August 21, 2020

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the matter of	
Pacific Gas and Electric Company	Docket Nos. 50-275-LAR
Diablo Canyon Nuclear Power Plant	50-323-LAR
Units 1 and 2	

DECLARATION OF DAVID A. LOCHBAUM
REGARDING PROPOSED NO SIGNIFICANT HAZARDS DETERMINATION
FOR PACIFIC GAS AND ELECTRIC COMPANY'S
EXIGENT LICENSE AMENDMENT REQUEST

Under penalty of perjury, I, David A. Lochbaum, declare:

1. My name is David A. Lochbaum. I reside in the state of Tennessee. I am a nuclear engineer by training, experience, and education.
2. I retired in October 2018 from the Union of Concerned Scientists (UCS) after working nearly four decades on nuclear power issues. My experience includes assignments at/for operating nuclear plants (Hatch, Browns Ferry, Grand Gulf, Hope Creek, Susquehanna, FitzPatrick, Wolf Creek, Salem, Peach Bottom, and Connecticut Yankee), working for the U.S. Nuclear Regulatory as a reactor technology instructor, and working for (UCS) on nuclear power safety issues. I have a Bachelor of Science degree in nuclear engineering from the University of Tennessee. My professional qualifications are detailed in my attached Curriculum Vitae (Exhibit A).
3. I have been retained by San Luis Obispo Mothers for Peace to evaluate an exigent license amendment request (LAR) by Pacific Gas & Electric Co. (PG&E) to the Nuclear Regulatory Commission (NRC) on August 12, 2020 (ML20225A303). If approved, the LAR would allow PG&E to remove portions of the auxiliary feedwater system (AFW) on Diablo Canyon Units 1 for longer periods of time than allowed under the current operating license should planned inspections of the AFW pipes indicate, as expected by PG&E, that walls have thinned to unacceptable thicknesses. If so, the thinned pipe sections would be replaced as they were on Unit 2 to restore the necessary safety levels.
4. I have examined PG&E's exigent LAR in detail.
5. I have also reviewed the following related documents:

a. Diablo Canyon Updated Final Safety Analysis Report Rev. 23, May 3, 2017 (ML17157B366);
b. §50.59 Changes, tests and experiments of Title 10 of the Code of Federal Regulations;
c. Bley, Dennis C, Wheeler, David M., Cate, Carroll L., Stillwell, Daniel W., and Garrick, B. John, "Reliability Analysis of Diablo Canyon Auxiliary Feedwater System," September 1980. (ML17095A390);
d. NRC Inspection Manual, Inspection Procedure 49001, "Inspection of Erosion-Corrosion/Flow-Accelerated-Corrosion Monitoring Programs," December 11, 1998;
e. Pacific Gas and Electric Company, "Diablo Canyon Power Plant Units 1 and 2 Technical Specification Bases," Revision 10, December 2016 (ML16356A266);
f. Pacific Gas and Electric Company, Diablo Canyon Unit 1 Technical Specifications, January 2008;
g. Email dated August 18, 2020, to me from Scott Morris, NRC Regional Administrator, Region IV;
h. Pacific Gas and Electric Company, "Diablo Canyon Unit 1 Licensee Event Report 1-92-022-00, Indications on the Main Feedwater Piping Near the Steam Generator Nozzles due to Thermal Fatigue," October 30, 1992 (ML16341G734);
i. NRC Information Notice 92-07, "Rapid Flow-Inducted Erosion/Corrosion of Feedwater Piping," January 9, 1992 (ML082380388);
j. NRC Information Notice No. 91-18, "High-Energy Piping Failures Caused by Wall Thinning," March 12, 1991 (ML031190529);
k. Pacific Gas and Electric Company, "Response to Generic Letter 89-08, Erosion/Corrosion," July, 19, 1989 (ML16342C228);
l. NRC Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning," May 2, 1989 (ML031200731);
m. Pacific Gas and Electric Company, "Response to NRC Bulletin No. 87-01, Thinning of Pipe Walls," September 8, 1987 (ML17083B938);
n. NRC Bulletin No. 87-01, "Thinning of Pipe Walls in Nuclear Power Plants." July 9, 1987 (ML031210862);

