power Seconds)	4.81	3.69	BENEFIT

* The use of Optimized Fuel Assemblies (OFA) and Vantage 5 fuel with a smaller rod diameter was considered to be a benefit and therefore the UFSAR specific information was not updated to reflect these fuel types.

** BWI refers to the Babcock and Wilcox International replacement steam generators.

*** W D5 refers to the Westinghouse model D5 steam generators.

BYRON/BRAIDWOOD UPRATE

TABLE OF
EVALUATION MODEL DIFFERENCES FOR
COMPARISION OF UFSAR VS UPRATE

PARAMETERS	VALUE		DIRECTION OF CONSERVATISM FOR THE UPRATE
	UFSAR WCAP-8264	UPRATE WCAP-10325	
Core Thermal Power	ESDR*	LICENSED	BENEFIT
Two phase Flow Model	Homogenous	Drift Flux	More Representative
Momentum Flux	None	Added	More Representative
Two phase Reactor Coolant Pump Model	Simple	Dynamic	More Representative
Core Heat Release Model	External Calc	Internal Calc	More Representative
Thin Metal Heat Release	External Calc	Internal Calc	More Representative
Wall Heat Transfer Correlation	Jens-Lottes	Thom	BENEFIT
Core Film Boiling Heat Transfer Correlation	Dougall-Rohsenow	Westinghouse Transition Boiling	BENEFIT - Added Accuracy
Decay Heat	ANS 1971+20%	ANS 1979+2σ	BENEFIT
Steam / Water Interaction in RCS Loop Piping	No	Yes	BENEFIT
Initial Core Stored Energy	+20%	+15%	BENEFIT
Depressurization and Equilibration of SGs	External Calc	Internal Calc	Simplified Calculational Stream - Added Accuracy

* Engineered Safeguard Design Rating (EDSR)

BYRON/BRAIDWOOD UPRATE TABLE OF CONTAINMENT ASSUMPTIONS AND ENGINEERED SAFEGUARDS PERFORMANCE UFSAR VS UPRATE			
PARAMETERS	VALUE***		DIRECTION OF CONSERVATISM FOR THE UPRATE
	UFSAR	UPRATE	
Containment and Engineered Safety Feature (ESF) Initial Conditions			
Initial containment pressure (psia)	15.7 / 15.7**	15.7	NO CHANGE / PENALTY
Reactor Containment Air Recirculation Fan Coolers			
Containment Hi-1 setpoint (psig)	NA	6.8	ADDED ANALYSIS ACCURACY
Delay time (sec)			
With Offsite Power	65.0#	27.0	BENEFIT
Without Offsite Power	65.0#	65.0	UNCHANGED
Fan Cooler Heat Removal Rates were Unchanged			
Containment Spray Pumps			
Spray Pump Performance and Associated Assumptions were Unchanged			
Containment Spray Actuation and Delays			
Containment Hi-3 setpoint (psig)	NA	24.8	ADDED ANALYSIS ACCURACY
Delay time (sec)			
With Offsite Power (delay after High High setpoint)	88.1#	53.1	BENEFIT
Without Offsite Power (total time from t=0)	88.1#	88.1	PENALTY
Containment Spray Recirculation Switchover, (sec)			
Minimum Safeguards	2893#	3778	BENEFIT
Maximum Safeguards	1885#	3363	BENEFIT

BYRON/BRAIDWOOD UPRATE TABLE OF CONTAINMENT ASSUMPTIONS AND ENGINEERED SAFEGUARDS PERFORMANCE UFSAR VS UPRATE			
PARAMETERS	VALUE***		DIRECTION OF CONSERVATISM FOR THE UPRATE
	UFSAR	UPRATE	
Emergency Core Cooling System (ECCS) Flows (GPM)			
Minimum ECCS			
Injection alignment (gpm)	5049#	5686	BENEFIT
Recirculation alignment (gpm)	4504.69#	994.1	PENALTY
Maximum ECCS			
Injection alignment (gpm)	10672#	12305	BENEFIT
Recirculation alignment (gpm)	9190.83#	11917.1	BENEFIT
ECCS Recirculation Switchover, sec			
Minimum Safeguards	2893#	1110	PENALTY
Maximum Safeguards	1500 / 1885#	695	PENALTY
Residual Heat Removal (RHR) System			
The RHR System Parameters remain Unchanged			
Component Cooling Water Heat Exchangers			
The Component Cooling Water Parameters remain Unchanged			
Additional Heat Loads, (BTU/hr)			
Miscellaneous Heat Load on the CCW	$2.0X10^6$	$6.8X10^6$	PENALTY
Containment Heat Sinks			
Containment Heat Sinks for the Uprate are identical to the UFSAR			

** The Unit 2 value listed in UFSAR Table 6.2-1a is 15.0 psia, however, Unit 2 was evaluated for operation at 15.7 psia initial pressure as indicated in the UFSAR table foot note.

*** Differences in values are attributed to the different evaluation models.

These values are applicable to Byron Station, Units 1 and 2 and Braidwood Station, Units 1 and 2. The current UFSAR has been updated to reflect these values for Unit 1 only.

I.1 *For the loss of coolant accident (LOCA) and main steamline break (MSLB) Containment Analyses (Sections 6.4 and 6.5 of uprate report), please indicate key input parameters that are different from [the] updated final safety analysis report (UFSAR) besides power related and the effect on the peak containment pressure and temperature.*

I.1 – Response for Main Steam Line Break (MSLB)

Revision 8 of the UFSAR was recently submitted to the NRC in a letter from R. M. Krich to the NRC, "Byron and Braidwood Stations Updated Final Safety Analysis Report, Revision 8," dated December 16, 2000. Compared to the UFSAR, Revision 8, MSLB mass/energy release and containment response analyses for Byron and Braidwood Stations, the input parameters for the power uprate analysis were changed primarily for power-related issues. These items include changes in nominal steam generator pressures and temperatures as appropriate. The changes in these nominal conditions also slightly changed the timing of when different protection setpoints are reached, which also has the effect of changing the break size area that defines the split rupture cases. Since the limiting (i.e., the largest) split rupture is defined as the break area that does not produce a secondary-side protection signal, the area is increased to account for the reduction in the initial pressure in the steam generator at the time of the postulated break. The increased break size coupled with the lower steam pressure has offsetting effects such that the containment response is similar.

There is only one change made in the input parameters that is not an effect of the power uprating. For full double-ended rupture (DER) cases, the previous analyses had based the reverse break flow on the cross-sectional area occurring at the integral flow restrictors in each unfaulted steam generator. However, when the input was being developed for the power uprate program, it was identified that the cross-sectional area of the Main Steam Isolation Valve (MSIV) on the faulted loop would limit the reverse break flow. This is a smaller effective area (i.e., 2.64 ft^2) than the sum of the integral flow restrictor cross-sectional areas for the three unfaulted loops, and has two effects on the mass/energy releases. It reduces the reverse break flowrate from the three unfaulted steam generators. It also modifies the treatment of the steam within the steamline piping. A full DER typically results in the rapid blowdown of the initial mass in the steamline piping, since the pipe break area is larger than the combined flow restrictor area on the three unfaulted loops. However, with the reverse flow area defined at the faulted-loop MSIV, most of the mass in the steamline piping affects the break blowdown after steamline isolation (i.e., conservatively assuming a failure of the MSIV on the faulted loop for all full DERs). The input changes to model the reverse break flow, based on the MSIV flow area, were made to more accurately reflect the physical configuration of the plant. The effect on the containment response is to lower the temperature and pressure.

For the containment response analysis following the MSLB, there are no different input parameters used in the calculations.

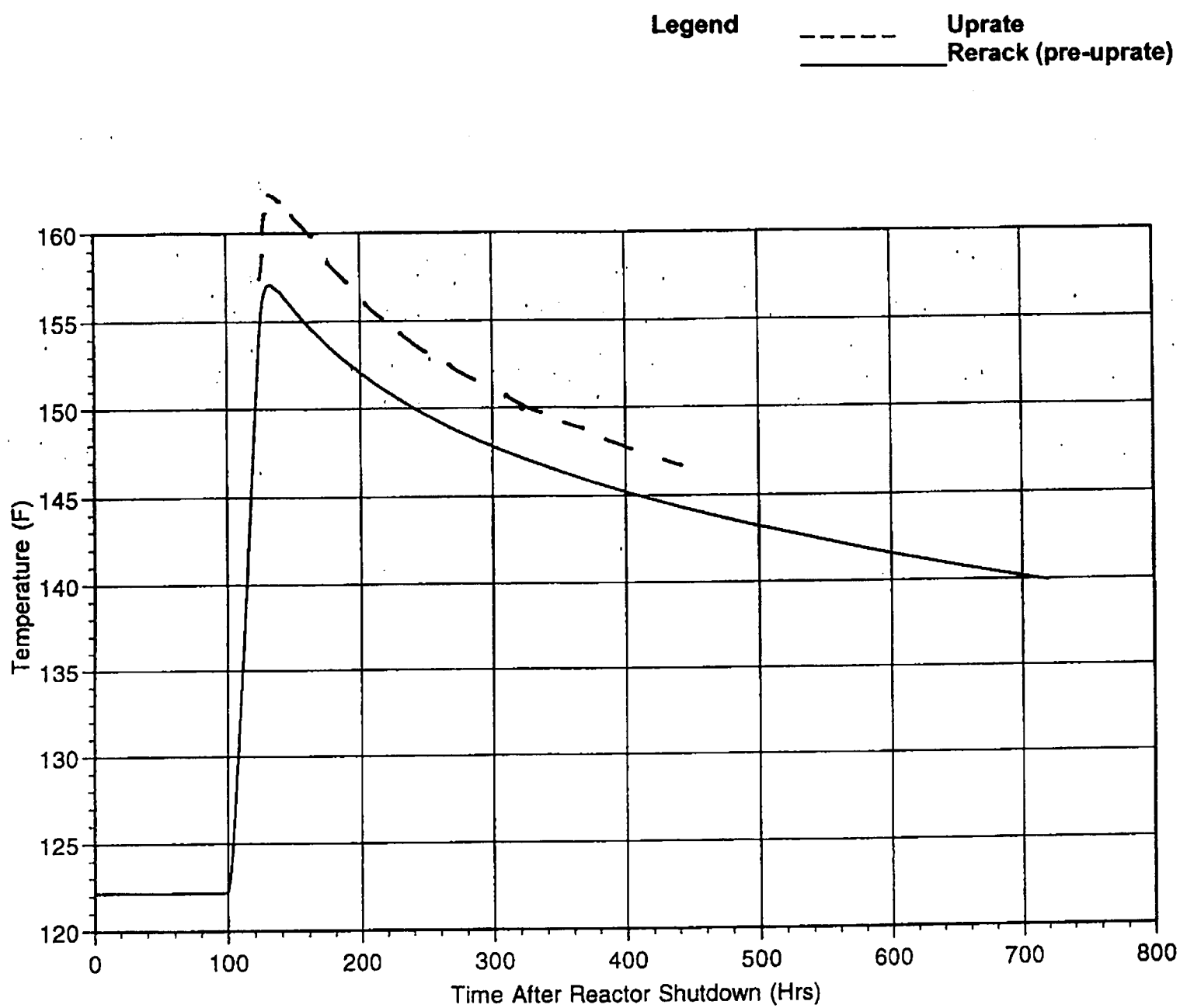

FIGURE 1 Bulk Spent Fuel Pool Temperature following Full Core Offload with One Train of Cooling

ATTACHMENT A

Calculation No. 8.1.12-BRW-96-754
"Evaluation of Spent Fuel Pool for Elevated Temperature"

and

Document Change Request (DCR) #990668 to Calculation No. 8.1.12-BRW-96-754

Exhibit C
NEP-12-02
Revision 3

Calculation Title Page

Calculation No.: *8.1.12-BRW-96-754* Page No.: *1*

☒ Safety Related ☐ Regulatory Related ☐ Non-Safety Related

Calculation Title:

Evaluation of Spent Fuel Pool for Elevated Temperature

Station/Unit: *Braidwood - Units 1 & 2* System Abbreviation: *N/A*

Equipment No. (if appl.): *N/A* Project No. (if appl.) *09050-051*

Rev.: *0* Status: APPROVED QA Serial No. or Chron No. N/A Date: 11/27/96

Prepared by: Chris Sward Chris Sward Date: 11/26/96

Revision Summary:

This is the original issue of this calculation.

Pages 1 - 11, A1 - A3, B1 - B3, C1 were added.

Electronic Calculation Data Files Revised:
(Name ext/size/date/hour/: min/verification method/remarks)
None

Do any assumptions in this calculation require later verification? ☐ Yes ☒ No

Reviewed by: Yau Wang Yau Wang Date: 11/27/96

Review Method: *Detailed* Comments (C, NC or CI): NC

Approved by: R. Montgomery R. MONTGOMERY Date: 11/27/96

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CALCULATION NO. *8.1.12-BRW-96-754*	REV. NO. *0*	PAGE NO. *2*
DESCRIPTION: *Evaluation of Spent Fuel Pool for Elevated Temperature*	PAGE NO.	SUB-PAGE NO.

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PURPOSE/OBJECTIVE:

The purpose of the calculation is to demonstrate the adequacy of the spent fuel pool for the following pool temperature conditions:

1. 138°F resulting from a normal core discharge (i.e. 1/3 of fuel)
2. 155°F resulting from a full core discharge
3. 158°F resulting from an abnormal discharge (i.e. normal discharge followed shortly after by a full core discharge)

The existing design of the spent fuel pool (Reference 1) is based on a maximum pool temperature of 150°F.

METHODOLOGY & ACCEPTANCE CRITERIA:

The spent fuel pool walls and slab will be evaluated for the elevated temperatures by extrapolating from the analysis in the existing design basis (Reference 1) and verifying that the resulting stresses are within the design basis allowables. This will be accomplished by identifying the critical element(s) from the previous design calculations, increasing the element stresses for the temperature related basic loads, recomputing the load combinations, and determining the revised maximum stresses.

Case 1 (138°F) is a normal operating condition and thus will be treated as T_o. Case 2 (155°F), while not an ordinary occurrence will conservatively be treated as a normal load, T_o. Case 3 (158°F) is an abnormal load, T_a, but may conservatively be treated as T_o.

ASSUMPTIONS:

None.