o. Pacific Gas and Electric Company, "Diablo Canyon Power Plant Units 1 and 2 Individual Plant Examination Report," April 1992 (https://adamswebsearch2.nrc.gov/webSearch2/main.jsp?AccessionNumber=9204240016);
p. Nuclear Regulatory Commission, "Final Significance Determination of a Red Finding, Notice of Violation, and Assessment Follow-up Letter (NRC Inspection Report No. 05000259/2011008) Browns Ferry Nuclear Plant," May 9, 2011 (ML111290482);
q. Pacific Gas and Electric Company, "Response to NRC Request for Additional Information Regarding "License Amendment Request 20-01, Exigent Request for Revision to Technical Specification 3.7.5, "Auxiliary Feedwater System"," August 16, 2020; and
r. Pacific Gas and Electric Company, "Response to NRC Request for Additional Information Regarding "License Amendment Request 20-01, Exigent Request for Revision to Technical Specification 3.7.5, "Auxiliary Feedwater System"," August 18, 2020.

6. Having examined these documents, it is my professional opinion that PG&E's exigent LAR should not be approved by the NRC because it would likely expose the community around Diablo Canyon Unit 1 to an unduly elevated accident risk. Therefore, I strongly disagree with PG&E's and the Staff's determinations that the proposed license amendment poses "no significant hazards."
7. My expert opinion is based on the following reasons:
   a. The "how safe is safe enough" question for the Unit 1 AFW system is defined by the technical specifications, an integral part of the reactor operating license issued by the NRC.
   b. The current technical specifications permit Unit 1 to continue operating for up to 72 hours when one of the three AFW pumps is inoperable (except in the case of a steam supply problem for the one turbine-driven AFW pump). If the full complement of AFW pumps cannot be restored within 72 hours, the reactor must be shut down within 6 hours. If two AFW pumps are unavailable, the technical specifications require the reactor to be shut down within 6 hours. If all three AFW

pumps are unavailable, the technical specifications require the reactor to be shut down immediately.

c. The time-frames in the technical specifications were not pulled from a hat; they depend on the safety function performed by systems and components and the likelihood that plant conditions would require the systems and components to function to prevent or mitigate the consequences.

d. The reason for the relatively short time-frames governing AFW components in the technical specifications is evident from risk analyses performed by PG&E for Diablo Canyon. Many of these risk analyses are hidden from public view by the NRC as purportedly security-related information, but PG&E's Individual Plant Examination (IPE) from April 1992 is reasonably believed to still accurately portray the relative risk of the AFW system and the safety rationale for the associated timeliness requirements for AFW components in the technical specifications.

e. Figure 1 on page 12 below is Table 3.4.2-2 from PG&E's IPE. It lists 29 initiating events having the highest calculated risk of reactor core damage. A handful (*e.g.*, Medium Loss of Coolant Accident, Large Loss of Coolant Accident, Excessive Loss of Coolant Accident, Core Power Excursion, and Interfacing System Loss of Coolant Accident) are not mitigated by the AFW system. However, the AFW system has a safety function to perform in mitigating the rest of the 29 initiating events.

f. As shown in Figure 1, Loss of Offsite Power, the initiating event with the highest risk in PG&E's IPE, contributes nearly half (41%) of the risk of reactor core damage at Diablo Canyon. The AFW system has a vital safety function to perform to mitigate this initiating event. The loss of offsite power inherently results in the loss of the main feedwater system. The AFW system is designed to automatically start in event the main feedwater system is unavailable. The turbine-drive AFW pump and/or the two motor-driven AFW pumps (which can be powered from the onsite emergency diesel generators when offsite power is lost) can continue to remove decay heat produced by the reactor core.

g. Figure 2 on page 13 below is Table 3.4.2-4 from PG&E's IPE. It ranks safety systems at Diablo Canyon by their importance in preventing reactor core damage. The AFW system placed 6th on this list, ahead of Residual Heat Removal (RHR) Trains A and B, Emergency Diesel General 2-2, 125 volt bus G and many other systems and components. And note that these results assumed the out-of-service times for AFW components from the current technical specifications, not the significantly relaxed times sought by PG&E in its exigent LAR. With the longer times, the AFW system could only move higher on the risk list, not lower.

h. Figure 3 on page 14 below is Table 3.4.2-6 from PG&E's IPE. It ranks safety systems at Diablo Canyon by two related risk measures: Risk Achievement Worth (RAW) and Risk Reduction Worth (RRW). The RAW values are determined by two computer runs, one assuming the system or component reliability based on operating experience and the second assuming the system or component has zero reliability (i.e., 100% chance it fails when needed). The RRW valves are also determined by two computer runs, but this time the second run assumes that the system or component has perfect reliability (i.e., 0% chance of failing when needed). The AFW system's systems motor-driven and turbine-driven pumps occupy two of the top ten risk-rankings, 4th and 8th.

i. PG&E's IPE considered common-cause failures that would prevent the AFW system from fulfilling its necessary safety function. But those common-cause failures were limited to failures of active components (e.g., check valves, dump valves, etc.).

j. The only common-cause failure of passive components (e.g., pipes, tanks, heat exchangers, etc.) in PG&E's IPE affecting the AFW system was the rupture of a main feedwater or AFW pipe that flooded the AFW pump rooms and disabled both of the motor-driven AFW pumps. This potential flooding scenario was one of the three postulated internal flooding events having risk significance. PG&E's no significant hazards analysis for the exigent LAR stated: "The AFW System is not an initiator of any design basis accident or event" – a statement apparently contrary to their own IPE's analysis. See Initiating Event 21 on Figure 1 which is Table 3.4.2-2. PG&E's description of this initiating event explained that rupture of either a main feedwater or an AFW pump could flood the AFW pump rooms and disable BOTH of the motor-driven AFW pumps. Yet PG&E's no significant hazards analysis fails to explain how the technical specification changes it seeks will not adversely affect this initiating event.

k. Similarly, a third-party evaluation reported in September 1980 of the reliability of the AFW system at Diablo Canyon considered common-cause failures as factors in lessening the reliability of the system. As in PG&E's IPE, this evaluation was limited to failures of active components to common-causes (e.g., inadequate maintenance, design errors, installation miscues, etc.). Thus, the increased likelihood of AFW system piping ruptures until its pipes thinned to unacceptable thicknesses are not modeled in the risk analyses. In other words, risk analyses that exclude consideration of pipe ruptures due to common-causes (i.e., thinning) cannot be used to justify continued reactor operation. The risk tool does not apply to the question being asked and therefore cannot provide a righteous answer.

l. Appendix 9.5A to the Updated Final Safety Analysis Report (UFSAR) for Diablo Canyon describes how the AFW system provides removal of reactor core decay heat in event of postulated fires in various Fire Areas throughout the plant. The fire hazards analyses that are summarized in the UFSAR were performed to fulfill Appendix R to 10 CFR 50 adopted in the early 1980s following the Browns Ferry fire. Unlike the response to other design bases events, the response to a postulated fire need not assume the worst-case single failure. Thus, while the AFW system has redundancy in terms of three pumps and four flow pathways to steam generators for decay heat removal, UFSAR Appendix 9.5A describes cases where the fire takes away all but a single AFW flow pathway. If NRC approves PG&E's exigent LAR, that sole safety net could be removed for 7 days at a time as PG&E fixes up to four unsafe AFW flow pathways. PG&E's no significant hazards analysis for the exigent LAR does not mention the potential impact on the fire hazards and safe shutdown analyses, which rely considerably if not entirely on AFW.