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DESIGN INPUT:

The results of the finite element analyses performed in Reference 1 will be used as input to this calculation. The following run IDs will be used:

494RCD, dated 8/5/76, SLSAP analysis for basic loads 1 & 2

906RCD, dated 8/5/76, SLSAP analysis for basic loads 4, 5, 6 & 7

445RCD, dated 8/9/76, SLSAP analysis for basic loads 8 & 9

897RCD, dated 8/5/76, SLSAP analysis for basic loads 10 & 11

808RCD, dated 8/10/76, SLSAP post processor to compute combined stresses

The following loads and combinations have been defined in these analyses:

Basic loads:

BL1: Liner expansion
BL2: Horizontal hydrostatic pressure
BL3: Thermal gradient - computed as a basic load but not included in load combinations because the SLSAP analysis is overly conservative - see discussion under "Background"
BL4: Axial expansion
BL5: Dead load (racks + pool + vertical hydrostatic pressure)
BL6: Dead loads for combination with accident cask drop
BL7: Accident cask drop
BL8: E-W OBE
BL9: E-W SSE
BL10: N-S OBE
BL11: N-S SSE

Load combinations:

Normal:	1.4D + 1.7L + 1.3T
Abnormal (cask drop):	1.0D + 1.0L + 1.0T
Severe envir. (OBE):	1.4D + 1.7L + 1.3T + 1.9E_O
	1.2D + + 1.3T + 1.9E_O
Extreme envir. (SSE):	1.0D + 1.0L + 1.0T + 1.0E_S

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From these load combinations, the following load factors were developed:

		BL1	BL2	BL4	BL5	BL6	BL7	BL8	BL9	BL10	BL11
1	Normal	1.3	1.4	1.3	1.4						
2	Normal	1.3	1.4	1.3		1.4					
3	Abnormal	1.0	1.0	1.0		1.0	1.0				
4	Severe Envir.	1.3	1.4	1.3	1.72			1.9		1.9	
5	Severe Envir.	1.3	1.4	1.3	1.72			1.9		-1.9	
6	Severe Envir.	1.3	1.4	1.3	1.72			-1.9		1.9	
7	Severe Envir.	1.3	1.4	1.3	1.72			-1.9		-1.9	
8	Severe Envir.	1.3	1.2	1.3	1.52			1.9		1.9	
9	Severe Envir.	1.3	1.2	1.3	1.52			1.9		-1.9	
10	Severe Envir.	1.3	1.2	1.3	1.52			-1.9		1.9	
11	Severe Envir.	1.3	1.2	1.3	1.52			-1.9		-1.9	
12	Extreme Envir.	1.0	1.0	1.0	1.32				1.0		1.0
13	Extreme Envir.	1.0	1.0	1.0	1.32				1.0		-1.0
14	Extreme Envir.	1.0	1.0	1.0	1.32				-1.0		1.0
15	Extreme Envir.	1.0	1.0	1.0	1.32				-1.0		-1.0

Note: The input seismic loads (BL8 - BL11) do not include acceleration of the dead loads in BL5. This acceleration is included by deriving load factors as follows:

LC4-LC7: $1.4D + 1.9E_o = 1.4(BL5) + 1.9(g_{vOBE})(BL5) = 1.72(BL5)$

LC8-LC11: $1.2D + 1.9E_o = 1.2(BL5) + 1.9(g_{vOBE})(BL5) = 1.52(BL5)$

LC4-LC7: $1.0D + 1.0E_s = 1.0(BL5) + 1.0(g_{vSSE})(BL5) = 1.32(BL5)$

where $g_{vOBE} = 0.17$ & $g_{vSSE} = 0.32$

REFERENCES:

1. Calculation 8.1.12, "Fuel Handling Building Spent Fuel Pool Analysis and Design," Revision 6.

2. DC-ST-03-BY/BR, "Byron/Braidwood Structural Project Design Criteria," Revision 22.

3. SLSAP and SLSAP Post processor Computer Program, S&L Program No. 09.7.130-4.0O.

4. TEMCO Computer Program, S&L Program No. 09.7.072-5.2O.

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CALCULATIONS:

Background

The original analysis of the pool walls and slab was performed in a three step process. First, a finite element analysis was performed for 11 basic load cases using the SLSAP computer program. SLSAP performed a linear finite element analysis of the pool model to determine forces on the plate elements. Second, the basic load cases were combined for all elements using the SLSAP post processor. Third, eight critical elements were selected and analyzed using the TEMCO computer program. The combined element forces and thermal gradient were input to TEMCO which performed a cracked section analysis to determine design stresses. Temperature effects were included in this analysis process as follows:

1. Liner Expansion

   As the liner heats up it expands and exerts a force at the face of the walls & slab through its anchorage. The liner expansion forces were computed for a temperature rise of 80°F based on a maximum pool temperature of 150°F and the base temperature of 70°F. The reactions from the liner were input to the finite element analysis (FEA) as nodal forces and moments (due to the eccentricity of the applied force).

2. Axial Expansion

   Axial expansion was input to the FEA in terms of the temperature rise for each element. The temperature rise was computed assuming a linear gradient across the element. The inside temperature was taken as 150°F and the outside temperature was taken as 70°F for the walls and 50°F for the slab. The temperature rise was taken as half of the gradient.

3. Thermal Gradient

   The thermal gradient across the thickness of the concrete (hot on the pool side and cool on the other) causes bending stresses in the element. Thermal gradients were computed by TEMCO based on temperature inputs at each face of the element. TEMCO was utilized because it performs a cracked section analysis which will produce more realistic results than the overly conservative analysis of thermal gradient in the linear finite element analysis.

   For wall elements, the temperature gradient was input by specifying 70°F at one face and 174°F at the other. For the slab (and elements along the base of the walls where there is soil on the outside face), these temperatures were input as

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50°F and 180°F. These values were determined by increasing the actual gradient to account for load factors as follows:

Walls: Actual gradient = 150° - 70° = 80°F
Factored gradient = 1.3 x 80° = 104°F

Slab: Actual gradient = 150° - 50° = 100°F
Factored gradient = 1.3 x 100° = 130°F

This was done in lieu of load combinations since the combined element forces that were input TEMCO were already factored.

Subsequent analyses were performed to evaluate the pool for additional loads resulting from the change to high density fuel racks (Reference 1, Section 8.1.12.10). In these analyses, additional loadings due to the racks were computed and combined with the previous SLSAP results. These revised loadings were then run in TEMCO with the temperature gradients noted above. The results of this analysis for the critical elements along with design margins is provided in Attachment A (copied from Reference 1, Section 8.1.12.10, page 162).

The lowest factor of safety from this analysis is 1.27 for rebar stress in element #7. This element is located on the west wall below the fuel handling building base slab. This element corresponds to the following designations:

Element #7	finite element model (Ref. 1, Section 8.1.12.2)
Section #4	original TEMCO analyses (Ref. 1, Section 8.1.12.5).
Section #2	new TEMCO analysis for high density racks (Ref. 1, Section 8.1.12.10)

Analysis for Revised Temperatures:

For the element indicated above, the increased pool temperature will be used to recompute the combined element forces which will then be input to TEMCO for a new analysis with the corresponding larger gradient. This evaluation will be done for a temperature of 158°F treated as a normal temperature. This will be conservative and will envelope the other two temperature cases.