m. PG&E seeks the NRC's approval to revise the answer to the "how safe is safe enough" question for the Unit 1 AFW system to allow portions of the system to unavailable for longer periods than currently permitted by the technical specifications, extending the current 72-hour time limit for one AFW pump to be unavailable to 7 days.

n. PG&E stated in their exigent LAR that they will inspect the Unit 1 AFW system piping and expect to find pipe walls thinned to less than allowed by the ASME code. If so, they propose to replace the unacceptably thinned pipe sections to restore the required safety levels. PG&E stated that, based on their experience replacing unacceptably thinned pipe sections on Unit 2, the safety restoration could take up to 7 days.

o. PG&E stated in their exigent LAR that they discovered a 3.9 gallon per minute leak from an AFW pipe on Unit 2 last month and discovered six other AFW pipe sections thinned to less than thicknesses allowed under the ASME code.

p. Following the rupture of a corroded or otherwise thinned pipe in December 1986 at the Surry nuclear power plant that killed four of the eight workers in the vicinity, the NRC required all nuclear plant owners to develop and implement monitoring programs to detect pipe wall thinning and replace sections before they thinned to unacceptable thicknesses. Not every inch of every pipe is monitored. Based on factors such as fluid flow rate, fluid temperature, fluid pressure, and pipe configuration (e.g., straight run versus pipe bend), vulnerable sections are monitored

q. The seven AFW pipe sections replaced on Unit 2 (i.e., the leaking section and the six other thinned locations) may or may not have been monitored under PG&E's pipe monitoring program mandated by the NRC. If so, PG&E's failure to adequately implement a monitoring program mandated many years ago by the NRC is insufficient justification for them to now be given longer time to remedy their self-inflicted cause.

r. NRC Region IV Administrator Scott Morris emailed me that the leakage on the Unit 2 AFW pipe was caused by external corrosion. By letter dated August 16, 2020, in response to NRC's Request for Additional Information, PG&E identified the cause of the pipe degradation as being external corrosion from the highly corrosive coastal marine environment. PG&E stated that Unit 2 was subjected to higher corrosion due to localized weather patterns and noted that "Unit 2 has historically experienced more forced outages and consequently operated its 10 percent atmospheric steam dumps more frequently. The steam dump exhaust, being located above the AFW piping, results in a wet environment due to falling condensation. The other two trains (supplying SGs 3 and 4 for each unit) are located indoors."

s. PG&E's implication that the Unit 1 AFW piping will likely have less degradation due to milder marine coastal environmental conditions seems contrary to its exigent LAR. If PG&E's implication was accurate, whenever they get around to conducing inspections on Unit 1 would confirm that notion. In that case, neither the pipe replacements nor the longer out-of-service times requested via the exigent LAR would be necessary. If, on the other hand, the Unit 1 AFW piping has degraded as much or more than that on Unit 2, the reactor's operation with multiple AFW trains impaired is not justified. The Unit 1 piping should be replaced with the unit offline as was properly done on Unit 2.

t. By letter dated August 18, 2020, in response to NRC's Request for Additional Information, PG&E explained its position that Unit 2 was subjected to a harsher marine coastal environment than Unit 1. PG&E stated that its "review of the Corrective Action Program identified on the order of twice as many condition reports documenting corrosion and coating on the Unit 2 pipe rack versus Unit 1." In other words, PG&E had ample warnings that exposed piping on both units was degrading due to exposure to the corrosion marine coastal environment but took zero steps to prevent that identified degradation mechanism from compromising necessary safety margins until workers discovered the 3.9 gallon per minute leakage on Unit 2 in July 2020. As noted below where NRC sanctioned another plant owner for documenting but not resolving signs of problems, the NRC should neither tolerate nor facilitate such abysmal licensee performance.

u. PG&E identified seven sections of Unit 2 AFW piping with thicknesses less than allowed by the ASME code. The identifications led PG&E to replace the thinned sections before restarting Unit 2.

v. PG&E strongly suspects that sections of the Unit 1 AFW piping will also be thinner than allowed by the ASME safety code, requiring replacement to restore the necessary safety levels. PG&E seeks the NRC's permission to fix this safety problem on Unit 1 while Unit 1 continues to operate — something they recently opted NOT to do when the problem was found on Unit 2.

w. Whether caused internally (i.e., thinning due to erosion/corrosion) or externally (i.e., exposure to corrosive agents), PG&E's current aging monitoring program for AFW system piping is demonstrably inadequate. If thinned internally to less than thicknesses allowed by the ASME code, the NRC-mandated monitoring program failed to detect and correct this slowly developing condition until thicknesses dropped below acceptable levels. If thinned due to external corrosion, a failure mode not anticipated by and therefore not adequately managed by the AFW system aging management program is involved.

x. In its exigent LAR and in other publicly available records, PG&E has not explained how the degradation resulting in AFW system being thinned to unsafe thicknesses will be prevented in the future by either a revision to its pipe wall thickness monitoring program and/or the development of a new program to monitor for external corrosion degradation. Absent such discussion, the efficacy of merely replacing the pipes cannot be judged. The erosion/corrosion monitoring programs mandated by the NRC protect against internal degradation, but PG&E contends that the current degradation mechanism is external corrosion from the marine coastal environment. Just as PG&E has a formal monitoring program for internal pipe degradation, a comprehensive corrective action for external pipe degradation necessitates a comparable monitoring program. PG&E has failed to describe such a program as part of its corrective actions for this safety impairment.

y. The situation on Diablo Canyon Unit 1 mirrors the situation NRC uncovered at the Tennessee Valley Authority's Browns Ferry Nuclear Plant (see May 9, 2011, NRC Red Finding letter). The disc of a valve in the Residual Heat Removal (RHR) system separated from its stem. The reactor operated for several years with this degraded condition. TVA periodically tested the valve by stroking it opened and closed. But whilst the stem moved here and there, the disc did not. TVA contended that the valve's failure was unforeseen and the NRC could not blame them for not having found and fixed it sooner. The NRC disagreed. TVA contended that had an emergency happened, workers would have quickly noticed that the valve was not positioned properly and found means to force the valve to its proper position. The NRC disagreed. TVA contended that even if the valve remained in the wrong position, there were redundant trains available to perform the necessary safety function. The NRC disagreed, pointing out that TVA took credit for this valve and associated RHR train in mitigating fires in certain Fire Areas. While there were indeed redundant trains for non-fire events, this valve and its train were the only safety net protecting against fires in certain Fire Areas. This reality led NRC to push the overall risk of this deficiency into the Red zone – a space occupied by a handful of reactors over the 20 years of the NRC's Reactor Oversight Process's color-coding.

z. Appendix 9.5A to the UFSAR for Diablo Canyon describes how the AFW system performs roles during postulated fires in many Fire Areas, without backup. PG&E's exigent LAR was silent with regard to how the fire hazard would be properly managed during the proposed extended AFW system impairments.

aa. The current technical specifications for AFW system component unavailability were developed based on the safety function to be performed during design bases events and the likelihood that such events occur. PG&E repaired multiple sections of AFW piping on Unit 2 and anticipates needing to do so on Unit 1, hence the submittal of the exigent LAR seeking longer time to implement the overdue replacements.

bb. If the Unit 1 AFW system currently has pipe sections thinned to unacceptable thicknesses in multiple pathways, they could be in condition that warrants immediate shutdown of the reactor for safety reasons — Technical Specification LCO 3.7.5 Condition C. NRC must not allow PG&E to pretend that only one AFW train at a time is impaired when it has such ample grounds to suspect a larger problem.

cc. Had PG&E shut down Unit 1 on August 12, 2020, instead of asking NRC's approval for an online repair effort, workers could have inspected and repaired AFW trains in parallel rather than in series as proposed. IF PG&E is correct in estimating that repairs take up to seven days, they'd have fixed all AFW system piping by now and could safety restart the Unit.