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The element forces for element #7 from the original analysis are shown Table 1. The first half of the table lists the forces for each basic load as determined by the SLSAP analyses indicated. The second half of the table computes combined element forces according to the load combinations specified under Design Inputs. This computation of combined forces is verified by comparison of the results to the SLSAP post processor run (run id 808RCD).

Table 1
Element # 7 - Forces from Original Analysis

Basic Loads:		Membrane Stresses (kips/ft^2)			Bending Moments (kip-ft)		
	Run ID	SXX	SYY	SXY	MXX	MYY	MXY
BL1	494RCD	0.7604	3.4750	0.0566	-1.6930	-10.9400	0.0991
BL2	"	0.1477	0.2398	0.0339	1.9640	8.8710	0.1118
BL3	Not Used						
BL4	906RCD	-128.5000	9.7630	2.2810	-113.9000	-716.8000	4.7630
BL5	"	0.9508	2.3670	-0.5253	1.1080	10.6600	-0.3801
BL6	"	0.9734	2.3940	-0.5183	1.1710	10.6500	-0.3356
BL7	"	3.4480	6.9420	0.4676	8.8750	20.6600	3.9870
BL8	445RCD	-0.0422	-0.1289	0.0023	-0.1818	0.5879	-0.0956
BL9	"	-0.0461	-0.1426	0.0023	-0.3232	0.0112	-0.1099
BL10	897RCD	0.1254	1.0340	0.6479	-0.0468	-0.2874	0.2675
BL11	"	0.2137	1.7680	1.0960	-0.0835	-0.5028	0.4709

Load Combinations:	Membrane Stresses (kips/ft^2)			Bending Moments (kip-ft)		
	SXX	SYY	SXY	MXX	MYY	MXY
LC1	-164.5	20.9	2.35	-146.0	-918.7	5.95
LC2	-164.5	20.9	2.36	-145.9	-918.7	6.01
LC3	-123.2	22.8	2.32	-103.6	-687.6	**8.63**
LC4	-164.1	23.3	3.42	-146.0	-914.7	6.15
LC5	-164.5	19.4	0.96	-145.9	-913.6	5.13
LC6	-163.9	**23.8**	3.41	-145.4	-917.0	6.51
LC7	-164.4	19.9	0.95	-145.2	-915.9	5.50
LC8	-164.3	22.8	**3.52**	**-146.7**	-918.6	6.20
LC9	**-164.8**	18.9	1.05	-146.5	-917.6	5.19
LC10	-164.1	23.3	3.51	-146.0	**-920.9**	6.57
LC11	-164.6	19.4	1.05	-145.8	-919.8	5.55
LC12	-126.2	18.2	2.78	-112.6	-705.3	4.83
LC13	-126.6	14.7	0.58	-112.4	-704.3	3.89
LC14	-126.1	18.5	2.77	-111.9	-705.3	5.05
LC15	-126.5	15.0	0.58	-111.8	-704.3	4.11

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Since the relationship of thermal growth to temperature change is linear, the results of the finite element analysis can be extrapolated linearly. To account for the increase in pool temperature, the element forces due to thermal loading will be increased in proportion to the temperature increase.

For walls: $\Delta T = (158° - 70°) / (150° - 70°) = 88° / 80° = 1.10$ . . . . . 10% increase

For slab: $\Delta T = (158° - 50°) / (150° - 50°) = 108° / 100° = 1.08$ . . . . . 8% increase

In Table 2, the element forces for basic load 1, liner expansion, and basic load 4, axial expansion, are increased 10%. Combined loads are then recomputed using the same formulations as in Table 1.

Table 2
Element # 7 - Recomputed Forces for T = 158°F

Basic Loads	Membrane Stresses (kips/ft^2)			Bending Moments (kip-ft)		
	SXX	SYY	SXY	MXX	MYY	MXY
BL1	0.8364	3.8225	0.0622	-1.8623	-12.0340	0.1090
BL2	0.1477	0.2398	0.0339	1.9640	8.8710	0.1118
BL3 (Not Used)						
BL4	-141.3500	10.7393	2.5091	-125.2900	-788.4800	5.2393
BL5	0.9508	2.3670	-0.5253	1.1080	10.6600	-0.3801
BL6	0.9734	2.3940	-0.5183	1.1710	10.6500	-0.3356
BL7	3.4480	6.9420	0.4676	8.8750	20.6600	3.9870
BL8	-0.0422	-0.1289	0.0023	-0.1818	0.5879	-0.0956
BL9	-0.0461	-0.1426	0.0023	-0.3232	0.0112	-0.1099
BL10	0.1254	1.0340	0.6479	-0.0468	-0.2874	0.2675
BL11	0.2137	1.7680	1.0960	-0.0835	-0.5028	0.4709

Load Combinations	Membrane Stresses (kips/ft^2)			Bending Moments (kip-ft)		
	SXX	SYY	SXY	MXX	MYY	MXY
LC1	-181.1	22.6	2.65	-161.0	-1013.3	6.58
LC2	-181.1	22.6	2.66	-160.9	-1013.3	6.64
LC3	-135.9	24.1	2.55	-115.1	-760.3	**9.11**
LC4	-180.7	25.1	3.72	-161.1	-1009.3	6.78
LC5	-181.1	21.1	1.26	-160.9	-1008.3	5.77
LC6	-180.5	**25.5**	3.71	-160.4	-1011.6	7.15
LC7	-181.0	21.6	1.25	-160.2	-1010.5	6.13
LC8	-180.9	24.5	**3.82**	**-161.7**	-1013.2	6.84
LC9	**-181.4**	20.6	1.36	-161.5	-1012.2	5.82
LC10	-180.7	25.0	3.81	-161.0	**-1015.5**	7.20
LC11	-181.2	21.1	1.35	-160.8	-1014.4	6.18
LC12	-138.9	19.6	3.01	-124.1	-778.1	5.32
LC13	-139.4	16.0	0.82	-124.0	-777.1	4.38
LC14	-138.9	19.8	3.01	-123.5	-778.1	5.54
LC15	-139.3	16.3	0.81	-123.3	-777.1	4.60

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The governing element #7 forces are as follows:

	M_ϕ	M_θ	N_ϕ	N_θ
LC6	-168 kip-ft	-1019 kip-ft	-1083 kips	153 kips
LC8	-169 kip-ft	-1020 kip-ft	-1085 kips	147 kips
LC9	-167 kip-ft	-1018 kip-ft	-1088 kips	124 kips
LC10	-168 kip-ft	-1023 kip-ft	-1084 kips	150 kips

Where:

M_ϕ = MXX + |MXY| Meridional moment
M_θ = MYY + |MXY| Hoop moment
N_ϕ = SXX * 6 ft. Meridional radial shear force
N_θ = SYY * 6 ft. Hoop radial shear force
ϕ = Meridional direction = SLSAP local x-direction = horizontal
θ = Hoop direction = SLSAP local y-direction = vertical
6 ft = Element thickness; used to convert the SLSAP element stresses to force on the TEMCO section.

From the previous analyses it is apparent that the hoop (θ) direction is critical, therefore LC6 (max. M_θ) and LC10 (max. N_θ) will be checked.