I declare under penalty of perjury that the foregoing facts are true and correct to the best of my knowledge, and the foregoing opinions are based on my best professional judgement.

Executed August 21, 2020

David A. Lochbaum

Figure 1

Table 3.4.2-2. Initiating Event Contribution to Core Damage Frequency			
	Initiating Event	Point Estimate Core Damage Frequency (Per Year)	Percent Contribution
1.	Loss of Offsite Power	3.6e-05	41
2.	Reactor Trip	6.9e-06	7.9
3.	Turbine Trip	5.3e-06	6.0
4.	~~Medium LOCA~~	4.7e-06	5.3
5.	Partial Main Feedwater Loss	4.0e-06	4.5
6.	Total Loss of Auxiliary Saltwater	3.4e-06	3.9
7.	Loss of 125V DC Bus F	3.2e-06	3.6
8.	Flood - Loss of Component Cooling Water	2.6e-06	3.0
9.	Loss of 480V Switchgear Ventilation	2.5e-06	2.8
10.	Loss of 125V DC Bus G	2.4e-06	2.8
11.	~~Large LOCA~~	2.4e-06	2.7
12.	Total Loss of Component Cooling Water	2.0e-06	2.3
13.	Excessive Feedwater	1.8e-06	2.1
14.	Loss of 125V DC Bus H	1.8e-06	2.0
15.	Steam Generator Tube Rupture	1.8e-06	2.0
16.	Loss of Primary Flow	1.0e-06	1.2
17.	Loss of Condenser Vacuum	1.0e-06	1.2
18.	Steam Line Break Inside Containment	8.0e-07	< 1
19.	Inadvertent Safety Injection	7.4e-07	< 1
20.	Small LOCA, Nonisolable	6.3e-07	< 1
21.	Flood - Loss of Both Motor-Driven AFW Pumps	6.1e-07	< 1
22.	Total Main Feedwater Loss	5.4e-07	< 1
23.	Closure of One Main Steam Isolation Valve (MSIV)	5.3e-07	< 1
24.	Loss of Control Room Ventilation	4.0e-07	< 1
25.	Small LOCA, Isolable	2.7e-07	< 1
26.	~~Excessive LOCA~~	2.6e-07	< 1
27.	~~Core Power Excursion~~	1.3e-07	< 1
28.	Closure of All MSIVs	1.0e-08	< 1
29.	~~Interfacing System LOCA at RHR Pump Discharge~~	6.3e-08	< 1

Figure 2

Table 3.4.2-4. Top Event Core Damage Frequency Importance Ranking - Nonguaranteed Failures

Ranking	Top Event ID	Percentage of Core Damage Frequency	Event Description
1	GG	28	Emergency Diesel Generator 1-2 (Bus G)
2	PR	24	RCS Pressure Relief and PORV Reclosure
3	CC	21	Component Cooling Water System
4	SE	17	RCP Seal Integrity
5	GF	16	Emergency Diesel Generator 1-3 (Bus F)
6	AW	16	Auxiliary Feedwater System
7	SW	13	Emergency Diesel Generator 1-3 Aligned to Unit 2
8	RE	13	Sequence Specific Recovery Actions
9	GH	12	Emergency Diesel Generator 1-1 (Bus H)
10	DG	9.3	125V DC Bus G
11	DH	8.5	125V DC Bus H
12	AS	7.2	Auxiliary Saltwater System
13	TG	6.9	Emergency Diesel Generator 2-2 (Unit 2 Bus G)
14	BG	6.5	Unit 2, Electric Power Train G
15	TF	5.0	Emergency Diesel Generator 1-3 Unavailable Due to Scheduled Maintenance.
16	VI	4.7	Vessel Integrity
17	RA	4.5	Crosstie Unit 1 to Unit 2 ASW
18	BH	3.4	Unit 2, Electric Power Train H
19	SB	3.1	SSPS, Train B
20	SA	2.9	SSPS, Train A
21	RF	2.7	Operator Switches to Containment Sump Recirculation
22	OB	2.5	Operator Initiates Bleed and Feed Cooling
23	LA	2.3	RHR Train A
24	AF	2.3	Vital 4.16 kV AC Bus F
25	LB	2.3	RHR Train B

Note: The top event importances listed are not mutually exclusive.

Figure 3

Table 3.4.2-6. Risk Achievement Worth Ranking of Nonguaranteed Failure Event Tree Split Fractions

Rank	Split Fraction		Importance Measures			
	Name	Description of Item Failed (with Boundary Conditions)	Reference Split Fraction Value	Percentage of CDF with This Split Fraction	Risk Achievement Worth	Risk Reduction Worth
1	FO1	Diesel generator fuel oil transfer system (with loss of offsite power).	1.0e-4	1.5	151	0.985
2	DG1	DC Bus G (all support available).	7.1e-4	9.2	129	0.909
3	DH1	DC Bus H (all support available).	7.1e-4	7.2	102	0.928
4	AW3	Auxiliary feedwater system (support for one motor-driven AFW pump unavailable).	6.8e-4	4.2	62	0.959
5	CC3	Component cooling water system (with 4kV Bus G unavailable).	1.5e-3	6.8	44	0.933
6	CC2	Component cooling water system (with 4kV Bus H unavailable).	1.4e-3	5.2	36	0.948
7	AF1	4kV AC Bus F (all support available).	6.3e-4	2.2	35	0.978
8	AWD	Auxiliary feedwater system (support for turbine-driven AFW pump unavailable).	2.0e-4	0.6	33	0.994
9	AS3	Auxiliary saltwater system (with pump train 1-2 unavailable, but pump trains 1-1, 2-1, 2-2 available).	2.4e-4	0.7	31	0.993
10	OG1	Loss of power to 4kV or 12kV Buses (offsite grid is available).	7.4e-4	0.2	28	0.980

Note: Frequencies are presented in exponential notation, i.e., 1e-1 = 1 X 10^{-1}.

David A. Lochbaum

EDUCATION

June 1979 Bachelor of Science in Nuclear Engineering, The University of Tennessee at Knoxville

EXPERIENCE SUMMARY

03/10 to 11/18 *Director – Nuclear Safety Project*
Union of Concerned Scientists

Responsible for directing UCS's nuclear safety program, for monitoring developments in the nuclear industry, for serving as the organization's spokesperson on nuclear safety issues, for initiating action to correct safety concerns, for authoring reports and briefs on safety issues, and for presenting findings to the Nuclear Regulatory Commission, the US Congress, and state and local officials. Co-authored with Edwin Lyman and Susan Stranahan the book *Fukushima: The Story of a Nuclear Disaster* published by The New Press.

03/09 to 03/10 *Reactor Technology Instructor*
U.S. Nuclear Regulatory Commission
Technical Training Center

Responsible for providing initial qualification and re-qualification training on boiling water reactor technology for NRC employees. Activities included revising chapters of the training manual, conducting classroom and control room simulator training sessions, maintaining the test question database, administering examinations, and assisting the development of an interactive 3-D model of the reactor pressure vessel and its internals.

10/96 to 02/09 *Director - Nuclear Safety Project*
Union of Concerned Scientists

Responsible for directing UCS's nuclear safety program, for monitoring developments in the nuclear industry, for serving as the organization's spokesperson on nuclear safety issues, for initiating action to correct safety concerns, for authoring reports and briefs on safety issues, and for presenting findings to the Nuclear Regulatory Commission, the US Congress, and state and local officials.

11/87 to 09/96 *Senior Consultant*
Enercon Services, Inc.

Responsible for developing the conceptual design package for the alternate decay heat removal system, for closing out partially implemented modifications, reducing the backlog of engineering items, and providing training on design and licensing bases issues at the Perry Nuclear Power Plant.