These will be combined with the additional loads due to the high density fuel racks (Reference 1, Section 8.1.12.10, page 69) as follows:

Addt'l Load	M_ϕ	M_θ	N_ϕ	N_θ
(D + V)	2 kip-ft	13 kip-ft	7 kips	16 kips
(N. shear)	-2 kip-ft	-3 kip-ft	4 kips	32 kips

This results in the following loads to be input to TEMCO:

Load Case	M_ϕ	M_θ	N_ϕ	N_θ
LC6	-168 kip-ft	-1009 kip-ft	-1072 kips	201 kips
LC10	-168 kip-ft	-1013 kip-ft	-1073 kips	198 kips

These loads will be input to TEMCO along with the factored temperature gradient based on 50°F outside temperature and 158°F inside temperature.

Factored gradient = 1.3 x (158° - 50°) = 1.3 x 108° = 140°F

The TEMCO analysis is provided in Attachment B.

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Final

The results of the TEMCO analysis are as follows:

Table 3
Element #7 - Revised Stress Summary

	Reinforcing Steel				Concrete	
	Horizontal		Vertical		Horiz.	Vert.
	Inside	Outside	Inside	Outside		
Section Stresses	4.4 ksi	21.0 ksi	3.0 ksi	50.9 ksi	1850 psi	626 psi
Allowable Stress (1)	54 ksi	54 ksi	54 ksi	54 ksi	2975 psi	2975 psi
Factor of Safety	12.27	2.57	18.00	1.06	2	5

(1) Allowable stresses*:

Reinforcing steel . . . . 0.9 f_y = 0.9 (60 ksi) = 54 ksi

Concrete . . . . 0.85 f'_c = 0.85 (3500 psi) = 2975 psi

(* The dynamic factor used in the previous summary is conservatively omitted.)

SUMMARY & CONCLUSIONS:

The revised maximum concrete stress for element #7 is 1850 psi which is 7% higher than the previous analysis. The revised factor of safety for the concrete stresses is 2. This is greater than 1.0 and therefore is acceptable.

The revised maximum reinforcing steel stress is 50.9 ksi which is 9% higher than the previous analysis. The revised factor of safety for rebar stress is 1.06. This is greater than 1.0 and therefore is acceptable.

The other elements would increase approximately the same percentage as for element #7. For concrete stress, the previous minimum factor of safety was 1.57 for element #21. With a change of 9% this would still be acceptable. By comparison, all elements are acceptable.

The spent fuel pool walls and slab are adequate for a maximum pool temperature of 158°F.

TEMCO

CRACKED SECTION ANALYSIS RESULTS

REF. RUN 1.D SRAN, 8.24.87. ALSO SEE NOTES BELOW TABLE

SECTION	LOCATION	MERIDIONAL STRESS				HOOP STRESS			
		CONCRETE		REBAR		CONCRETE		REBAR	
		IN/SIDE	OUT/SIDE	IN	OUT	IN	OUT	IN	OUT
1a	BASEMAT	-	-2.073	2.845	-23.266	-	-1.873	22.225	-19.402
1-b	"	-	-2.115	1.425	-24.079	-	-1.815	13.67	-19.183
2	WEST WALL	-.287	-1.727	-3.535	-19.216	-.537	-	-2.308	46.754
3	NORTH WALL	-	-1.351	0.372	-13.187	-	-0.826	18.2	-5.639
4	EAST WALL (SAP ELE. 98)	-.275	-1.759	-4.125	-19.549	-	-0.681	15.011	-4.652
5	EAST WALL (SAP ELE. 21)	-.538	-2.366	-7.491	-29.778	-.103	-1.133	-2.128	-11.239

NOTES: 1. MERIDIONAL CORRESPONDS TO X- HORIZ. DIRECTION; HOOP = Y- VERT.

2. FOR BASEMAT, MERID. = X = EAST-WEST; HOOP = Y = NORTH-SOUTH

3. TEMCO 'IN' (SIDE) = ACTUAL OUTSIDE FACE OF POOL & VICE VERSA

4. STRESSES ARE IN 'KSI' UNITS; +VE STRESS = TENSION, & VICE VERSA

5. ALLOWABLE STRESSES / REBAR: .9 Fy = 54 KSI, CONC. .85 f'c = 2975, say 3 KSI

Client CECO
Project Ev/BR 1 & 2
Proj. No. 4591-2/4683-4 Equip. No.

Prepared by [illegible] Date 8/25/87
Reviewed by [illegible] Date 8/27/87
Approved by [illegible] Date 11/11/87

SARGENT & LUNDY ENGINEERS

Calcs. For Temco Results

Safety-Related X | Non-Safety-Related

Calc. No. 8.1.12-BRW-46-754 | Rev. 3 | Date 1/12/10
Page 74 of

Revision 0

ATTACHMENT A
Page A1

Form GQ-3.08.1 Rev. 2 SL-F647 10-85 KPS

Revision 0

ATTACHMENT A
Page A2

SARGENT & LUNDY ENGINEERS

Client: CECO
Project: 3Y / BR 1 & 2
Proj. No. 4351-2 / 4553-4 Equip. No.

Calcs. For: TEMCO RESULTS
☒ Safety-Related ☐ Non-Safety-Related

Calc. No. 8112.10
Rev. 2 Date
Page 159 of

Prepared by: [illegible] Date 7/22/87
Reviewed by: [illegible] Date 7/22/88
Approved by: [illegible] Date 7/28/88

TEMCO RESULTS (WALL IMPACT SECTIONS)
(MECH + THERMAL LOADS)

REF RUN I.D. S711JH, 7/22/88.
REF. FOR RESULTS INTERPRETATION - PG. 74 OF THIS CALC SECT & ALSO SEE FOOTNOTE

SECTION* / SAP ELE	LOCATION	MERIDIONAL STRESS CONCRETE (psi) IN	MERIDIONAL STRESS CONCRETE (psi) OUT	MERIDIONAL STRESS REBAR (ksi) IN	MERIDIONAL STRESS REBAR (ksi) OUT	HOOP STRESSES CONCRETE (psi) IN	HOOP STRESSES CONCRETE (psi) OUT	HOOP STRESSES REBAR (ksi) IN	HOOP STRESSES REBAR (ksi) OUT	MIN* F.S. CONC/Rebar
1/20	EAST WALL BOT. CENTER	131	2005	-2.20	-15.06	20	835	-0.8	-6.03	1.85/N.A
2/98	BOT QUARTER	4	2136	-1.34	-15.96	-	489	14.87	-1.9	1.24/3.99
3/22	CENTER WALL/ IMPACT ELEV.	-	2091	.37	-15.49	-	1273	7.34	-8.37	1.78/8.09
4/154	SIDE WALL / IMPACT ELEV.	-	936	4.51	-6.58	-	582	25.96	-1.37	3.97/2.29
	NORTH WALL									
5/226	BOT CENTER	-	1882	3.77	-13.59	192	-	-0.56	16.14	1.98/3.68
6/271	CENTER WALL / IMPACT ELEV.	-	1823	10.82	-12.55	-	1140	33.49	-4.35	2.04/1.77
7/259	SIDE WALL / IMPACT ELEV.	340	1229	-2.96	-9.34	324	-	-2.0	8.34	3.03/7.42
	MIN. F.S.*	10.94	1.74	5.49	N/A	11.48	2.92	1.77	3.68	

NOTES:
1. MERID. CORRESPONDS TO X - HORIZ DIRECTION; HOOP ⇒ Y, VERT.
2. TEMCO 'IN' = ACTUAL OUTSIDE FACE OF CONCRETE WALL & VICE VERSA
* 3. MIN. FACTOR OF SAFETY = ALLOW. STRESS X DIF / ACTUAL STRESS, SEE NEXT PAGE.

Form GQ-3.08.1 Rev. 2 SL-F647 10-85 KPS

SARGENT & LUNDY ENGINEERS

Client: CECO
Project: BYRON 1&2
Prol. No.: 13512 / 6682-4 Equip. No.:

Calc. For: Stress & F.O.S. Summary

Safety-Related: X
Non-Safety-Related:

Calc. No.: C.1.12.10
Rev.: 2
Date:
Page 162 of

Prepared by: [illegible] Date: 7-25-88
Reviewed by: [illegible] Date: 7-26-88
Approved by: [illegible] Date: 7/28/88

CRITICAL LOCATION	MAXIMUM REBAR STRESS (KSI) HORIZONTAL BAR INSIDE	HORIZONTAL BAR OUTSIDE	HORIZONTAL BAR MIN. F.S.	VERTICAL BAR INSIDE	VERTICAL BAR OUTSIDE	VERTICAL BAR MIN. F.S.	MAXIMUM CONCRETE COMPRESSIVE STRESS (psi) HORIZ	VERTICAL	MIN. F.S.
1. WEST WALL BELOW FHB BASE SLAB	-19.22	-3.54	N/A[1]	46.75	-2.31	1.27	1727	537	2.15
2. NORTH WALL (202) (BOTTOM)	-13.19	0.37	160.5	-5.64	18.20	3.26	1351	826	2.75
3. NORTHWALL (226)	-13.59	3.77	15.76	16.14	-0.56	3.68	1882	192	1.90
4. NORTH WALL (221) (MID CENTER)	-12.55	10.82	5.49	-4.35	33.49	1.77	1823	1140	2.04
5. NORTH WALL (259) (MID SIDE)	-9.34	-2.96	N/A	8.34	-2.0	7.12	1229	324	3.03
6. EAST WALL (20) BOTTOM	-15.06	-2.20	N/A	-6.03	-0.8	N/A	2005	825	1.85
7. EAST WALL (98)*	-19.55	-4.12	N/A	15.04	-4.65	3.95	2136	681	1.74
8. EAST WALL (21) MID CENTER	-29.78	-7.49	N/A	-11.24	-2.13	N/A	2366	1133	1.57
9. EAST WALL (22)	-15.49	9.37	15.76	-8.37	7.34	8.09	2091	1273	1.78
10. EAST WALL (159) (MID SIDE)	-6.58	4.51	13.17	-1.37	25.96	2.29	936	583	3.97
11. BASE MAT (12) CENTER QUARTER	-23.26	2.85	20.84	-19.40	22.23	2.67	2073	1873	2.82
12. BASE MAT (13)	-24.08	1.43	41.54	-19.18	13.67	4.35	2115	1815	2.76

* ENVELOP OF WITHOUT WALL IMPACT (P.74) & WITH IMPACT

NOTE: 1 : F.S. IS COMPUTED FOR REBAR TENSILE STRESSES ONLY.

SARGENT & LUNDY ENGINEERS - TEMCO PROGRAM NO. 03.7.255-1.00

DATE : 11/26/96

```
         1         2         3         4         5         6         7         8
12345678901234567890123456789012345678901234567890123456789012345678901234567890

B/B Spent Fuel Pool - Adequacy of a pool temperature of 180 F
    1   3   2             2
    4   2   4      72.0
      2.75      3.12     67.00      3.12
      4.00      3.12      6.75      1.56     64.25      1.56     65.50      3.12
 1 LOAD 6                     L6
    -168.0    -1009.    -1072.     201.
 2 LOAD 10                    L10
    -168.0    -1013.    -1073.     198.
 3 LOAD T                     T        1

  50. 190.
  LA    1.        1.
  LB          1.   1.
     150.      3500.       66.

12345678901234567890123456789012345678901234567890123456789012345678901234567890
         1         2         3         4         5         6         7         8
```

REINFORCED CONCRETE SECTIONS UNDER ECCENTRIC LOADS AND THERMAL GRADIENT

ANALYSIS OF SECTION 4
SECTION THICKNESS (IN.)= 72.00

REINFORCEMENT DATA

LAYER	MERIDIONAL DISTANCE (IN)	MERIDIONAL AREA (SQ.IN)	HOOP DISTANCE (IN)	HOOP AREA (SQ.IN)
1	2.75	3.12	4.00	3.12
2	67.00	3.12	6.75	1.56
3			64.25	1.56
4			65.50	3.12

MATERIAL PROPERTIES
(ASSUMED NON-LINEAR)

UNIT WEIGHT OF CONCRETE(LB/CU.FT)	0.1500E+03
COEF. OF THERMAL EXP. OF CONCRETE AND REINF. STEEL(/D.F)	0.5560E-05
RATIO OF COMPRESSION TO TENSION MODULI FOR REINF. STEEL	0.1000E+01
MODULUS OF ELASTICITY OF REINF. STEEL(KSI)	0.2900E+05
YIELD STRENGTH OF REINF. STEEL(KSI)	0.6600E+02
COMPRESSIVE STRENGTH OF CONCRETE(PSI)	0.3500E+04
COMPRESSIVE STRAIN OF CONCRETE CORRESPONDING TO MAX. STRESS	0.2000E-02
ULTIMATE(CRUSHING) STRAIN OF CONCRETE	0.4000E-02

LOAD NO.,TYPE AND IDENT.	SYMBOL	MPHI (FT-KIPS)	NPHI (KIPS)	QRPHI (KIPS)	NPHITHETA (KIPS)	MPHITHETA (FT-KIPS)	TI (D.F)	TO (D.F)	TL1 (D.F)	EFF DIST1 (IN)	TL2 (D.F)	EFF DIST2 (IN)
1 LOAD 6	L6	-168.00	-1072.00	0.00	0.00	0.00	0.0	0.0	0.0	0.00	0.0	0.00
2 LOAD 10	L10	-168.00	-1073.00	0.00	0.00	0.00	0.0	0.0	0.0	0.00	0.0	0.00
3 LOAD T	T *	0.00	0.00	0.00	0.00	0.00	50.0	190.0	0.0	0.00	0.0	0.00

* INDICATES THERMAL LOAD CASES,READ USER MANUAL FOR MORE INFORMATIONS

LOAD NO.,TYPE AND IDENT.	SYMBOL	MTHETA (FT-KIPS)	NTHETA (KIPS)	QRTHETA (KIPS)	NTHETAPHI (KIPS)	MTHETAPHI (FT-KIPS)	TI (D.F)	TO (D.F)	TL1 (D.F)	EFF DIST1 (IN)	TL2 (D.F)	EFF DIST2 (IN)
1 LOAD 6	L6	-1009.00	201.00	0.00	0.00	0.00	0.0	0.0	0.0	0.00	0.0	0.00
2 LOAD 10	L10	-1013.00	198.00	0.00	0.00	0.00	0.0	0.0	0.0	0.00	0.0	0.00
3 LOAD T	T *	0.00	0.00	0.00	0.00	0.00	50.0	190.0	0.0	0.00	0.0	0.00