Responsible for developing a topical report on the station blackout licensing bases for the Connecticut Yankee plant.

Responsible for vertical slice assessment of the spent fuel pit cooling system and for confirmation of licensing commitment implementation at the Salem Generating Station.

Responsible for developing the primary containment isolation devices design basis document, reviewing the emergency diesel generators design basis document, resolving design document open items, and updating design basis documents for the FitzPatrick Nuclear Power Plant.

Responsible for the design review of balance of plant systems and generating engineering calculations to support the Power Uprate Program for the Susquehanna Steam Electric Station.

Responsible for developing the reactor engineer training program, revising reactor engineering technical and surveillance procedures and providing power maneuvering recommendations at the Hope Creek Generating Station.

Responsible for supporting the lead BWR/6 Technical Specification Improvement Program and preparing licensing submittals for the Grand Gulf Nuclear Station.

03/87 to 08/87 *System Engineer*
General Technical Services

Responsible for reviewing the design of the condensate, feedwater and raw service systems for safe shutdown and restart capabilities at the Browns Ferry Nuclear Plant.

08/83 to 02/87 *Senior Engineer*
Enercon Services, Inc.

Responsible for performing startup and surveillance testing, developing core monitoring software, developing the reactor engineer training program, and supervising the reactor engineers and Shift Technical Advisors at the Grand Gulf Nuclear Station.

10/81 to 08/83 *Reactor Engineer / Shift Technical Advisor*
Tennessee Valley Authority
Browns Ferry Nuclear Plant

Responsible for performing core management functions, administering the nuclear engineer training program, maintaining ASME Section XI program for the core spray and control rod drive systems, and covering STA shifts at the Browns Ferry Nuclear Plant.

06/81 to 10/81 *BWR Instructor*
General Electric Company
BWR/6 Training Center

Responsible for developing administrative procedures for the Independent Safety Engineering Group (ISEG) at the Grand Gulf Nuclear Station.

01/80 to 06/81 *Reactor Engineer / Shift Technical Advisor*
Tennessee Valley Authority
Browns Ferry Nuclear Plant

Responsible for directing refueling floor activities, performing core management functions, maintaining ASME Section XI program for the RHR system, providing power maneuvering recommendations and covering STA shifts at the Browns Ferry Nuclear Plant.

06/79 to 12/79	*Junior Engineer* *Georgia Power Company* *Edwin I. Hatch Nuclear Plant*
	Responsible for completing pre-operational testing of the radwaste solidification systems and developing design change packages for modifications to the liquid radwaste systems at the Edwin I. Hatch Nuclear Plant. Also qualified as a station nuclear engineer and covered shifts during startups, control rod pattern exchanges, and other power maneuvers.

OTHER QUALIFICATIONS

January 2010	Certified as a boiling water reactor technology instructor at the U.S. Nuclear Regulatory Commission
April 1982	Certified as a Shift Technical Advisor at the TVA Browns Ferry Nuclear Plant
May 1980	Certified as an Interim Shift Technical Advisor at the TVA Browns Ferry Nuclear Plant

Member, American Nuclear Society (since 1978).

PUBLICATIONS (ABRIDGED LIST)

Books

Fukushima: The Story of a Nuclear Disaster. Co-authored with Edwin Lyman and Susan Q. Stranahan. 2014. The New Press. New York, NY

Nuclear Waste Disposal Crisis. 1996. PennWell Book. Tulsa, OK.

Reports

The Nuclear Power Dilemma: Declining Profits, Plant Closures, and the Threat of Rising Carbon Emissions. Co-authored with Steve Clemmer, Jeremy Richardson, and Sandra Sattler. 2018. Union of Concerned Scientists. Cambridge, MA.

The Nuclear Regulatory Commission and Safety Culture: Do As I Say, Not As I Do. February 2017. Union of Concerned Scientists. Cambridge, MA.

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