* INDICATES THERMAL LOAD CASES,READ USER MANUAL FOR MORE INFORMATIONS

LOADING COMBINATIONS

LA= 1.00L6 + 1.00T
LB= 1.00L10 + 1.00T

LD COMB	MPHI (FT-KIPS)	NPHI (KIPS)	QRPHI (KIPS)	NPHITHETA (KIPS)	MPHITHETA (FT-KIPS)	TI (D.F)	TO (D.F)	TL1 (D.F)	EFF DIST1 (IN)	TL2 (D.F)	EFF DIST2 (IN)	MPHIT (FT-KIPS)	NPHIT (KIPS)	QRPHIT (KIPS)	NPHITHETAT (KIPS)
LA	-168.00	-1072.00	0.00	0.00	0.00	50.0	190.0	0.0	0.00	0.0	0.00	0.00	0.00	0.00	0.00
LB	-168.00	-1073.00	0.00	0.00	0.00	50.0	190.0	0.0	0.00	0.0	0.00	0.00	0.00	0.00	0.00

LD COMB	MTHETA (FT-KIPS)	NTHETA (KIPS)	QRTHETA (KIPS)	NTHETAPHI (KIPS)	MTHETAPHI (FT-KIPS)	TI (D.F)	TO (D.F)	TL1 (D.F)	EFF DIST1 (IN)	TL2 (D.F)	EFF DIST2 (IN)	MTHETAT (FT-KIPS)	NTHETAT (KIPS)	QRTHETAT (KIPS)	NTHETAPHIT (KIPS)
LA	-1009.00	201.00	0.00	0.00	0.00	50.0	190.0	0.0	0.00	0.0	0.00	0.00	0.00	0.00	0.00
LB	-1013.00	198.00	0.00	0.00	0.00	50.0	190.0	0.0	0.00	0.0	0.00	0.00	0.00	0.00	0.00

SARGENT & LUNDY ENGINEERS - TEMCO PROGRAM NO. 03.7.255-1.00
B/B Spent Fuel Pool - Adequacy of a pool temperature of 180 F

DATE : 11/26/96

REINFORCED CONCRETE SECTIONS UNDER ECCENTRIC LOADS AND THERMAL GRADIENT
CRACKED SECTION ANALYSIS
TENSILE STRESSES/STRAINS ARE POSITIVE

****
MERIDIONAL SECTION
********** ϕ, X, horizontal

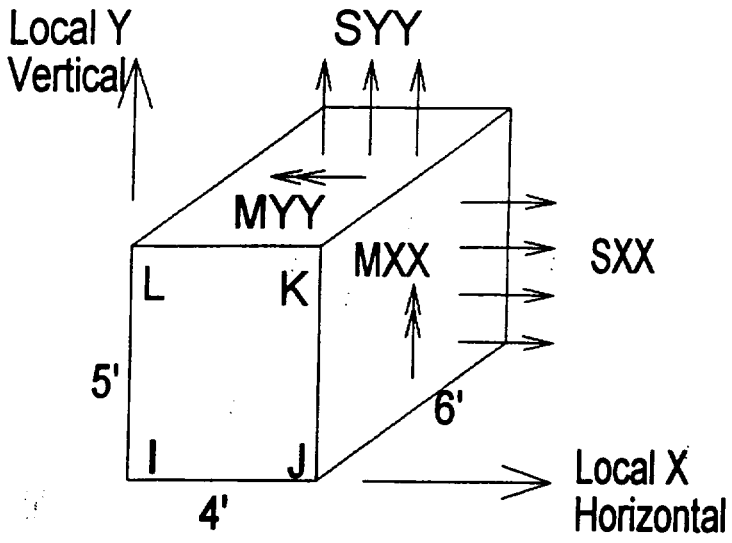
FOLLOWING ARE FINAL EFFECTS ON THE SECTION DUE TO MECHANICAL AND THERMAL LOADINGS

LOAD COMB	NA DISTANCE (IN)	STRAINS INSIDE (IN/IN)	STRAINS OUTSIDE (IN/IN)	STRESSES INSIDE (KSI)	STRESSES OUTSIDE (KSI)	REINF STRESSES INSIDE (KSI)	REINF STRESSES OUTSIDE (KSI)	THERMAL MOMENT (FT-KIPS)	THERMAL FORCE (KIPS)	INTERNAL MOMENT (FT-KIPS)	INTERNAL FORCE (KIPS)	EQUIV. THERMAL GRADNT (D.F)	EQUIV. THERMAL AXIAL (D.F)
LA	-14.02	-0.00013	-0.00077	-0.362	-1.850	-4.352	-21.028	932.16	0.00	764.16	-1072.00	140.0	0.0
LB	-14.07	-0.00013	-0.00077	-0.363	-1.850	-4.366	-21.042	931.93	0.00	763.93	-1073.00	140.0	0.0

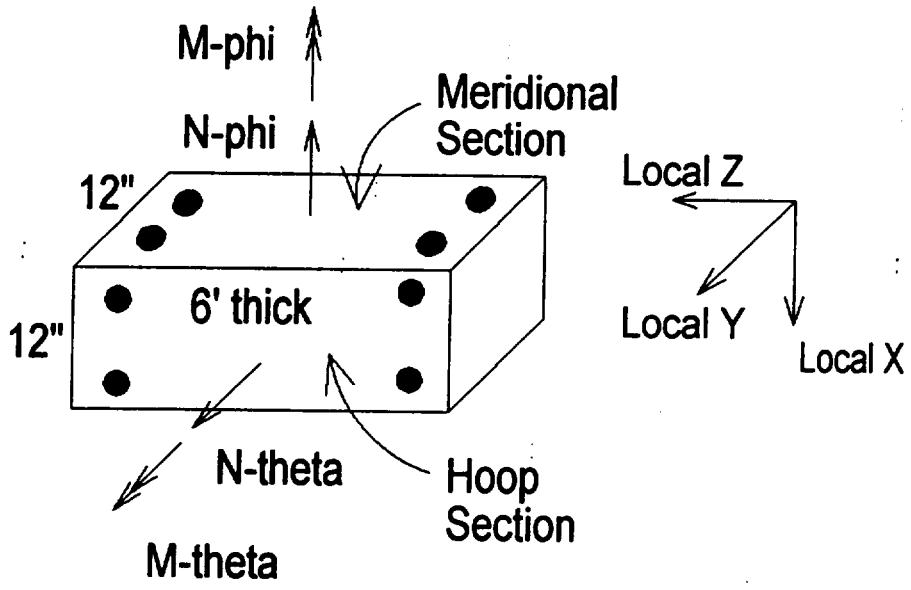
****
HOOP SECTION
********** θ, Y, vertical

FOLLOWING ARE FINAL EFFECTS ON THE SECTION DUE TO MECHANICAL AND THERMAL LOADINGS

LOAD COMB	NA DISTANCE (IN)	STRAINS INSIDE (IN/IN)	STRAINS OUTSIDE (IN/IN)	STRESSES INSIDE (KSI)	STRESSES OUTSIDE (KSI)	REINF STRESSES INSIDE (KSI)	REINF STRESSES OUTSIDE (KSI)	THERMAL MOMENT (FT-KIPS)	THERMAL FORCE (KIPS)	INTERNAL MOMENT (FT-KIPS)	INTERNAL FORCE (KIPS)	EQUIV. THERMAL GRADNT (D.F)	EQUIV. THERMAL AXIAL (D.F)
LA	7.21	-0.00022	0.00195	-0.611	0.000	-2.806	50.877	338.12	0.00	-670.88	201.00	140.0	0.0
LB	7.41	-0.00022	0.00194	-0.626	0.000	-2.974	50.695	338.48	0.00	-674.52	198.00	140.0	0.0



SLSAP Axes Definition
Ref. 1, Sect. 18.1.12.2, page 6



TEMCO Axes Definition
Ref. 1, Sect. 18.1.12.5, page 1

ComEd

NEP-12-02.08
Effective Date:
8/30/99

EDITORIAL/MINOR/TEMPORARY CALCULATION CHANGE TITLE PAGE

DCR #: 990668

AFFECTED CALC. NO. 8.1.12-BRW-96-754 Rev. 0 PAGE NO. 1

CHECK ONE:
- ☐ EDITORIAL
- ☐ EVALUATION OF TEMPORARY CONDITION
- ☒ MINOR/ADMINISTRATIVE CHANGES TO AN EXISTING CALCULATION (SAFETY MARGIN NOT AFFECTED)

☒ SAFETY RELATED ☐ AUGMENTED QUALITY ☐ NON-SAFETY RELATED

CALCULATION TITLE: Evaluation of Spent Fuel Pool for Elevated Temperature

☒ EDITORIAL CHANGE CONTINUED - See Attached

STATION/UNIT: BRW Unit 1 & 2 SYSTEM CODE: FC

EQUIPMENT NO. AND COMPONENT TYPE (IF APPL.):

Approvals required when initiating Calc Revision in accordance with Section C.5.4

PREPARED BY: WILLIAM J. PANANOS / WJ Pananos DATE: June 12, 2000
PRINTED NAME/SIGNATURE

REVISION SUMMARY: The pool temperature increases from 155 degrees F to 162.7 degrees F, due to core uprate, for a full core discharge. The structural design basis temperature increases from 158 to 162.7 degrees F.

ELECTRONIC CALCULATION DATA FILES: None
(Software name, Version, Name ext/size/date/hour/: min)
For Example: PIPSYS VERSION 2.3, SX01.dat, 63.5kb, 5/30/97, 14:23

DO ANY ASSUMPTIONS IN THIS CALCULATION REQUIRE LATER VERIFICATION?
☐ YES ☒ NO IF YES, INDICATE TRACKING NUMBER

REVIEWED BY: BRIAN A. LAWTON / BA Lawton DATE: 6/13/00
PRINTED NAME/SIGNATURE

REVIEW METHOD: COMMENTS (C, NC OR CI):

APPROVED BY: ME Hanley / ME Hanley DATE: 6/14/00
PRINTED NAME/SIGNATURE

PEPP-E FORM

DCR #:990668

AFFECTED CALC. NO.	8.1.12-BRW-96-754	Rev. 0	PAGE NO. 2

Purpose/Objective:
The purpose of this DCR is to document the effects of increases in the spent fuel pool temperature loading due to core uprate on the Spent Fuel Pool structure.

References:
1. Calculation No. ~~BRY~~ BYR 2000-007 / BRW-00-0010-M. DH 10-17-00
2.
3.
4.
5.

Calculation:
Revise page 3 of calculation 8.1.12-BRW-96-754 as shown on page 3 of this DCR.
Revise page 4 of calculation 8.1.12-BRW-96-754 as shown on page 3 of this DCR.
Revise page 5 of calculation 8.1.12-BRW-96-754 as shown on page 3 of this DCR.
Revise page 7 of calculation 8.1.12-BRW-96-754 as shown on page 3 of this DCR.
Revise page 9 of calculation 8.1.12-BRW-96-754 as shown on page 3 of this DCR.
Revise page 10 of calculation 8.1.12-BRW-96-754 as shown on page 4 of this DCR.
Revise page 11 of calculation 8.1.12-BRW-96-754 as shown on page 4 of this DCR.

Summary and conclusions:

Include these changes in the next revision of the calculation.

Justification:
This calculation was prepared to evaluate temperature effects due to the installation of high density fuel racks. The calculation evaluates the most highly stressed element comprising the Spent Fuel Pool structural model and uses the linear extrapolation of the temperature increase as a basis to determine the stresses. This approach is also applicable to the increased temperature resulting from core uprate.

SCOPE OF CALCULATION CHANGE

1. Purpose/Objective Calculation Page 3
   Add to first sentence "...pool temperature conditions due to core power uprate".

   Change temperatures as follows:

   1. 138°F to 141.2°F
   2. 155°F to 162.7°F
   3. 158°F to 173.7°F

2. Methodology & Acceptance Criteria Calculation Page 3
   Make the following changes to the last paragraph:
   (138°F to 141.2°F),
   (155°F to 162.7°F),
   (158°F to 173.7°F),
   delete "but may conservatively be treated as T_o"

3. Design Input Calculation Page 4
   In the section " Load Combinations" make the following change:
   Abnormal (temperature or cask drop). After loading combination add Normal temperature is applicable for the cask drop loading combination.

4. References Calculation Page 5
   Add reference for revised spent fuel pool temperatures.
   Calculation No. BRY2000-007 / BRW-00-0010-M.

5. Calculations/Analysis for Revised Temperatures Calculation Page 7
   Revise first paragraph as follows:
   158°F to 162.7°F

   Add the following at the end of the first paragraph:
   The abnormal temperature is multiplied by a load factor equal to 1.0 whereas the normal temperature is multiplied by a load factor equal to 1.3. The loading combinations incorporating the normal temperature with the load factor of 1.3 will result in greater thermal loads and stresses than the loading combinations for the abnormal temperature condition. This analysis addresses the loading combinations which incorporate the factored normal temperature.

6. Calculations/Analysis for Revised Temperatures Calculation Page 9
   Substitute 162.7°F for 158°F and recomputed Table 2 values for Element #7. For walls the thermal loading will increase 15.9%. For the slab the thermal loading will increase by 12.7%.

7. Calculations/Analysis for Revised Temperatures Calculation Page 10
Revise the values of the forces to reflect the increase in the thermal loading effects. For the factored gradient equation replace 158°F with 162.7°F.

8. Calculations/Analysis for Revised Temperatures Table 3 Calculation Page 11
Re-compute Section Stresses in Table 3. Limiting reinforcing steel stress of 50.9 ksi will become 53.7 ksi.

9. Summary and Conclusions Calculation Page 11
Replace 50.9 with 53.7 ksi.
Replace 9% with 14.9%

Revise the second paragraph to read:
"The revised maximum steel reinforcing stress is 53.7 ksi which is 14.9% higher than the previous analysis. The design margin for the allowable stress is greater than 1.0 and this revised loading condition is acceptable."

Revise the last paragraph to read:
"The spent fuel pool walls and slab are adequate for a maximum normal pool temperature of 162.7°F."